IRPA9

1996 International Congress on Radiation Protection

Proceedings

Vienna, Austria, April 14-19, 1996

ganised by the Austrian Association for Radiation Protection

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1996 International Congress on Radiation Protection

Ninth International Congress of the
International Radiation Protection Association

April 14-19, 1996

Congress Center Hofburg
Vienna, Austria

Proceedings / Volume 1

ISBN 3-9500255-4-5

Organised by the
Austrian Association for Radiation Protection

Published by
International Radiation Protection Association
K.E. Duftschmid, IRPA Vice Pres. f. Congr. Affairs,
Chairman Organising Committee

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Printed by Berger, Austria

ISBN 3-9500255-4-5

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IRPA

The International Radiation Protection Association (IRPA) is a world-wide international association made up of individual members who are members of an affiliated national or regional Associate Society.

IRPA now has more than 15.000 individual members in more than 35 Associate Societies which are active in over 40 different countries.

The primary purpose of IRPA is to provide a medium whereby those engaged in radiation protection activities in all countries may communicate more readily with each other and through this process advance radiation protection in many parts of the world. This includes relevant aspects of such branches of knowledge as science, medicine, engineering, technology and law, to provide for the protection of man and his environment from the hazards caused by radiation, and thereby to facilitate the safe use of medical, scientific and industrial radiological practices for the benefit of mankind.

It is a major task of IRPA to provide for and support international meetings on radiation protection. The Congresses of IRPA itself are the most important of these meetings. These have been held about every four years since 1966.

1966	Rome, Italy	1980	Jerusalem, Israel
1970	Brighton, UK	1984	West Berlin, FRG
1973	Washington, DC, USA	1988	Sydney, Australia
1977	Paris, France	1992	Montreal, Canada

For all Associate Societies of IRPA and individual members, it is an important objective to attend the next International IRPA Congress in Vienna. For many other related professions it is an excellent opportunity to communicate their achievements, scientific knowledge and operational experience in radiation protection.

Professional training will be an important part of the Congress programme. Additional to the scientific sessions, the IRPA9 Congress offers a wide selection of refresher courses.

During the Congress the General Assembly of IRPA will be convened. As an adjunct to the business function of this meeting, the Associate Societies Forum will take place.

IRPA and the Austrian host society, with the help of related international organizations and the Associate Societies of IRPA, will provide support mechanisms for a substantial number of qualified colleagues who otherwise would be unable to participate in the Vienna Congress, solely for financial reasons. A particular reason for IRPA to hold the Congress in Vienna is to enable a larger participation by colleagues from the Central and Eastern European regions.

It is my privilege to invite you all on behalf of IRPA to the international IRPA9 Congress in Vienna, Austria.

Charles B. Meinhold, IRPA President

WELCOME ADDRESS TO THE PARTICIPANTS

Dear Colleagues,

The preparations for the IRPA9 Congress in Vienna are finished and the proceedings being printed. We have totally received almost 1000 abstracts from 52 countries, most of the abstracts have been accepted for poster presentation, some also for oral „mini-presentations". The details of the scientific programme have been finalised by the scientific programme committee. It consists of seven plenary lectures, including the „Sievert Lecture" by Dan Beninson, four parallel "Symposia" each day with 2 - 4 invited speakers, who will discuss general aspects and new developments of 16 particular topics in radiation protection. There will be totally 12 "Mini-Presentations", four in parallel on Monday, Tuesday and Thursday, where selected poster authors present the salient points of their work, to be seen afterwards on the poster and discussed during the successive "Posterdiscussions". Friday will be devoted to Chairmen Summaries, which will condense and recapitulate important trends and new developments presented during the congress, to provide a complete overview and the basic message of IRPA9. As during IRPA8, there will be 16 early morning „Refresher Courses", four in parallel every day except Monday. So we can expect an excellent scientific programme, bringing together world-renowned experts with a representative group of today's Radiation Protection Community.

To complete the scientific programme there will be an extensive Technical Exhibition comprising all modern instrumentation in ionising and non-ionising radiation protection by the leading manufacturers from all over the world. The exhibition will be located directly adjacent to the conference rooms and the posters and open during the whole week.

In Montreal we have promised to spend every effort, that IRPA9 will enjoy a much larger participation by our colleagues from the countries of the former "Eastern Block", as it has been possible for the previous IRPA Congresses. As a first step towards this goal, the organising committee, in agreement with the IRPA Executive Council has decided to reduce to conference fee by 50% for participants from these countries. We are aware, that this alone will not be sufficient, and therefore we raised Sponsoring money from different sources, such as IRPA Associate Societies, IRPA8, the CEC, the Austrian government. Thanks to our sponsors we have been able to additionally support about 100 colleagues from the Eastern European Countries.

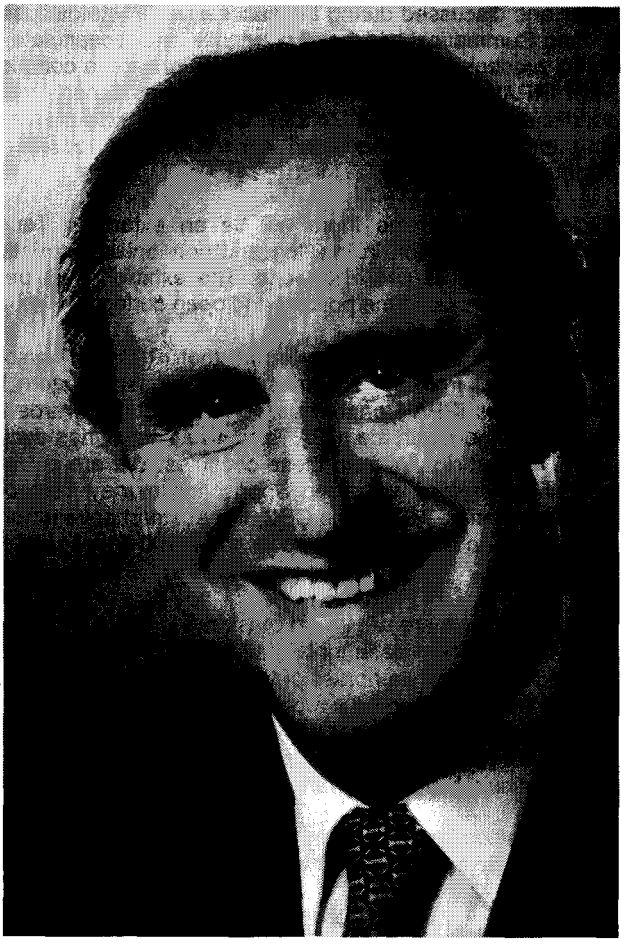
Apart from the scientific events, there will be an outstanding social programme, including a Baroque Concert in the magnificent ceremonial hall of the Imperial Palace, and a special performance of the world-famous white horses of the "Spanish Riding School", shown in our LOGO on the first page. There will also be a reception by the Major of Vienna at the Vienna City Hall and a Congress dinner at a typical Viennese „Heurigen-Restaurant". There will be plenty of other opportunities for cultural events, for the Ladies Programme and for pre- and post-congress tours to particularly beautiful parts of Austria and it neighbouring countries.

So we look forward to welcoming you all in April 1996 in the magnificent halls of the Vienna Imperial Palace and wish IRPA9 every success.

Klaus E. Duftschmid
Chairman, Organising Committee

Herwig G. Paretzke
Chairman, Programme Committee

Under the Patronage of
The President of
the Republic of Austria
Dr. Thomas Klestil

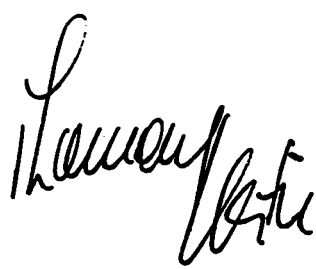


Welcome address to the participants

I greet all participants to the 9th Congress of the "International Radiation Protection Association" and I welcome them very warmly to the Congress Centre at the Vienna Hofburg Palace. I congratulate the Austrian Radiation Protection Association (ÖVS) on the honourable task of organising this important convention.

Ever since nuclear energy was first applied for peaceful purposes, radiation protection has assumed increasing importance for the general public. In this respect the Austrian Association, in cooperation with the other members of IRPA, fulfils the important responsibility of raising the awareness of the public by providing objective and honest information. In addition, I also would like to thank you for your important work towards the task of deducing, from the body of scientific knowledge, the appropriate suggestions on how the protection of the population can best be guaranteed above all by means of effective legal provisions in the various countries.

I therefore wish this 9th Congress of the IRPA every success in its debates, analyses and resolutions!



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RISK OF RADIATION AT LOW DOSES

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1. INTRODUCTION

Risk and risk sources have been increasingly studied in recent years. The essentials of risk consist of a combination of the idea of loss with that of chance or probability. The idea of chance is crucial: the inevitable can be utterly unpleasant but, lacking the element of chance, is not a risk.

Even without analyzing the different components of the concept of "loss", it should be recognized that to be exposed to risk is not necessarily bad. The achievements of modern life imply the exposure to several sources of risk, and past evolution would have been impossible without the risk incurred by our ancestors.

A special type of risk, pertinent to our discussion, is exemplified by the health threats due to low levels of natural or man-made chemicals and low radiation levels. It constitutes a risk very difficult to analyze, not because the effects are unknown but because they are already very familiar, and exposed groups only manifest a slightly increased frequency of such effects.

2. RADIATION RISK

At high doses ionizing radiation is clearly detrimental, the scene being dominated by the deterministic effects, e.g. death via the acute radiation syndrome. There is no doubt here of the causal relation between radiation exposure and effect. At somewhat lower doses, deterministic effects are not produced but, if the exposed group of individuals is large enough, a clear increase of induction of cancer over the spontaneous rate can be demonstrated. While the relationship between radiation and cancer is quite clear in these cases, it is not possible to state with certainty if a given individual will be affected or if a given case of cancer is the result of the exposure.

At even further lower doses, the observed relationship between radiation and cancer blurs due to increasingly larger uncertainties, reaching a point where an effect, if it exists, can not be detected. Many discussions have stemmed from this fact, where defenders of the existence of a threshold have claimed that no effect exists at all below such doses. This, of course, could be true but certainly not because of the lack of observation.

2.1. STATISTICAL DETECTABILITY AND CLAIMS OF THRESHOLD

Even assuming a non-threshold linear relation between risk (here used in a loose way meaning probability as the considered effect is only cancer) and dose, the required number of individuals, N, incurring a dose D, for achieving detectability increases steadily with a reduction of dose. If all other influencing factors are kept constant, the excess number of cancers attributable to radiation and its standard deviation are given by

$$\text{Excess} = rDN$$

$$\sigma = \sqrt{2bN + rDN}$$

where b is the "natural" risk of cancer, appropriate to the group under study, and r is the risk per unit dose in the group.
In order to be detectable the excess must be larger than a stipulated number of standard deviations (usually two, for a level of significance of about 95%). Therefore

$$rDN \geq 2\sqrt{2bN + rDN}$$

In most cases, the "natural" cancer risk is substantially larger than rD and therefore (2b + rD) is practically constant. It follows after a simple algebraic manipulation that, for that stipulated level of significance,

$D^2 N \geq$ constant

For example, if a given type of cancer has been shown to be related to radiation in a group of a few thousands having incurred a dose of the order of 1 Gy, then to show the same relation with doses of the order of 100 mGy one would require groups of a few hundred thousand individuals.

This argument is simplistic as it ignores most of the complicating factors involved in epidemiological studies but is sufficient to dismiss most of the reported efforts to prove significant thresholds. On the other hand, it must be recognized that epidemiological studies at the lower dose, specially those of cancer types of smaller "natural" incidence can contribute to the progress of our knowledge, but extreme prudence is required when the results are negative.

2.2. ANIMAL EXPERIMENTS AND THE DOSE-EFFECT RELATION AT LOW DOSE

Experiments with animals, usually small, offer the possibility of increasing the number when necessary and to plan the exposures in order to cover the required range. Obviously the results are not directly applicable to man (for example the observed slopes), but general information can be derived on the shape of the dose-effect relationship and on the action of parameters thought to influence the relationship.

On the other hand, the same problem of detectability appears at low doses. To improve (but not solve) the situation, the experiments would require, as it is usually stated, too many "mice", too much money and too much time to be practicable. Even if the experiments were possible, they would always leave a region of dose where direct observations are not available, and it is that region that interest us the most. The main issue is how valid are the extrapolations of results observed at higher dose to the dark region.

2.3. EXTRAPOLATION FROM DOSES AT WHICH OBSERVATIONS WERE MADE

Sometimes it is claimed that searching for the function that best fits the observations would provide the solution to the extrapolation problem. This, of course is nonsense: there are infinite functions that would pass exactly through all the points representing the observations.

In order to carry out regression analysis, it is necessary *first to select the functions to be tested.* It is then possible to make conclusions about the goodness of the fit, to compare the two or more pre-selected functions, etc.. It seems, then, unavoidable to conclude: a) neither epidemiology nor animal experiments will establish the shape of the dose-effect relationship at low doses; b) if a shape is selected on other grounds, they will help in obtaining values of the parameters applicable to man and in testing the model when any new datum is obtained below the existing set of data.

2.4. CELLULAR AND MOLECULAR RADIOBIOLOGY AND THE MODEL DOSE- EFFECT RELATIONSHIP

One common criticism against the use of modeling of the dose-effect relationship (on the basis of the results of fundamental radiobiological research) is that the predictions made by such modeling are "unscientific", because of the lack of epidemiological data in the region of doses of the prediction.

This criticism is in itself utterly unscientific. It is sufficient just to mention the discovery of a new planet and the observation of the predictions of relativity to dispose of the criticism. While some natural science is description of what is observed, most of it is the blend of modeling from some observations, predictions sometimes leading to other observations, theoretical constructions and searching for new and crucial experiments.

A very wide knowledge exists on the effects of radiation on cells. At present the consensus is that for all effects of interest the target of radiation is the DNA. Energy deposition by ionizing radiation occurs by ionisation and excitation. About half the energy deposited in the cells is due to

excitation, but this is of less consequence of ionisation. Energy deposited in DNA affects the molecule either by direct ionisation or by the action of free radicals generated by ionisation in the immediate vicinity.

The immediate effects of such energy deposition are the loss or damage of one of the bases or a segmental loss in the DNA molecule. Sophisticated and efficient repair mechanisms become operative and usually cancel the effect, except in a small proportion of cases, resulting in what is called a misrepair.

The existence of the natural background of radiation reduces the importance of the dose-response relationship at doses close to zero. Almost all the data on stochastic changes in cells, irradiates "in vitro" with low LET radiation can be summarized and interpreted as follows:

a) at low doses (and even at higher doses but with low dose rates) it is very unlikely that more than one ionizing event will occur in the relevant parts of DNA within the time the repair mechanisms operate. Taking account of the Poissonian distribution of ionizing events, the small exponent involved and a small fraction of misrepairs, the dose-response relationship will be linear, as in fact it is.

b) at higher doses and dose rates, two ionizing events may be able to combine effects before the repair mechanisms could cancel the effect of the first event, producing an enhanced probability of DNA transformation, which is reflected by a dose quadratic term in the dose-response relationship.

Obviously, there is quite a distance between transformed cell and clinical cancer. There is, at present, consensus that cancer initiates in a single cell. When the stem cells of a tissue are irradiated, more than one transformation is likely to occur and the number of such transformations is a Poissonian random variable with an average of NP, where N is the number of stem cells and P the probability per cell of transformation. In turn this probability is a linear-quadratic function of dose.

It can be shown that, provided the transformed cell has a developing advantage (somewhat shorter division time), the over all probability that at least one transformed cell results in an established clone that would grow without bound, is also related to dose by a linear-quadratic relationship.

It should be noted that as the dose increases, another cell effect becomes competitive with transformation: interference with cell division and cell death. This would result in a decrease of the probability of inducing cancer.

2.5. LINEAR-QUADRATIC RELATIONSHIP AND EPIDEMIOLOGY

Good epidemiological results (at high doses and dose rates) correspond presumably to the region of dose where the effects are most probable. It is interesting to predict the location of this region of dose using the linear-quadratic relationship:

$$P = (a\,D + b\,D^2)\,e^{-cD}$$

where p is the probability of cancer, a and b are constants and the factor e^{-cD} is the survival fraction of exposed cells.

Deriving and equalizing to zero, the following expression can be obtained:

$$cD_m = \frac{\frac{a}{b} + 2D_m}{\frac{a}{b} + D_m}$$

where D_m is the dose that maximizes the probability of induction of cancers.

Without indulging in discussions of the values of a and b, one can take two extreme cases: in one the ratio $\frac{a}{b}$ is assumed to be vanishingly small compared to D, and in the second ratio $\frac{a}{b}$ is assumed to be very much larger than D_m. For these two cases the product cD_m would tend to the values of 2 and 1 respectively. The cell killing coefficient c has been experimentally measured for many tissues, and for humans a value of 1 Gy^{-1} can be taken as typical.

It follows that the region of dose with good epidemiological results is predicted to be between one and two gray, in very good correspondence with reality.

A very important issue in the evaluation of the risk (probability of attributable cancer death) per unit dose at low doses, is the extrapolation to the low dose region of the epidemiological observation at high doses and dose rates. A usual procedure is firstly to assume a straight line between the observation and the origin of the coordinates and then divide the resulting slope by the so-called Dose and Dose Rate Effectiveness Factor.

In terms of the linear-quadratic relationship, the risk (probability of attributable cancer death) at a high dose D with high dose rate extrapolated linearly to the origin would give a slope of a + bD, and the DDREF is given, therefore,

by the ratio $\frac{a + bD}{a}$

$$DDREF = 1 + \frac{b}{a} D$$

It can be observed that the factor will increase linearly with D, the dose at which the epidemiological results apply. At typical values, where the linear component of the relationship contributes to the probability about the same than the quadratic in the vicinity of one gray and taking the range of observations as one to two grays, the factor appears to be in the range of two to three. This range of values agrees well with many reported human data. Animal experiments that have a wider range of factors, have also a range of doses greater than the human experience.

2.6. CRITICISM OF THE LINEAR NON-THRESHOLD RELATION

As always with emerging solutions of scientific issues there is a main stream of consonant opinions and voices of dissent. The dissent is sometimes a valid scientific discussion but in other cases reflects an assortment of gut feelings, reactions to public opinion and even interests.

a) Scientific discussions

Of the many issues raised, two recent ones are dealt with here. One stems from the genetics of cancer development and the other from consideration of the "adaptive" response to radiation.

It has been shown that several mutations are required for transformation and acquisition of malignancy of given cancer types. If this is true, the argument goes, then radiation cancer probability should be strongly curvilinear with dose with negligible risk at low doses. If the target for each mutation requires at least one ionizing event then the probability of mutation can be expressed as $[1 - e^{-kD}]$ and for similar n targets the overall probability P will be given by

$$P = [1 - e^{-kD}]^n$$

With usual values of k (mutation rate per unit dose) and with n having reported values ranging from 2 to 7, the argument seems quite correct. However it should be remembered that there are also "spontaneous" mutation rates for the same targets. These rates must be substantial to account for the cancer frequency prevailing in man. The radiation attributable cancer probability is then given by the difference

$$\Delta P = [1 - e^{-(St + kD)}]^n - [1 - e^{-St}]^n$$

where S is the rate of spontaneous mutation of a target and t the age. Two basic concepts emerge from the analysis and graphical representation of the above expression: a) provided St is substantially larger than kD, then the radiation attributable risk AP appears to be linear with dose. b) it is necessary to have important spontaneous mutation frequencies to experience radiation risks at low doses. Our risk values per unit dose would then be valid for our present environment.

Another scientific argument against the linear-quadratic relationship (which at low doses or low dose rates becomes the linear non-threshold relation) relates to the denominated adaptive response to radiation. It has been shown by experiments involving irradiation following a pregiven dose, that repair mechanism can be stimulated and the repair rate increased. This, it is claimed, would completely change the shape of the relation at low doses.

The issue is very complex. An increased rate of repair could also increase the rate of misrepairs, being the misrepairs a fraction (small) of the repairs. In an extensive analysis, UNSCEAR has concluded that "Extensive animal experiments and limited human data provide at present no evidence to support the view that the adaptive response in cells either decreases or increases human risk at low doses".

b) Other types of criticism:

It is difficult even to attempt to classify all the non-scientific criticism raised against the linear non-threshold relation. In most of them one can find elements of arrogant ignorance, apparent concern for the peace of mind of the public and gut feelings.

In many cases the criticism is only one component of a larger "defense" of a particular risk source. This is particularly the case of nuclear power, which does not need nor it deserves these self-appointed defenders. Even the more honest types of such defenders indulge in statements such as "if the public would just know the facts (of course not presented as the radiation protection community would present) then"...

Some criticisms are really requests for "putting the risk in perspective", referring to a risk source, usually nuclear power. Since a risk source has many attributes, the comparison must involve comparable attributes. An essential fact, many time ignored is called the principle of "ceteris paribus", which means that all factors which are not explicitly presented in the risk characterizations must be mutually equivalent, in a valid risk comparison.

3. CONCLUDING REMARKS

The linear non-threshold relationship, is at present the best tool to predict the risk probability of radiation at low doses. It fulfills all the requirements to be considered "realistically representative", using modeling terminology.

Practical decisions can be made under this relationship, and the radiation protection system recommended by the ICRP provides a method for such decisions.

ONE HUNDRED YEARS OF X RAYS AND RADIOACTIVITY-- RADIATION PROTECTION: THEN AND NOW

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It is not particularly remarkable that this topic was one that I chose for an IAEA Symposium which I presented in 1974 (1). My thesis in 1974 was that the basic recommendations and regulations on dose limitation were unchanged from the early 1920s to the date of that lecture. What is remarkable is that during the middle to late 1970s the basis for such recommendations changed to a scientific approach based on risk, and as a result, the recommendations have been under change and modification ever since, although perhaps, as we will see, we may be at a point of some stability once again. I will return to the historical developments, particularly relevant during this Congress when we are celebrating the discovery of the X ray by William Conrad Roentgen just 100 years ago this past November, and the discovery of radioactivity by Becquerel 100 years ago last month. You should also understand that much of this presentation will focus primarily on activities of both the National Council of Radiation Protection and Measurements (NCRP), with particular emphasis on the International Commission on Radiological Protection (ICRP).

Of course, man has evolved in a sea of ionizing radiation. Enhanced exposure to natural radiation took place the first time man moved to a cave where the radon progenies were there for him to inhale. The first occupational exposure that we can trace back in recorded history was to the miners of Joachimsthal and Schneeburg in the 15th and 16th Century Czechoslovakia and Saxony who developed lung cancer from breathing radon progeny while mining for lead (2).

In the middle to late 1850s, gas-discharge physics was a hot topic and the source of wide-ranging experiments in virtually every physics laboratory. These tubes could be found in every high school science laboratory and in any university physics laboratory. On November 8, 1985, Wilhelm Conrad Roentgen was working in his Würzburg laboratory with a Crookes discharge tube. As he was adjusting the high voltage on his gas-discharge tube that he had covered with dark cardboard, he saw his screen of florescent materials lying on the table nearby fluoresce. He realized that he was observing the results of a highly penetrating ray, which he called the X ray. He spent the next two months carefully investigating in detail the properties of this new X ray. During this period, he discovered virtually all of the classical physics properties of the X ray. During these two months he told no one about this discovery except for an anecdotal story which relates that his wife was complaining about his missing meals and being extremely introspective and uncommunicative. Roentgen reportedly took her to his lab where he took an X-ray photograph of her hand -- to her complete astonishment and to his great relief -- he was not, after all, losing his mind! He submitted a paper describing his observations in less than 60 days, during December of 1895 (3). The results of his work were reported in the popular press in Vienna on January 5, and in London and New York by the middle of January 1896. Everyone who owned a gas-discharge tube learned that if they applied high enough voltage they could generate X rays. Thomas Alva Edison was one of the first to see the potential commercial applications of these X rays. For example, in early February, he began a highly publicized attempt to X ray the human brain. Edison had hoped to market an X-ray light bulb, but eventually came to understand the inherent dangers associated with such practices when his assistant, Clarence Dally, died in 1904 as a result of his excessive exposures (4). Dally's death, which was widely reported, had a sobering effect on all of those who were using X rays. In fact, Edison completely stopped working with X rays at this point, although he had already developed a hand-held fluoroscope (5).

When Antoine Henri Becquerel learned of Roentgen's discovery of the X ray using fluorescent materials, he was determined to study these processes in more detail. The material Becquerel chose to work with, fortunately, was a double sulfate of uranium and potassium, which he exposed to sunlight and then placed on photographic plates wrapped in black paper. When developed, the plates revealed an image of the uranium crystals (6). His conclusion was that "The phosphorescent substance in question emits radiation which penetrates paper opaque to the light." He believed that the sun's energy was being absorbed by the uranium which then emitted Roentgen's X rays. However, because the weather was poor on the 26th and 27th of February, Becquerel returned to a desk drawer the uranium-covered plates that he had intended to expose to the sun. On the first of March, when he developed these plates, he expected only very faint images. To his surprise, however, they were clear and strong.

He now realized that the uranium itself was emitting radiation without an external source of energy, and he had discovered radioactivity (7). All by the first of March 1886.

Marie and Pierre Curie, quickly realizing the importance of Becquerel's findings, separated the uranium from pitch blend and eventually found the elements radium (8) and polonium (9), which they laboriously separated from the ore over a period of four years. By 1902, they had a tenth of a gram of radium. During this period, Henri Becquerel had obtained a sample of radioactive material from the Curies, which he placed in a waistcoat pocket. He observed that having worn this waistcoat for less than six hours, he had received a deep burn on his chest (10). He recognized that if this could be destructive to healthy tissue, it should also be destructive to cancerous tissue. As a result of his and the Curies' work, radium followed the same path as X rays in the development of both the medical and nonmedical use of radiation.

By and large, it was the medical community that recognized the enormous potential of the X ray and radium. It was interesting that medicine at that time was dealing with a difficult problem of the use of electrotherapy. Although this practice was being discouraged by the medical community as a whole, the practitioners were still there, and their equipment was ideally suited to the generation of X rays.

During the next few years, medical use of the X ray expanded rapidly, and indeed, this became known as the era of "bullets, bones, and kidney stones." The physicians realized from the beginning that while the medical benefits were unlimited, there were potential hazards from radiation exposure. There were reports in the scientific literature and in the popular press of ulcers that did not heal and scores of skin burns, both among the patients and the physicians (11). The first ulcerating skin lesion was reported by an electrotherapist named Grubbé on January 26, 1886, within a month of the discovery of the X ray (12). By 1915, only 15 years after the introduction of the X ray, both the German Radiological Society and the British Radiological Society had prepared recommendations for physicians on avoiding unnecessary exposure (13). Although these rules were not very definitive, they demonstrated that the societies understood that there was a problem.

As indicated above, the medical community had adopted this technology, and once a medical association takes ownership of a modality of this kind, they tend to protect it as their own. In the United States, and pretty much in England and in France, a physicist could not publish an article unless he had a physician sponsoring the paper. As a result, most of the literature was related to clinical effects and to clinical use. The situation was different in Germany, where physics and medicine grew up together, and the medical community embraced the physics community as its equal. This was primarily because medicine was more heavily regulated in Germany than it had been in these other countries.

Protection advice was not heavily organized until, in 1921, the newly organized X-Ray and Radiation Protection Committee in England presented a set of detailed recommendations as rules that every physician was expected to use (14). The pressure for these recommendations resulted from the development of the hot cathode tube by Coolidge, an engineer at General Electric (15). This tube was able to produce much higher currents and much higher energies. Many of the radiologists now recognized the significant hazard that the use of this equipment posed for them and their patients. Second, World War I had just ended, and hundreds of X-ray machines, mostly with the new Coolidge tube, had been used in the battlefield and were implicated in the many reports in the public press about anemia in the returning soldiers.

It is interesting to note that these military X-ray machines had an enormous impact on the course of radiation measurements as well. The Army Quartermaster Corps wanted to be certain they got what they paid for, i.e., these battlefield machines had to meet military standards. As a result, the National Bureau of Standards was called upon to provide standards, and the physicists involved became more interested in measurement and quantification than had the physicians who had depended upon the redness of skin and whether or not they obtained a good image (16).

Radium commerce also had an impact on measurement. The only way one could specify the quantity of radium was through detailed measurement, which at $100,000 per gram, was very important. Commerce ensured that, at last, there was attention being paid to the measurement of radiation and radioactivity.

In 1922, Mutscheller, in the United States, and Sievert, in Sweden, were concerned about the adequacy of radiation protection. Mutscheller visited a number of well-run clinics in New York City and found that they could operate quite well without anyone being exposed to more than .01 of an erythema dose in 30 days (17). The erythema dose, which is the dose to cause reddening of the skin, had become a common measure of exposure

at this time, primarily since there was no generally accepted physical measurement. Nearly every X-ray operator knew how long it would take to develop an erythema at given locations around their X-ray facilities. At the time Mutscheller made this recommendation, Sievert, in Sweden, arrived at a recommendation of .1 erythema dose in a year (18). It is remarkable that these two independent investigators ended up with virtually the same number. Inherent in their recommendation is the concept of a threshold dose. For example, Mutscheller stated, "for in order to be able to calculate the thickness of a protective shield, there must be known the dose which an operator can, for prolonged period of time, tolerate without ultimately suffering injury." Mutscheller's assumption of a "tolerance" level is consistent with the classical threshold response curve so common in toxicological studies. In fact, it is the kind of relationship we see now in most toxicological studies.

In 1925, the International Congress of Radiology at its meeting in London, formed an X-ray unit committee which was to become the International Commission on Radiation Units (ICRU) (19). Even at the time of formation, the international society recognized the need for an internally accepted definition of an exposure quantity. In 1928, the International Congress held in Stockholm adopted a recommendation from this new committee that defined the Roentgen as "the exposure when the X- or gamma- ray field produces 1 e.s.u. of positive charge and 1 e.s.u. of negative charge in 0.00129 grams of air" (20). This definition remained essentially unchanged for 50 years.

At the Stockholm meeting, the International Congress formed the origins of the International Commission on Radiological Protection, the Advisory Group on X-ray and Radium Protection. The U.S. representative to that meeting was Dr. Lauriston S. Taylor, of the National Bureau of Standards. Dr. Taylor was instructed to return to the United States and form a similar organization for the United States so that they could bring a unified position to the future meetings of International Congresses. Taylor returned to the United States and formed the origins of the National Council on Radiation Protection (the U.S. Advisory Committee on X-ray and Radium Safety). Lauriston Taylor was to chair this advisory committee and its successor organizations, the National Committee on Radiation Protection and Measurements and the National Council on Radiation Protection and Measurements, for 49 years until his retirement in 1977.

Shortly after the ICRU provided the definition of the Rontgen, both the International Commission on Radiological Protection (ICRP) and the National Council on Radiation Protection and Measurements (NCRP) made recommendations dealing with exposure levels. The ICRP recommended no more than .2 R/day (21). This is a reasonable measure of the exposure that would result in about .01 of the erythema dose in thirty days. What they had done was to adopt, in a way that could be measured, what Mutscheller and Sievert had recommended earlier. This recommendation, although quantifiable, was still based on skin reddening. Three years earlier, in 1931, the NCRP recommended .1 R/day (22). The ICRP recommendations applied to measurements made at the surface of the body, while the NCRP recommendations applied to measurements made free in air. Measurements of exposure made at the surface of the body with low energy X rays would indeed be just about twice what they would be free in air. In fact, the NCRP and the ICRP recommendations provided virtually the same level of protection.

Dr. Failla noted, in the 1960 hearings before the U.S. Congress' Joint Committee on Atomic Energy, that he endorsed a limit of .1 R/day based on his observation that two technicians who received that level of exposure showed no observable effects and this limit could thereby be judged to be safe (23).

In the middle 1920s, there were a number of young women working as radium dial painters in New Jersey and elsewhere who tipped their brushes between their lips -- the famous radium dial cases. A New York dentist, Theodore Blum, noted in a three-line footnote to a paper on osteomyelitis of the jaw that he had seen what he termed "radium jaw" in a girl working in a New Jersey dial-painting plant (24).

Much of the early attention to the dial painters came from the National Consumers League, which began under Florence Kelly, and became a virtual crusade (25). By the end of 1926, most of the dial painting intakes had stopped; however, the medical and quasi-medical use of radium and its emanation products were booming. In 1932, a prominent steel executive named Eben Byers, who was a well known amateur golf champion, died of excessive use of a patent medicine, Radithor. Since each one-half bottle contained one microcurie of 226-Radium and one microcurie of 228-Radium, it is not surprising that Mr. Byers' habit of ingesting four bottles per day over an extended period of time resulted in radium poisoning (26). The Los Angeles County Health Department simply could not understand how such a thing could be happening in California, so they went to the California Institute of Technology, where they were put in touch with Robley Evans. This began a long and careful analysis of the effects of radium in bone. By 1941, Evans had studied twenty-seven cases of Radium ingestion, and noted

that there were seven cases with residual body burdens below 0.5 micrograms of Radium and no injuries, and 20 with 1.2 to 23 micrograms with various degrees of injury. He presented this data to the Advisory Committee on X-Ray and Radium Protection. Their consensus opinion was that they would accept Dr. Evans' suggestion of .1 microgram (.1 microcuries) of radium as a level "we would feel perfectly confident if our wife or daughter were the subject" (27). This value was published in NBS Handbook 27, March 2, 1941.

Eisenbud has made the point that I will reiterate here, that it was remarkably fortuitous that, before Pearl Harbor and just after the discovery of plutonium, the community had at its disposal two recommendations, an external exposure limit of .1 R/day and body burden limit on internally deposited radium of .1 ug Ra. Without these numbers, it is hard to imagine what the consequences to workers might have been during the Manhattan Project.

During the Second World War there was extensive research in radiation biology going on in places like Oak Ridge, the University of Rochester, the University of California at Berkeley, and the University of Washington, to try to obtain information on the effects of ionizing radiation. Data was obtained on dose and dose rate effects, depth dose, R.B.E.s, radionuclide metabolism, and dosimetry. Perhaps the most influential radiation protection recommendation to come out of this work was that developed by a committee at the Tripartite Conference Meetings held among scientists from Canada, the United States, and Great Britain, countries with access to extensive wartime data (28). They brought their recommendations to the ICRP and the NCRP in the late 1940s. By the middle 1950s, both the NCRP and ICRP had produced new sets of dose limits derived from all the data obtained during World War II (29,30).

They recommended 600 mrem per week for the skin, and 300 mrem per week for other organs. I was fascinated to realize that .1 R/day is .6 R/week, which is 600 mrem per week, which means that the 600 mrem per week for the skin is based on the .01 of the erythema dose of 1928. The 300 mrem per week limit is more interesting. If the body is irradiated with 150 kV X rays, the dose at a depth of 5 cm is just about half of that at the surface. If protected by a limit of .1 R/day with soft X rays, the limiting dose to the organs at 5 cm would be .05 R/day. If, however, the body is irradiated with high-energy gamma rays, and the same level of protection is desired as that with 150 Kv X rays, then the limit for the skin must be 600 mrem/wk (.1 R/day) and one half of that value or 300 mrem/wk (0.5 R/day) for the organs taken to be at 5 cm.

Starting in about 1954, we entered a new era characterized by weapons testing and the public response to it. Perhaps one of the most important contributors to the public's fear of radiation can be traced to the worldwide reaction to the fallout from the Bravo Weapons Test on Bikini in March 1954. The subsequent plight of the crew of the Lucky Dragon fishing vessel made headlines, and was coupled in the U.S. with the *Life* magazine cover published on April 29, 1954 depicting, for the first time, a thermonuclear explosion. Now people all over the world became concerned about radioactive fallout. Specifically, there were two individuals in the U.S. who led the scientific community in expressing concern: Mueller, a geneticist, who had been speaking about the linearity of genetic effects even during the late 1930s, and Linus Pauling, who worried about internal exposures. As a result of the public concern about fallout, a National Academy of Sciences Committee on the Biological Effects of Atomic Radiation (BEAR) in the United States and the Medical Research Council (MRC) in the United Kingdom were asked to review the radiobiological data (31,32). Both committees came up with about the same estimate of detriment, having focused their attention on genetics. They said that it was unlikely that all of man's suffering and pain from genetic abnormalities came from natural radiation background, but that some of it did. Such a consideration bracketed the genetic risk since they knew the natural radiation background levels and the natural incidence of genetic effects. Based on this analysis, both committees came up with an estimate that suggested individuals (workers) should not receive more than 50 rem to age 30 and another 50 rem to age 40. (The MRC actually recommended 50 R to age 30 and 200 R lifetime). For the population the BEAR Committee suggested a limit of 10 rem to age 30 for all exposure except natural background. I might add that I was able to discuss this with Eugene Cronkite many years ago. Dr. Cronkite was Chairman of the Hematological Effects Subcommittee of the BEAR Committee at the time of the preparation of the 1956 report. I asked him if the recommendations on exposure limitation came from considerations of the radiologists who had been shown to have an excess incidence of leukemia. He answered that the dosimetry was so uncertain that they could not estimate the dose nor the risk per unit dose associated with leukemia among the radiologists. He noted that what they did decide was that they would accept the genetic panel recommendations, and the Academy recommendations were therefore based almost entirely on the genetic estimates based on a linear extrapolation.

The NCRP and ICRP had to decide the way in which they would recommend that the worker be protected under these new recommendations (33,34). As we know, the answer was an annual limit of (age - 18) x 5 rem,

which delivered 60 rem to age 30, etc. The whole-body limit was 3 rem/quarter and (age - 18) x 5 and 15 rem/year for individual organs. By the way, 300 mrem/wk for 50 weeks results in 15 rem/year. Again, the organ limit of 15 rem finds its way back to .01 of the erythema dose in 30 days.

As noted above, my thesis on this subject in 1974 was that there was not a very strong scientific basis for our dose limits. However, this situation changed dramatically by 1977. This was a result of information that came, not in 1977, but from the period 1960-77, and was based primarily on data that was becoming available from the Japanese survivors of the atomic bombing of Hiroshima and Nagasaki who had been under study from the time of the bombs. This study is performed by the Radiation Effects Research Foundation (RERF) under sponsorship by the U.S. Department of Energy (DOE) through the National Academy of Sciences and by the Government of Japan.

I would like to stop here for a moment because everyone should understand the enormous contribution those survivors and the government of Japan have made by their continuing participation in this study. I should add that funding for continuing this important work is now is in question by the U.S. DOE, and it is incumbent on us all to see if we can help to maintain it and to support the absolute necessity for the RERF Directors and Scientific Councillors to set the research agenda.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the National Academy of Science's Biological Effects of Ionizing Radiation (BEIR) Committee review the data that comes from RERF. The UNSCEAR was a product of the same issue that brought about the 1956 NAS BEIR Committee: worldwide fallout from atmospheric nuclear weapons tests. It was created by the United Nations General Assembly in December 1955. The UNSCEAR noted that in 1962 the incidence of solid cancer in the Japanese survivors was slightly greater than might have been expected in that population if it had been unirradiated, but that excess leukemia was clearly evident (35), and in 1964, they estimated that other cancers were about equal to leukemia (36). In the early 1970s, they estimated that other cancers were about two times leukemia (37). In 1977, the UNSCEAR provided a fatal leukemia risk estimate of 2×10^{-5} per rem and a total fatal cancer risk estimate of 1×10^{-4} per rem, i.e., the solid tumor risk is about five times the leukemia risk (38).

Based on its own review, the ICRP adopted in 1977 total nominal risk of fatal cancer of about 1×10^{-4}/rem (39). They then compared this radiation risk with the average risk of accidental death in safety industries. In safe industries at that time, one person in ten thousand died each year (1×10^{-4}/year) from accidents, and the ICRP suggested that the radiation workers ought to have at least that level of protection. The ICRP then set a limit of 5 rem/year on the expectation that most people who were protected by a limit of 5 rem/year would be unlikely to exceed 1 rem/year, and, therefore, the average risk fatal cancer will be the same as that for workers in safe industries. In addition, the ICRP suggested that the annual limit on intake (ALI) of radionuclides be based on the specific fatal cancer risk of each tissue results from that intake over the next 50 years. Inherent in the total risk approach is the need to combine internal and external radiation.

The recommendations of the ICRP Publication 60 are based on further changes (40). In 1986, a later set of data from Japan became available which suggested two things. First, there is evidence of increased risks based on new dosimetry and some additional solid cancers. This new data also gave further evidence that cancer from exposure to radiation follows a multiplicative projection model, i.e., attributable cancers will occur at the age they would if there were no exposure, so it isn't until people approach their mid-seventies that these cancers are likely to occur. The ICRP and NCRP have adopted this new risk projection model. Having such a model is needed to estimate what is going to happen to the Japanese survivors over the next 40 years or so. The ICRP and the NCRP had both used an additive model prior to 1990. It is very clear from the Japanese survivor data that exposure to radiation at high dose rates results in excess cancer. You will note "high dose rate" since the doses that show these excess cancers are about 1 Sv, but 1-2 Sv is on the order of the lifetime exposure we might expect for the most highly exposed radiation workers. Therefore, we are talking about an extrapolation from high dose rates to low dose rates, and we must ask the question of whether there is time for recovery and repair which might alter our estimate of risks at lower dose rates. ICRP's Task Group on Risk, chaired by Dr. Arthur Upton, suggested you might be able to reduce estimates from very high doses (dose rates) by about a factor of two to get the best estimate in the risk at low doses (low dose rates) (41). The NCRP Committee on Risk, chaired by Dr. Michael Fry, suggested the risk at high doses (dose rates) could be reduced by a factor of two to three (42). What all this means is that although we now are on a very firm basis in stating that there is excess cancer in the Japanese, we still have concern about whether we are overestimating the risk by a factor of two or three, or underestimating it by about the same factor. But at least this gives us confidence that we have a fairly firm understanding of the risks that people face. In fact, the latest data from the former Soviet Union suggest that this reduction factor

might be about three (43). As we apply these risk estimates to deriving dose limits, the ICRP and the NCRP recognized that the risk estimates had increased by about a factor of four since 1977, when ICRP Publication 26 was published. Since the annual limit was 5 rem in 1977, they might logically have been expected to divide by four and obtain a new limit of 1 rem/yr. The ICRP did note, however, that the new projection model also changed the most likely age of death from an attributable cancer. That changed from an expectation of death in the middle sixties to expectation of death in the late seventies. As a result, the ICRP felt it was important to base the limit on the risk to the most highly exposed individuals (for whom the limit is needed). In this regard, they also noted that the risk of accidental death in industry has been decreasing by ~2% per year. "Safe" industries are now at ~5 x 10^{-5} rather than 1 x 10^{-4} yr^{-1}. Rather than using the safe worker criteria, the Commission felt that it was more appropriate to base their limits on a comparison with an individual worker at the upper end of safe industry risks. This turned out to be about 10^{-3}/year. On this basis, the ICRP recommended a limit of 100 mSv over 5 years and the NCRP recommended a limit based on age in tens of mSv, i.e., if you are 45, you shall not have a cumulative dose >450 mSv (45 rem).

These approaches are tolerable for the rare individual operating at the dose limit, but are totally unacceptable to use for any kind of average exposure for individuals who are working in the industry. It is for this reason that both the NCRP and the ICRP stress that the dose limits themselves are entirely unsatisfactory as a basis for designing a protection system and that optimization should be the focus of our efforts.

The data on exposure to workers and the general public demonstrate the remarkable effectiveness in the application of the optimization philosophy. We can rest assured that the breathtaking advances in medicine and industry can flourish for the benefit of all mankind.

It is only the fear of radiation engendered by incidence the fallout from atmospheric weapons testing (the Lucky Dragon incident), reactor accidents (Three Mile Island, Unit 2), and reactor disasters (Chernobyl) which threaten to derail this remarkable resource. It is essential that those of us in the radiation protection sciences begin to understand public perception and public value so that we can be active and effective participants in public decision-making efforts.

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ASSESSMENTS BY UNSCEAR OF RADIATION SOURCES AND EFFECTS

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ABSTRACT

Since 1955, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has evaluated the exposures of the world population to the various sources of ionizing radiation and from available radiobiological and epidemiological data has assessed the health effects and risks of radiation. The latest scientific evaluations of the Committee were published in the UNSCEAR 1993 and 1994 Reports. In this paper a summary is presented of the main results of analyses in each of the scientific annexes of these reports.

EXPOSURES FROM NATURAL SOURCES OF RADIATION

The assessment in the UNSCEAR 1993 Report [2] of the annual effective dose from natural sources of ionizing radiation in areas of normal background has not changed from the previous estimate of 2.4 mSv provided in the UNSCEAR 1988 Report [1], although there have been minor adjustments in the various components. One third of the total is due to external exposure to cosmic rays and terrestrial radionuclides and two thirds to internal exposure. The largest component of exposure, half of the total, is from radon and its decay products.

Additional data have been compiled from national surveys of external exposure rates and of radon concentrations. The population-weighted average dose rate in air in areas of normal background is 57 nGy h^{-1} outdoors and 80 nGy h^{-1} indoors. The indoor-to-outdoor ratio is thus 1.5, but the ratio can vary from less than 1 for lightweight houses to around 2 when the construction materials make substantial contributions to exposures. The concentration of radon is typically 10 Bq m^{-3} outdoors and 40 Bq m^{-3} indoors. In tropical areas with houses of lightweight construction and high ventilation, there should be little difference in indoor and outdoor levels. There are, however, too many factors that determine the concentrations, and measurements are necessary in all areas.

The dosimetry of radon is presently under review. UNSCEAR has retained the assumptions used previously, namely equilibrium factors of 0.4 indoors and 0.8 outdoors and a dose conversion factor of 9 nSv h^{-1} per Bq m^{-3} of equilibrium equivalent concentration (EEC) of radon. With these parameters the average annual effective dose from radon and its decay products is estimated to be 0.13 mSv for exposures outdoors and 1.0 mSv for exposures indoors. The average annual effective dose from inhalation of thoron (^{220}Rn) and its decay products is 0.07 mSv.

UNSCEAR has assessed the natural radiation exposures resulting from energy production using coal, oil, peat, natural gas and geothermal energy and the use of phosphate rock in fertilizers and building materials and of mineral sands. The highest exposures result from the use of phosphate by-products in buildings, the domestic use of coal for cooking and heating, and the use of phosphate fertilizers. The overall annual effective dose from all such sources averaged over the world's population is 0.02 mSv.

EXPOSURES FROM MAN-MADE SOURCES OF RADIATION

The assessment of the radiation exposures caused by releases of radionuclides to the environment from man-made practices or events has been updated in the UNSCEAR 1993 Report. Atmospheric testing of nuclear weapons resulted in the largest releases of radionuclides into the environment from man-made sources. Most of the testing occurred in 1952-1958 and 1961-1962. The last atmospheric test was conducted in 1980. From the many measurements that have been made throughout the years, UNSCEAR has evaluated transfer coefficients relating the input of radionuclides into the atmosphere to the resulting dose to humans. The collective effective dose to the world's population from atmospheric nuclear testing is estimated to be 30 million man Sv. Of this total, 86% is due to long-term, low-level exposure from ^{14}C. The contributions to dose in decreasing order of importance are ^{14}C, ^{137}Cs, ^{95}Zr, ^{90}Sr, ^{106}Ru, ^{144}Ce and ^{3}H. Only residual irradiation from ^{14}C, ^{137}Cs, ^{90}Sr and ^{3}H remains to be received by the present and future world population. The collective dose from this practice is equivalent the 2.4 years of exposure of the present world population to natural radiation sources.

There has been an increasing trend in electrical energy generation by nuclear reactors since the practice began in 1956. At present, about 20% of the world's electrical energy is generated by nuclear means. At the end of 1994 there were 432 reactors operating in 29 countries. During routine operation of installations associated with the nuclear fuel cycle (uranium mining and milling, fuel fabrication, reactor operation, reprocessing and waste disposal) radionuclides are released to the environment. The data on radionuclides released are quite extensive and complete, especially for reactor operations.

There have been generally decreasing trends in normalized releases of radionuclides from nuclear fuel cycle installations as operating practices have improved. This has meant that the trend in collective dose to the world population has been increasing somewhat less than the trend in electrical energy generated. The estimate of collective dose from nuclear power production was 43,000 man Sv during 1990 and 400,000 man Sv for the entire period 1956-1990. Even when the collective dose caused by the Chernobyl accident (600,000 man Sv) is added, the total collective dose has been just 8% of that which the world's population receives in one year from natural radiation sources.

Other man-made sources of radiation exposures caused by releases of radionuclides to the environment that have been assessed in the UNSCEAR 1993 Report include exposures to local populations near the Semipalatinsk, Nevada, Australian and Pacific nuclear test sites, exposures from underground nuclear testing, exposures from nuclear weapons fabrication, exposures from radioisotope production and use and exposures from accidents at the Three Mile Island and Chernobyl reactors, Kyshtym and Windscale plutonium production plants, crashes of airplanes carrying nuclear weapons, satellite re-entries and lost or mishandled radiation sources, as at Goiânia.

MEDICAL RADIATION EXPOSURES

The use of x rays and radiopharmaceuticals for diagnostic examinations and therapeutic treatments is quite common throughout the world. Most of the equipment and the procedures

performed are in industrialized countries, in which 25% of the world's population is located. UNSCEAR has assessed the exposures from medical radiation exposures from information obtained in questionnaires distributed to all countries. Four regions of health care have been designated, based on availability of facilities, and the data have been extrapolated to the world's population. The variations in medical radiation exposures among individuals are great, ranging from no dose to those not examined or treated to high doses to those receiving therapeutic treatments. The largest portion of the total dose from medical radiation sources arises from diagnostic examinations due to their relatively high frequency. At the highest level of health care the annual effective dose averaged over the population from all diagnostic examinations is 1.1 mSv. The comparable value is 0.05 mSv at the lowest level of health care. The population-weighted world average is 0.3 mSv, and the annual collective effective dose is 1.8 million man Sv. The collective effective dose from medical radiation usage has been evaluated to allow comparisons among countries and the evaluation of trends. Much, and optimally most, of the collective dose from medical uses of radiation is offset by direct benefits to the examined or treated patients.

OCCUPATIONAL RADIATION EXPOSURES

Occupational radiation exposures have been assessed from data submitted to UNSCEAR by national authorities in response to questionnaires. The data summarized in the UNSCEAR 1993 Report are quite extensive. Five-year average data for various occupations are reported for the period 1975-1989. The exposures from man-made sources are given the most attention; these data are usually required in countries for regulatory purposes.

The collective effective dose depends on the average individual doses and the number of exposed workers. The highest component of collective dose from man-made sources is from the nuclear fuel cycle (2,500 man Sv). There are 880,000 workers in this industry worldwide. For the 2.2 million medical radiation workers the annual collective dose is 1,000 man Sv. Fewer workers and lower collective doses arise in industrial uses of radiation (510 man Sv) and in defence activities (250 man Sv).

The collective effective dose to workers exposed to natural radiation sources is estimated to be 8,600 man Sv, which is two times higher than that from man-made sources. Some 5.2 million workers have been considered. The individual doses are more uncertain than from man-made sources. The largest component of occupational exposures from natural sources is from underground mining of coal and other minerals. Aircrew and some other occupations form secondary components.

MECHANISMS OF RADIATION ONCOGENESIS

Recent advances in molecular biology have been considered in one of several annexes in the UNSCEAR 1993 Report that deal with biological topics. The principal theories of oncogenesis and the results of experimental cellular and molecular studies are reviewed. The basic processes of induction, promotion and progression are recognized in oncogenesis, but it is not always possible to clearly differentiate these stages. Point mutations, chromosomal translocations and deletions, some of which are common and others specific to different neoplasms, may play roles in initiation and progression. Loss of function of tumour suppressor genes is considered a major

initiating factor in oncogenesis. Evidence of these genes being targets of radiation action comes from studies of germ-line mutations that predispose to cancer. Some of these genes appear to play a central role in control of the cell cycle.

The action of specific chemicals, hormones and growth factors on cell surface receptors alter proliferative responses of cells and lead to neoplastic progression. In some cases, the enzyme protein kinase C is thought to mediate promotional processes. Mutagenesis and repair of DNA damage are being studied in *in vivo* and *in vitro* systems. Although the complexities are great, the application of modern methods of cell and molecular biology in studies of radiation oncogenesis offer promising prospects of better understanding. Some aspects of research needs and future perspectives are briefly considered in this annex.

INFLUENCE OF DOSE AND DOSE RATE ON STOCHASTIC EFFECTS OF RADIATION

It is generally recognized that the effectiveness of radiation exposures becomes more than proportionally less at low doses and low dose rates. This is reflected in a quadratic term usually needed in describing the radiation response relationship. The factor of reduction may vary with the specific neoplasm and the physical and biological conditions of the exposure. In this annex of the UNSCEAR 1993 Report, the biological models of dose response in cells and organisms are reviewed and the experimental data available from studies of animals and cells in culture are analysed to derive the range of dose and dose-rate effectiveness factors.

The human epidemiological data on dose response are limited. Data from the survivors of the atomic bombings in Japan indicate that a reduction factor of about 2 would be appropriate for leukaemia but not much in excess of 1 for solid tumours. The results of studies of radiation workers are consistent with low values of the reduction factor. Information on thyroid cancer induction by acute external irradiation compared with low dose-rate exposure from intakes of ^{131}I are consistent with a reduction factor of about 3, although there are questions about the heterogeneity of the dose and uncertainties in the dose estimates and the effect that age makes to the overall reduction in risk. For female breast cancer, the information is conflicting, and a range of reduction factors from 1 to 3 can be derived.

For application of reduction factors the Committee considered that dose rates less than 0.1 mGy min^{-1} (averaged over about an hour) or acute doses less than 200 mGy may be regarded as low. The Committee concluded that reduction factors for low-LET exposures should be considered to be similar to those derived from the atomic bomb survivor data. Insufficient data are available to make recommendations for specific tissues. For high-LET radiation, there is little evidence of a consistent dose-rate or dose fractionation effect at low to intermediate doses. For hereditary disease, the adoption of a reduction factor of 3 is supported by experimental data in male mice; one study indicated the factor may be higher for female mice.

HEREDITARY EFFECTS OF RADIATION

It has not been possible to directly confirm radiation-induced mutations in human populations, so genetic risk estimates have had to rely on general knowledge of human genetics and extrapolation of results from animal experiments.

The understanding of human genetics at the molecular level is increasing rapidly. More precise analysis of the type of genetic damage caused by various agents, including radiation, is possible with new laboratory techniques. More recognition is also being obtained of non-traditional types of inheritance, such as mosaicism, genomic imprinting, uniparental disomy, gene amplification and cytoplasmic inheritance. The complexities involved may seem to make genetic risk estimation even more difficult and uncertain.

The specification of the genetic component of diseases and especially of the many so-called multifactorial diseases, which may occur throughout life and with varying severity, is a difficult problem. If some of the non-traditional mechanisms are involved, there could be trans-generational effects with manifestation of effects only after the F_1 or F_2 generations. There are few data to quantify these risks.

The Committee has concluded that there is no basis to alter present genetic risk estimates. Both the direct and indirect (doubling dose) methods of analysing animal data should be used, with due recognition of limitations of both methods. Radiation effects on multifactorial disease, gene regulation and non-traditional forms of inheritance are not well understood and may require different methods of estimation. The study of children of the atomic bomb survivors may be useful in setting outside limits on genetic risk estimates. These data indicate that hereditary effects from moderate acute exposure of a large human population are minimal. Further results are needed of both human and animal data analysed at the molecular level.

RADIATION EFFECTS ON THE DEVELOPING HUMAN BRAIN

The developing human brain is especially sensitive to ionizing radiation. This sensitivity reflects the structural complexity of the brain, its long developmental (and hence sensitive) period, the vulnerability of the undifferentiated neural cells, the need for cell migration to functional positions and the inability of the brain to replace most lost neurons.

The main effects of radiation and the sensitive periods have been derived from survivors of the atomic bombings in Japan exposed *in utero*. Thirty cases of severe mental retardation have been observed in 1,541 survivors. Most cases occurred in those exposed during the period 8-15 weeks following conception. A secondary period of reduced sensitivity occurred 16-25 weeks following conception.

The results indicate that damage caused by exposure to 1 Gy within the most vulnerable period (8-15 weeks following conception) increases the frequency of mental retardation to about 40% (background frequency: 0.8%) and lowers IQ by 25-30 points. The latter result is consistent with the observed increase in mental retardation. Exposure in the critical period also causes a decrement in average school performance and increases the risk of unprovoked seizures. There are no clear indications of thresholds for effects in those exposed in the most critical period. For the period 16-25 weeks, no cases of severe mental retardation were observed at exposure of less than 0.5 Gy. It is reasonable to assume the risks would be smaller for chronic exposures, but the data are too limited to provide quantitative estimates.

LATE DETERMINISTIC EFFECTS IN CHILDREN

Deterministic effects of ionizing radiation are the result of exposures that cause sufficient cell damage or killing to impair function in the irradiated tissue or organ. All tissues and organs may be affected, but tissues vary in their sensitivity to radiation. The ovary, testis, bone marrow, lymphatic tissue and lens of the eye belong to the most radiosensitive tissues.

Deterministic effects in children, with tissues actively growing, are often more severe than in adults. Examples of deterministic damage following radiation exposure in childhood include effects on growth and development, hormonal deficiencies, organ dysfunctions and effects on intellectual and cognitive functions. The review of deterministic effects of radiation in children stems mainly from the study of late clinical effects in children given radiotherapy treatments. As survival rates increase, some of the effects are becoming more apparent. The study groups are small, however, and the follow-up times are limited. The treatment modalities have usually included surgery and chemotherapy in addition to radiotherapy; it is thus not always possible to single out the effects of radiation alone.

The effects in tissues reviewed in this annex include those in the brain, endocrine system, gonads, skeleton, eye, cardiovascular system, lung, breast, liver and gastrointestinal tract, kidney and bone marrow. One objective was to determine the critical dose levels for the appearance of clinical deterministic effects. In general, younger children are more sensitive than older children. Owing to the paucity of data, however, it is not possible to quantify the effects by age in most situations.

EPIDEMIOLOGICAL STUDIES

The Committee previously reviewed the results of epidemiological studies in the UNSCEAR 1988 Report. The main basis for risk estimates in that report was the results of the Life Span Study of survivors of the atomic bombings of Hiroshima and Nagasaki. In the first of two annexes in the UNSCEAR 1994 Report [3], "Epidemiological studies of radiation carcinogenesis", the Committee presents a review of the wide range of epidemiological studies and provides comparative listings of risk estimates that can be derived from these results. The Committee feels that such broader analyses are necessary to establish more reliably the risk estimates. Results from a single study, although ostensibly providing statistically significant results for a particular site or type of cancer, may not represent the general case for one reason or another.

The epidemiological studies considered in the annex, in addition to the Life Span Study of survivors of the atomic bombings, include medically irradiated patients, occupationally exposed workers, individuals exposed to radionuclides released to the environment on various occasions and those exposed to elevated levels of natural background radiation. The characteristics of these studies, along with their strengths and weaknesses, are tabulated in the report.

The Life Span Study of survivors of the atomic bombings continues to be a primary source of epidemiological data on radiation effects. The large cohort includes individuals of both sexes and all ages with good dosimetric data covering a wide range of doses. Cancer incidence data for 1958-1987 are available for the first time, and the cancer mortality data have been extended for

another two years, available now for the period 1950-1987, for use in analysis of risk estimates in this annex.

The Life Span Study incidence and mortality data are broadly similar, with both sets of data demonstrating statistically significant effects for all solid tumours as a group, as well as for cancers of the stomach, colon, liver, lung, breast, ovary and bladder. The incidence data also provide evidence of excess radiation risks for thyroid cancer and non-melanoma skin cancers. Statistically significant risks were not seen in either the incidence or the mortality data for cancers of the rectum, gall-bladder, pancreas, larynx, uterine cervix, uterine corpus, prostate, and kidney or renal pelvis. An association with radiation exposure is noted for several types of leukaemia but not for lymphoma or multiple myeloma. Earlier data had indicated a possible association between radiation and multiple myeloma, but the new, extended analyses no longer indicate a statistically significant relationship.

Of the some 86,300 individuals in the Life Span Study cohort there were 6,900 deaths from solid tumours during 1950-1987. From comparisons with the control group, approximately 300 of these cancer deaths can be attributed to radiation exposure. The data for leukaemia incidence in this same period indicate that 75 cases can be attributed to radiation exposure.

The numbers of solid tumours associated with radiation exposure are not sufficient to permit detailed analysis of the dose response for specific sites or types of cancer. For all solid tumours together the slope of the dose-response curve is linear up to about 2 Sv. The dose-response curve for leukaemia is non-linear and is best described by a linear-quadratic function. Statistically significant risks for solid tumours in the Life Span Study are presently seen only above 0.2 Sv. The relative risks in the lower dose categories (0.01-0.05, 0.06-0.09, 0.10-0.19 Sv) all have positive nominal risk estimates, but they are not statistically different from unity. The slope of the dose response for doses lower than 0.5 Sv, while lower than the slope for all doses up to 4 Sv, does not differ significantly from it. An inherent limitation of epidemiological studies is to quantify results at doses less than 0.2 Sv because of the low statistical power of the available results.

Because of concerns about the role of cell-killing and the impact of errors in individual dose estimates on the slope of the dose-response curves at high doses, and because the Life Span Study risk estimates are primarily used to evaluate the effects at low doses, the analysis by UNSCEAR in this annex of lifetime risks have been limited to use of the data on individuals with shielded kerma of less than 4 Gy. The models for lifetime risk estimation allowed for differences in age at exposure and sex of the exposed individuals. Alternative assumptions were used for projections of risk beyond the present observational period. The relative risk was either assumed to remain constant to the end of life or to decrease to lower values at times greater than 40 years after exposure, as has been indicated to be the case in some epidemiological studies. The relative risks of leukaemia or lung cancer, for example, seem to decline after more than 20 years after exposure, although cancers of the GI tract can continue to occur for years longer.

The estimates of lifetime risk following exposure of 1 Sv, computed using sex- and age-at-exposure-specific relative risks estimated from the Life Span Study mortality data for 1950-1987 and using the demographic structure for Japan and the Japanese background cancer mortality rates for 1985, are 1.1% for leukaemia and 10.9% for solid tumours. These results may be compared with comparable values derived in the UNSCEAR 1988 Report of 1.0% for leukaemia and 9.7%

for solid tumours. The estimates for the alternative assumptions of risk beyond 40 years after exposure are 20%-30% less than the total risk estimate quoted above of 12% for an exposure of 1 Sv.

The estimates of risk are presented without adjustment for decreased effectiveness of radiation at low doses and low dose rates. The application of a small (<3) dose and dose-rate effectiveness factor was recommended in the UNSCEAR 1993 Report. If a factor of 2 is applied (as was used by ICRP in their 1990 recommendations), the risk estimate derived from the UNSCEAR 1988 Report would be 5% per Sv and from the 1994 Report 6% per Sv for a constant relative risk projection. The alternative projection methods would yield values from 4%-6% in the Japanese population (the applicability to other populations involves some additional uncertainty). Consequently, the use of a nominal value of 5% per Sv for mortality due to leukaemia and solid tumours from irradiation at low doses for a population of all ages (4% per Sv for an adult working population), as recommended by ICRP, still seems valid to the Committee based on these latest analyses.

Studies of other radiation-exposed populations, such as cervical cancer patients, ankylosing spondylitics and children treated for tinea capitis, provide risk estimates that generally support those derived from the data of the survivors of the atomic bombings. The other epidemiological studies provide additional information on issues that cannot be addressed by the atomic bomb survivor data, such as the effects of low chronic doses, highly fractionated exposures and variability among populations. For some sites of cancer, including leukaemia, breast and thyroid, and for bone and liver cancer from exposure to radium and thorium radionuclides, there are a number of very useful results from studies other than the Life Span Study. Large studies of occupationally exposed persons are also contributing tentative risk estimates. In general, there are no great discrepancies in risk estimates between the Life Span Study and the other studies.

Two studies provide some indications of risk at doses less than 0.2 Sv in sensitive subgroups of the population. These studies include cancers among those prenatally exposed to x rays with doses of about 0.01 Gy and thyroid cancer in children with doses of roughly 0.1 Gy. The risk estimates are in both cases, however, not yet well established.

UNSCEAR has further considered the reported incidences of leukaemia near nuclear installations in the United Kingdom, now thought unlikely to be related to environmental radiation or paternal exposure, and the evidence of cancer occurrence among participants of atmospheric nuclear tests. All such issues and results of epidemiological studies are reviewed in some detail in this annex.

ADAPTIVE RESPONSES

In a second annex in the UNSCEAR 1994 Report, "Adaptive responses to radiation in cells and organisms", the Committee considers the more recent research findings that are being reported on this interesting aspect of radiation response. The main impetus for the studies of adaptive responses has been the observation that human lymphocytes exposed to a conditioning dose of some 10 mGy while maintained with growth stimulants in culture media suffer about 50% fewer chromosome aberrations when subsequently irradiated with x rays to a dose of 1.5 Gy than when

exposed without the conditioning dose. It seems likely that the effect is linked to an increased capacity for DNA repair.

The adaptive response has been observed, as well, with proliferating bone marrow cells, spermatocytes and fibroblasts, but not with pre-implantation embryo cells. In some cases, *in vivo* exposure of an experimental animal (mouse or rabbit) has been able to provide the conditioning features which result in an adaptive response during subsequent *in vitro* exposure of lymphocytes.

Investigators have determined that the adaptive response for lymphocytes *in vitro* requires a conditioning dose of at least 5 mGy, delivered at a dose rate greater than 200 mGy min^{-1}, and no more than 200 mGy. The adaptive response occurs between 4 and 6 hours after conditioning and lasts for 3 cell cycles. The composition of the culture medium is quite important; it requires a narrow range of pH to be maintained and growth stimulating factors to be present. The degree of response depends on the stage of the cell cycle. Cells in active phases just prior to cell division are more likely to show the response than cells in other stages. Lymphocytes from different donors show variable response. Thus, the adaptive response is by no means a generally occurring phenomena; it requires rather carefully controlled experimental conditions to elicit the behaviour.

The possible mechanisms of adaptive responses are being investigated in current research. Cell cycle delay has been noted at relatively higher conditioning doses (>200 mGy). The delay allows cell damage to be repaired before the cell proceeds through the cell cycle. Since the adaptive response occurs at lower doses when this type of cell cycle delay is not apparent, this cannot be a central mechanism in the response at lower doses.

The evidence becoming available indicates that following radiation damage to cells a number of changes occur. Among these changes are the activation of genes that code for the synthesis of enzymes involved in the control of the cell cycle, proliferation of cells and repair of damage. Some of the enzymes seem to be similar to those induced by damage caused by other toxic agents. The adaptive response may therefore be part of a common mechanism involving cellular response to damage.

In addition to gene activation, enzyme production and at higher doses cell cycle delay, other cellular mechanisms involved may be detoxification of reactive radicals and activation of membrane-bound receptors stimulating cell proliferation. The immune response of the organism may operate at least transiently following radiation exposures in accelerating programmed cell death of damaged cells. All of these process are subjects of continuing investigation.

It is generally more difficult to demonstrate adaptive responses in the whole organism. Some earlier experiments reported seemingly stimulatory effects following low-level exposures. More recent experiments with rodents and beagle dogs exposed at various ages to low dose rates of low-LET radiation have been unable to demonstrate statistically significant differences in life-span of irradiated and control groups after accumulated doses of up to about one gray nor has reduced tumour induction been an outcome of low-dose exposures in these experiments. An important point to note is that along with adaptive response in cells, selective damage leading to malignant cellular transformations may also occur in parallel. This may explain the observed responses in these animal experiments.

The Committee thus judges that adaptive responses manifested by improved repair of cellular damage that take place under specific conditions in experimental cellular studies probably do not completely eliminate residual damage in cells that may ultimately result in malignant transformation. It will be important to continue these studies in order to understand more fully the molecular processes that occur following radiation exposure of cells and how these changes might be manifested in overall response of the organism. The Committee states that "at this stage it would be premature to draw conclusions for radiological protection purposes".

UNSCEAR has now begun a new programme of review of the sources and effects of ionizing radiation. The next scientific report of the Committee will be published probably in 1998.

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IRPA9
April 14 - 19, 1996 - Vienna

Developments in the Work of the International Commission on Radiological Protection of Importance for Radiation Protection

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ABSTRACT

The membership of the Main Commission and its four Committees for the period 1993-1997 was established in July 1993. The programmes of work are now well established and the Committees have met at least three times. Their progress is summarised in this paper.

INTRODUCTION

The Commission has four standing Committees on, Biological Effects, Secondary Standards, Protection in Medicine, and Application of the Commission's Recommendations. In July 1993 the four committees were reconstituted and by now have met at least three times to progress their programmes of work. This paper presents an overview of that work.

The method of working within ICRP and its Committees is also perhaps worth noting. When a Committee decides that it would wish to develop guidance on a particular issue, it proposes a Task Group on the subject to the Main Commission who approve its terms of reference and membership including the chairman, who will be a member of the sponsoring Committee. The other Task Group members will probably not be members of any Committee of ICRP. This procedure is necessary because the Commission supports the travel and subsistence costs of the Task Group meetings. In addition, the chairman of any Committee may appoint a Working Party within the Committee which is composed of one or more members of the Committee. The Commission takes no financial responsibility for Working Parties and they are usually formed to prepare papers for the Committee and their work often leads to the formation of a Task Group.

The distribution by country of people in ICRP is of interest. The largest contingent of 18 comes from the USA, followed by the UK with 11, France and Germany having 9 each, Japan and Sweden 5 each, China and Russia have 4, and 2 from each of the Netherlands, Argentina and India, and single membership from 9 other countries.

MAIN COMMISSION

The International Commission on Radiological Protection and its Committees met at Würzburg in September of the centennial year of Röntgen's discovery of x rays.

The Commission adopted three reports for publication in the *Annals of the ICRP*. The first was the report from a Task Group of Committee 2 and entitled 'Age Dependent Doses to Members of the Public from Intakes of Radionuclides: Part 4, Inhalation Dose Coefficients'. This report complements Publications 56, 67, and 69, which principally give ingestion doses for different age groups in the population, by presenting results using the new model of the respiratory tract. Only Publication 56 gave inhalation doses, using the old lung model, and these are now replaced.

The data presented in the age-dependent dose reports are restricted to radionuclides from the range of elements that are most likely to be of concern in environmental materials. For all of these elements, a detailed review of biokinetic behaviour has been undertaken.

In the second report adopted by the Commission, all data on dose coefficients from the four separate parts dealing with the public are brought together in a single issue for easy reference. As an annex, dose coefficients for the public are presented covering all the additional elements required for the IAEA Basic Safety Standards but for which biokinetic data had not been specifically reviewed in this case, the calculations are based on biokinetic data from Publication 30 and must be used with caution since they were derived for worker rather than public exposures. From now on, the work on internal dosimetry will mainly be directed at a comprehensive review of models and data applicable to workers and the public, the outcome of which will form the basis of calculated dose coefficients five or six years from now.

The third report approved was from the joint Task Group with the International Commission on Radiation Units and Measurements entitled 'Conversion Coefficients for use in Radiological Protection against External Radiation'. This has now also been approved by ICRU; it was agreed that it would be published by both Commissions in the *Annals of the ICRP*. A remarkable amount of work has been undertaken to obtain a large number of calculations for electron, neutron and photon exposures. The major concern for both Commissions was to ensure that the operational quantities recommended by ICRU achieved the objective of providing measurable quantities adequately representing the protection quantities of Publication 60. The analysis of the data in the report indicates that the operational quantities generally achieve this objective with only minor exceptions that are not important in practical circumstances; they are therefore a satisfactory basis for measurements in radiological protection against external radiations.

COMMITTEE 1

Committee 1 of the International Commission on Radiological Protection has the responsibility for maintaining the biological effects of ionising radiation under review and developing documents that relate such effects to the needs of radiological protection. During the last three years, two new task groups have commenced work and working parties have been formed to review a broad range of biological, biophysical and epidemiological topics.

Committee 1 met in Würzburg, Germany, in September 1995 and the outcome of this meeting is selectively summarised here.

TASK GROUPS

Genetic susceptibility to cancer

The Task Group on Genetic Susceptibility to Cancer had been set up by the Commission in order to provide an overview of human genetic predisposition to cancer and to make interim judgements on the possible implications for the estimation of cancer risk after radiation.
The first draft of the main body of the review contains sections on: DNA damage and repair, mechanisms and genetics of solid tumours and lympho-haemopoietic neoplasia, tumorigenic radiosensitivity and heritable

predisposition to cancer, together with computational modelling of genetically determined cancer risk after radiation.

There is particular emphasis on the potential importance of the repair of DNA double strand lesions for carcinogenic response and the relative genetic contributions to solid and lympho-haemopoietic neoplasia. Attention is also given to uncertainties on the apparently high genetic contributions to spontaneously arising breast and colon cancer and the limited extent of the database on which to estimate the degree to which cancer-prone human genetic disorders may contribute to excess cancer in irradiated human populations. These issues are critical to the computational modelling that has been proposed and will be the subject of further review and deliberation by the Task Group. A second draft of the report will be considered by Committee 1 in 1996.

Risk estimation for multifactorial diseases

It is recognised that considerable uncertainty surrounds the estimation of risk of induction of multifactorial diseases which depend upon genetic-environmental interactions for their expression. The brief of the Task Group on Risk Estimation for Multifactorial Diseases from the Commission is to review the classification, prevalence and genetics of human multifactorial diseases and to propose computational modelling procedures that will allow for more confident estimation of the impact of new radiation-induced mutations on disease incidence in the population. Major issues are the relative roles and multiplicity of major and minor genes in different disorders and the inherent difficulties of modelling the genetic and environmental interactions that may be involved.

One particularly important aspect of the current work is clarification by the Task Group of the dynamic nature of the mutational component of multifactorial disease. This had not been fully appreciated during the estimation of genetic risk in Publication 60 but was not deemed to have introduced a major source of further uncertainty. The first draft of the report of the Task Group is expected in 1996.

WORKING PARTIES

Committee 1 received reports from fifteen working parties and some of the topics reviewed are

Uncertainties in cancer risk estimates

The main uncertainties under review are those associated with the transfer of risk between different populations, calculation of lifetime cancer risks for tumour types and A-bomb dosimetry. The draft of an NCRP commentary on all these issues was circulated for comment. The overall conclusion from this and from the working party was that uncertainty on risk was not likely to exceed the current risk of around 5% Sv^{-1} for a whole population by more than a factor of around two in either direction.

Populations for new epidemiological study

Of particular note in this working party discussion was the potential for dosimetric and epidemiological follow-up of the population that had been exposed in the Altai region of the former Soviet Union as a consequence of explosions in 1949 and 1962. While the extent and magnitude of these population exposures was judged to be sufficient for such studies, there are inherent difficulties associated with such retrospective analyses, particularly since the Altai region has experienced a considerable degree of post-exposure migration.

Review of epidemiological data

Each year this working party provides a broad review of published epidemiological studies. In 1994–95 the working party had considered over 100 publications covering population studies, RERF reports, post-Chernobyl studies and studies on occupational and therapeutic exposures. Of particular importance were RERF studies on the induction of benign tumours of the gastrointestinal tract and stomach, and studies on lung cancer incidence in fluoroscopy patients, risk factors for breast cancer, and radon studies in China and West Bohemia together with studies on radon exposed miners showing no excess of extrathoracic tumours. In addition, the working party stressed the importance for low dose risk estimates of recent combined analyses of cancer in occupationally exposed nuclear workers.

Relative biological effectiveness and radiation weighting factors

The Committee has been reviewing data relevant to judgements on RBE and w_R and discussed the uncertainties involved, particularly those for different neutron energies and for heavy ions with LET values greater than 100 keV μm^{-1}. In spite of some justifiable concern on w_R values for these radiations, the Commission's current system for obtaining equivalent dose was, because of its simplicity, thought to have considerable practical merit; any proposed changes to the current system would need careful evaluation.

Attention was also given to a recent suggestion that a quality factor of 10 for alpha particles, particularly in respect of lung carcinogenesis, was more appropriate than the value of 20 or 25 advocated by the Commission and the International Commission on Radiation Units and Measurements, respectively. Following discussion of the relevant biological data and critical appraisal of the biophysical approach that had been used to justify an alpha particle RBE of 10, it was concluded that at present there was no good reason to contemplate such a change.

Tissue weighting for bone

A joint working party between Committees 1 and 2 had been seeking to develop an appropriate value for w_T for bone that took full account of the location of target cells for bone cancer at endosteal bone surfaces. It was agreed by Committee 1 that the ^{224}Ra patients provided the best epidemiological source of risk estimation and that a w_R of 20 remained appropriate for alpha particles. With due attention both to the factor necessary to convert the risk based on average bone dose to that which applies to bone surface and to the risk of fatal bone cancer suggested by the A-bomb data, a rounded w_T for bone surface of 0.01 was judged by the working party to be appropriate. A written report on the issue is expected in 1996 and this will include comment upon the implications of the new biokinetic model for radium. It is expected that this report will help to resolve uncertainties on w_T for bone that have existed since Publication 26.

Other working parties discussed new data on genetic effects, deterministic effects, target cells for carcinogenesis, the possible use of animal data for risk estimation, *in utero* effects, comparative aspects of radiation and chemical carcinogenesis, and adaptive responses to radiation. Good progress was being made in relating new findings in these areas to future judgements in radiological protection. Of particular note were new animal studies that strongly challenge earlier data on radiation-induced transgenerational carcinogenesis and the absence, so far, of a significant risk of *in utero* induced cancer in the A-bomb survivors.

COMMITTEE 2

Committee 2 has the responsibility for establishing secondary standards based on the Commission's recommended dose limits. The Committee and its task groups have been developing a series of documents related to both external radiation and internally incorporated radionuclides. The programme of work of Committee 2 was reviewed at the September meeting and is summarised here.

AGE-DEPENDENT DOSIMETRY

A number of reports have been prepared by the Committee giving age-dependent biokinetic models and dose coefficients (doses per unit intake) for members of the public for ingestion of selected radionuclides of 29 elements (Publications 56, 67 and 69). The elements/radionuclides were selected as those most likely to be of concern for public exposure. Following the issue of the new respiratory tract model (Publication 66) a report has now been prepared by the Task Groups on Internal Dosimetry (INDOS) and Dose Calculations (TGDC) giving age-dependent inhalation dose coefficients for these elements. The report, now adopted by the Commission, is expected to be issued as Publication 71, contains a summary of the new respiratory tract model, particularly in relation to age-dependent aspects. It also includes the results of literature reviews on the lung clearance of various chemical forms of the different elements, with the emphasis on the behaviour of environmental forms. The new respiratory tract model gives default absorption parameters for three default Types (Fast, Moderate and Slow). For radionuclides present in the environment, it will frequently be difficult to identify the chemical form. Dose coefficients will therefore be given in the report for all three absorption Types, with a summary of what is known about the behaviour of specific materials and guidance on the default Type to use when no information

is available. The summaries of data on the lung absorption of different chemical forms can be used as a guide for selecting absorption Types when there is more material-specific information. In general, the default Type adopted is M, on the basis that the calculated dose coefficient is unlikely to be greatly in error. In addition to the elements covered in the previous publications, dose coefficients will also be included for isotopes of calcium and curium. Annexes will be included in the document giving the biokinetic models for these elements which will be based on the generic models for the alkaline earths and actinides, respectively, as described in Publication 67. The table summarises the information given in these publications.

The publication of this report will complete the programme of work on dose coefficients for infants, children and adult members of the public. A summary report will therefore be issued giving a compilation of dose coefficients for both ingestion and inhalation from the above publications. The report will give doses to individual tissues, as well as effective dose. The Commission has also agreed that age-dependent dose coefficients for the radionuclides of the elements in Publication 30 not covered in Publications 56, 67, 69 and 72 will be given in an Annexe. These additional values are required by both IAEA, for the Basic Safety Standards, and EC, for the new EURATOM Directive. The biokinetic models used for these dose coefficients will be those given in Publication 30 for adults and allowance will be made only for changes in body mass and for doses from material in the excretion pathways. They must therefore be used with care for infants and children.

EMBRYO AND FETUS

A draft document has been prepared by INDOS, giving dose coefficients for the embryo and fetus following acute and chronic intakes of radionuclides by the mother, either before or during pregnancy. The report is at an advanced stage and it is expected that it will be completed in time for the next Committee 2 meeting to be held in the autumn of 1996. Dose coefficients for intakes by inhalation and ingestion will be given for radionuclides of the 31 elements for which age-dependent dose coefficients have already been prepared for infants, children and adults. Wherever possible, element-specific dosimetric models will be used (eg tritium, carbon, iodine, caesium and strontium). Where this is not possible, a generic model will be applied based on relative concentrations of radionuclides in fetal and maternal tissues obtained from studies with experimental animals. The report will include a comprehensive chapter on the development of the embryo and fetus, as well as the results of literature reviews on data in man and animals on the transfer of radionuclides to the developing embryo and fetus following intakes by the mother. Inhalation dose coefficients will be calculated for intakes of both 1 μm AMAD and 5 μm AMAD aerosols by the mother (default particle sizes for the public and workers, respectively). Dose coefficients for ingestion of radionuclides by the mother will be based on gut transfer factors, appropriate for both workers and members of the public.

DOSE COEFFICIENTS FOR WORKERS

Publication 68, giving dose coefficients for the inhalation and ingestion of radionuclides by workers, has recently been issued as a replacement for Publication 61. The dose coefficients apply the most recent biokinetic models recommended by the Commission, including the new respiratory tract model described in Publication 66. A revision of Publication 54 (Individual Monitoring for Intakes of Radionuclides by Workers) is now to be prepared, on a timescale of about a year, to provide information for interpreting monitoring data consistent with these models. The Commission has set up a working party of Committees 2 and 4 to develop this report. The models used for the calculations will be those adopted for Publication 68, for which extensive quality assurance checks have been carried out by the members of TGDC at BfS in Germany, NRPB, and ORNL in North America.

With the revision of Publication 54, Committee 2 and the Commission consider that there should now be some stability in these dose coefficients for workers. There remains, however, a need to develop a full revision of Publication 30. With the move to physiologically-based biokinetic models, the aim is to develop models that are appropriate both for bioassay interpretation and for dosimetry calculation. Thus the full revision of Publication 30 will take into account the work done for the revision of Publication 54. As it will be necessary to review the biokinetic models for all the elements that have not been reviewed in Publications 56, 67, 69 and 72, it is considered unlikely that the work will be ready for publication before the year 2000.

A technical report is to be prepared by INDOS giving guidance on the application of the new respiratory tract model to situations in which material-specific information is available which enables more accurate dose assessments to be made than would the use of general default parameter values. It is intended that in the revision of Publication 30, dose coefficients will be given for specific chemical forms of various radionuclides, in addition to those based on the default absorption Types. No ALIs are to be given as the Commission now considers that these should be established by management from a knowledge of local conditions. This technical report will provide information on the likely approach to be adopted in applying material-specific data.

RELIABILITY OF DOSE COEFFICIENTS

INDOS is preparing a report on the reliability of dose coefficients. This will examine the quality of the data used in the development of dosimetric models taking into account the extent to which they are based on human and animal data. The report will cover intakes by inhalation and ingestion, as well as the biokinetic models which describe the behaviour of systemic activity. Although it is unlikely that the document will be completed until 1997, a structured approach has been adopted which will be used to illustrate how the reliability of models for different elements can be judged.

MODEL DEVELOPMENT

It had been agreed at a previous meeting of Committee 2 that it would no longer be practicable to develop a full revision of Reference Man (Publication 23, 1975). Instead, a series of reports is in preparation covering the main information necessary for model development and for dose calculations. The first report, on the skeleton, has been approved for publication by the Commission and should be issued later in 1995. The next report, on basic anatomical and physiological data, should be completed in 1996. A third report will cover the regions of the gastrointestinal tract, as well as the pancreas, gall bladder and liver.

Summary of reports on age-dependent dose coefficients for members of the public from intakes of radionuclides

	PART 1[a]	PART 2[b]	PART 3[c]	PART 4[d]	PART 5[e]
ICRP Publication	56	67	69	72	–
Ingestion dose coefficients	+	+	+	–	+
Gastrointestinal tract model	30	30	30	30	30
Inhalation dose coefficients	+	–	–	+	+
Respiratory tract model	30	–	–	66	66
Tissue weighting factors	30	60	60	60	60

Notes

(a) For radioisotopes of H, C, Sr, Zr, Nb, Ru, I, Cs, Ce, Pu, Am and Np.
(b) For radioisotopes of S, Co, Ni, Zn, Mo, Tc, Ag, Te, Ba, Pb, Po and Ra.
(c) For radioisotopes of Fe, Sb, Se, Th and U.
(d) Elements in Parts 1, 2 and 3, plus Ca and Cm.
(e) Compilation of dose coefficients in Parts 1–4.

\+ Dose coefficients given in report.

In the development of ingestion dose coefficients for the public, the adult model for the gastrointestinal tract described in Publication 30 was used, as no age-dependent model is available. The Committee now intends to develop a new model for the gastrointestinal tract which will take into account the most recent information on anatomy and physiology as well as age-dependent characteristics. Additional information on the radiation sensitivity of different regions of the gastrointestinal tract and on the location of stem cells will also be considered. A working party has been set up, with membership from Committees 2 and 1, to develop the terms of reference for a task group. The aim is that the new model should be complete in time for the full revision of Publication 30.

The need to develop agreed phantoms for use in both internal and external dosimetry was considered by the Committee at some length. The MIRD phantom presently used for dose calculations has severe limitations as it is based on fixed geometrical shapes. This makes it difficult to introduce additional organs and tissues when the need arises as a result of new information on the sensitivity of tissues to irradiation. Comprehensive data are now becoming available from medical imaging and a number of laboratories are working towards developing realistic phantoms based on such data. Although there are difficulties in using such phantoms for dosimetry purposes, it was felt that this will be necessary before the revision of Publication 30. A working party is to consider how the Committee should proceed to develop new phantoms for the Commission.

COMMITTEE 3

The major work under this heading is the Task Group report on radiological protection and safety in medicine. Committee 3 had carefully considered a draft report and presented a revised outline to the Commission for agreement on a proposed way forward. The Commission accepted the proposals and decided that it was appropriate to form a joint task group between itself and Committee 3 with the objective of producing a report for adoption early in 1996; it would be aimed at relevant professional bodies and practising radiologists. The report would also serve as a source for the development of specific practical advice for protection in three important areas: nuclear medicine; diagnostic, dental and interventional radiology; radiotherapy.

COMMITTEE 4

Committee 4 of the International Commission on Radiological Protection has the responsibility for considering the practical application of the Commission's recommendations. This section reviews the deliberations of Committee 4 during its meeting in Würzburg in September immediately prior to the meeting of the Commission.

TASK GROUP REPORTS

The Committee had three draft reports from task groups to consider, all the titles of which contained qualifiers of the word 'exposure': namely, occupational, potential and chronic. The report on occupational exposure was the furthest advanced, although substantial progress had also been made with the other two.

Occupational exposure

The principal objective of the Task Group on General Principles for the Radiation Protection of Workers was to prepare a report that would replace Publication 35, which deals with the monitoring of workers, the intention being to take account of the developments in Publication 60. An additional objective was to review Publication 28, which deals with the procedures for handling emergency and accidental exposures of workers. However, it became apparent to the Task Group that this latter document should be subject to a separate review since it is mainly concerned with medical aspects of overexposure.

The general reaction of the Commission was favourable, although it recognised that there were issues of a policy nature and these needed careful consideration. The principal issues were the use of the public dose limits in the context of occupational exposure in Publication 60, the meaning and use of dose constraints, protection of the fetus of female workers, management of overexposed workers, and the grouping of workers for monitoring and dose

recording purposes.

This Task Group now plans to produce the next draft for review by the Commission early in 1996 before being considered by the Committee in May. Approval by the Commission will be sought towards the end of 1996.

Potential exposure

The report of the Task Group on Protection from Potential Exposure dealt with non-nuclear radiation sources, essentially those that present a significant risk. Particular examples are industrial irradiators and particle accelerators. There was no precedent for work in this area and as such the Task Group had to start with a 'clean sheet of paper'. The work is of considerable importance because such sources have been the cause of death and severe injury. The focus of the report was on the means of assessing the risks associated with such sources, rather than operational considerations. Nevertheless, recognising the significant contribution that human factors have played in the various accidents that have occurred in the past, the Committee felt that there was a need to include some consideration of these matters.

The general reaction of the Commission was that the report represented a major step forward. However, the Task Group was asked to give further consideration to detriment (including endpoints other than fatality), risk constraints and the role of the principle of optimisation of protection.

The intention is for the Task Group to meet again and a revised draft to return to the Commission late in 1996, after a further review by the Committee at its next meeting.

Chronic exposure

The report of the Task Group on Principles for Protection of the Public against Chronic Exposure Situations dealt with topics such as the exposure of the public from land contaminated by past practices. There are particular difficulties here in neatly fitting each situation into the categories of practice and intervention and the Committee felt that this needed to be more clearly recognised in the document. It was agreed that high natural background should be mentioned but the topic excluded from detailed treatment. Otherwise, the general approach was endorsed.

The Task Group intends to meet again in 1996 and the Committee will discuss the topic at its next meeting. The target date for final approval remains 1996-97.

WORKING PARTIES

The Committee considered a report from its working party on Publication 46, which deals with solid waste disposal. While it was felt that much of this publication remained valid, it was recognised that the material was not 'user friendly'. There were also questions over the use of future collective doses from waste repositories. It was agreed that a short policy document on radiological protection matters aimed at decision makers and their advisors involved in radioactive waste management would be useful, but not as a replacement for Publication 46. To undertake this work, the Committee proposed a task group. The Commission did not immediately accept this and has asked for an elaboration of the specific issues that the document would address, including long timescales and their influence on risk criteria and the use of collective doses for probabilistic events.

FUTURE PROGRAMME

The Commission intends to meet twice in 1996. The first meeting in March is planned for discussion of several of the reports being prepared by Task Groups prior to the Committees finally reviewing them. They include those on genetic susceptibility, protection in medicine, and occupational exposure. The second meeting is planned for November when, in addition to adopting any reports from the four Committee meetings, the major item will be preparation for the new Commission and Committees for the period 1997 - 2001.

THE ICRU PROGRAMME IN RADIATION PROTECTION

PAST, PRESENT AND FUTURE

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It is the purpose of this paper to present a panorama of the work that the ICRU has devoted - often in cooperation with the ICRP - to radiation protection. More precise information on recent achievements will be given in two companion papers by R.H. Thomas and P. Jacob. The first deals with an ICRP-ICRU joint report of great importance, the second is characteristic of one of the trends concerning the treatment by the ICRU of radiation protection problems.

The ICRU was created in 1925 to study the problem of "dosage" in radiotherapy and more specifically the unit to be used. It is thus not surprising that for three decades the major preoccupation of the ICRU concerned radiation units for radiotherapy with x and gamma rays and guidance for clinical applications.

In the 50's the energy domain as well as the quality (LET) domain of newly available sources of ionizing radiations were greatly increased and this presented the radiation protection community with two questions. The first, concerning the unit röntgen which is restricted to x and gamma rays, was in the process of being resolved by the introduction of the absorbed dose. The second, which is still today not entirely resolved, concerns the weighting of the absorbed dose according to the different biological effectiveness of the various radiations. The usage of the concept of RBE developed in biological research and the corresponding quantity "RBE dose" was reluctantly adopted by the ICRU, as the RBE of any radiation depends on many factors such as the type and degree of biological damage, the absorbed dose and its rate, the fractionation, the oxygen tension, the pH and the temperature. In trying to find a solution, the ICRP and the ICRU appointed in 1960 a Joint Committee to study the concept and use of RBE in considerations of radiation protection (1). The conclusion of the report was essentially as follows. The decision by ICRU to replace "RBE" in protection work by "quality factor" (*QF*) is endorsed. It is very necessary that a clear distinction should be made between the term "RBE" used in experimental work and the term used in protection calculations. The replacement of the term "RBE dose" by "dose equivalent" (*DE*), as defined below, is also endorsed. *DE* is defined as the product of absorbed dose, *D*, quality factor, (*QF*), dose distribution factor, *DF*, and other necessary modifying factors. The unit of dose equivalent is the "rem". The report also proposed an LET-*QF* relationship which is kept under review by the ICRP even today.

The same quantity, dose equivalent, is defined with a less cumbersome notation (*H* replaces *DE* and *Q* replaces *QF*) and appears in ICRU Report 19 (2) and its supplement (3) together with two new considerations. The first states that although *Q* is dimensionless and therefore *H* and *D* can be expressed in the same unit, it is highly desirable, for safety reasons, to keep the name rem instead of rad for expressing a dose equivalent. The second is the first attempt made by the ICRU to define quantities which are easy to measure for the specification of ambient radiation levels. These quantities, the absorbed dose index and the dose equivalent index, are defined at a point as the maximum absorbed dose or dose equivalent, respectively, within a 30 cm diameter sphere centered at this point and consisting of specified material. These quantities were developed later (4, 5) but finally abandoned mainly because of their lack of additivity.

During the same period (1970) the ICRU issued a report on radiation protection instrumentation and its applications (6), where instrument characteristics are reviewed to facilitate their choice and use.

In 1980, the ICRP and the ICRU established a Joint Task Group on Radiation Protection Quantities. The task group carried out a broad inquiry into basic approaches to radiation protection, an examination of the quantities that are required in their formulation, and a review of pertinent biological data. The corresponding report (7) was primarily focussed on the problem of radiation quality and its quantitative treatment in

radiation protection. It was intended to be an input *to* the two Commissions (and in particular to the ICRP) for their consideration in formulating subsequent recommendations *from* the Commissions.

A major breakthrough in the field of practical measurements in radiation protection was achieved in 1985 by the publication of ICRU Report 39 (8). This report gives definitions of *operational quantities* to be employed in the monitoring of ionizing radiation that provide a basis for effective radiation protection. These quantities are intended to supplement, for practical measurements, the recommendations concerning the use of the absorbed dose index and the dose equivalent index. A second report (9) indicates how the dose equivalents received by exposed individuals may be determined when it is appropriate to utilize additional data on irradiation or different calibration conditions. A third report (10) contains guidance on the design, calibration and use of instruments required to implement the recommended system.

In 1991, the ICRP (11) introduced two new quantities, based on radiation weighting factors, replacing the organ dose equivalent and the effective dose equivalent. These quantities are termed equivalent dose and effective dose, respectively. As these quantities cannot be evaluated experimentally, the quantities defined in terms of Q, namely *ambient dose equivalent, directional dose equivalent* and *personal dose equivalent*, are to be used for measurement purposes (11, 12). The compatibility of the operational quantities with the new protection quantities expressing dose limits will be analysed by R.H. Thomas.

In the future, the centre of interest of the ICRU will continue to be on concepts and measurement techniques. In addition to a report, in press, on dosimetry of external beta radiation for radiological protection, two main directions are being pursued in the field of radiation protection.

The first direction encompasses *methods and measurement techniques related to the assessment of radionuclides*. P. Jacob will present a report recently issued on gamma-ray spectrometry in the environment (13).

All measurements made in the environment are based on the sampling method. In order to be able to draw inferences for scientific interpretation or monitoring, requirements for radiological sampling have to be stated and this is new work which is just starting.

Another report - which is nearing completion - is concerned with direct (*in vivo*) measurement of radionuclides in the whole body or in specified organs or regions of the body.

The second direction is concerned with *patient dosimetry in diagnostic radiology*. A report is being prepared on the determination of absorbed dose in a patient from x rays used in diagnostic and interventional radiology.

The ICRU has an extensive programme on the assessment of image quality in medical imaging. A report which provides the theoretical framework for image quality evaluation of all the conventional imaging systems is in print (14). It will be followed by specific reports on image quality in mammography, nuclear medicine and chest radiography. All these reports will also deal with the problem of dose reduction which cannot be separated from considerations on image quality.

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CONVERSION COEFFICIENTS FOR EXTERNAL RADIATIONS

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Early in September 1995, at its meeting in Würzburg, the International Commission on Radiological Protection (ICRP) accepted the report of a Joint ICRP/ICRU Task Group on External Radiation. A week later, some 230 km to the northwest in Lennep, the report was also endorsed by the International Commission on Radiation Units and Measurements (ICRU).

The report will be of general interest to dosimetrists and in particular to those interested in the estimation of the doses arising as a result of the irradiation of the human body by ionizing radiations from sources outside the body.

Two sets of quantities are of importance in radiological protection. Dose limits are expressed in terms of *protection quantities* and compliance with these limits can be demonstrated by a determination of the appropriate *operational quantity*.

In Publication 60 the ICRP made significant changes in its definitions of the protection quantities and recommended that the equivalent dose, D_T, and effective dose, E, be used in radiological protection (1). These protection quantities are not directly measurable. For exposure of humans by sources of radiation outside the body (external radiations), the convention has been adopted that operational quantities defined by the ICRU should be used for practical measurements. The two sets of quantities may be related to radiation field quantities such as particle fluence and in turn, by sets of conversion coefficients, to each other.

The operational quantities now in use, ambient, directional and personal dose equivalent, [$H^*(d)$, $H'(d)$ and $H_p(d)$], were originally defined to be compatible with protection quantities defined by ICRP in the seventies (2, 3, 4). Since that time ICRU has published clarifications to the definitions of the operational quantities; new physical data bases have become available and improvements have been made in the

methods of performing radiation transport calculations. These improvements in physical data and computational techniques would alone warrant a review of the conversion coefficient data provided in ICRP Publication 51 (5).

The significant changes made in the definitions of the protection quantities in ICRP Publication 60 and the modifications to the definitions of the operational quantities naturally raise questions as to whether the operational quantities, as currently defined, are still a good measure of the new protection quantities.

In 1991 the ICRP and ICRU set up a Joint Task Group that was requested to review published conversion coefficients and related data and to provide to an agreed set of reference conversion coefficients data for application to external dosimetry for radiological protection.[1]

The Joint Task Group was asked to:

- provide conversion coefficients for protection quantities,
- provide conversion coefficients for operational quantities,
- examine the relationship between protection and operational quantities.

The new report, which will be published next year, revises and replaces much of the data given in ICRP Publication 51 and in ICRU Report 43 (4).

The report analyses the consequences of the changes in the definitions of the protection quantities recommended in ICRP Publication 60 and, finally, demonstrates that the operational quantities still function as adequate predictors of the protection quantities, *i.e.*, that in most practical circumstances they neither under-estimate nor significantly over-estimate protection quantities.

Published data were reviewed for photons with energies up to 10 MeV, for neutrons with energies up to 180 MeV and for electrons with energies up to 45 MeV. The review was not extended to higher energies because data are not yet available for the protection quantities defined in ICRP Publication 60.

Reference conversion coefficients relating the protection and operational quantities to the appropriate physical quantities (*e.g.*, air kerma, absorbed dose or particle fluence) are in the report.

ICRP Publication 60 redefined the Protection Quantity recommended in ICRP Publication 26 (Effective Dose Equivalent) essentially by changing the organs and tissues to be considered, recommending new tissue weighting factors and finally by changing the methods by which absorbed doses be modified to account for radiation quality. The operational quantities have been changed both by the revision of the *Q-L* relationship in ICRP Publication 60 and new stopping power data reported in ICRU Report 49. The Joint Task Group Report examines the consequences of these changes in some detail and concludes that, in most practical circumstances, the operational quantities defined by ICRU in 1985 still provide a satisfactory basis for measurements for radiological protection against external radiations. Remarkably, this is so despite changes in the definitions of the protection quantities, changes in the specifications of the operational quantities, new physical data and improved methods of calculation.

For photons and electrons, measurement of the operational quantities for area monitoring will continue to provide a reasonable over-estimate (typically 20 % or more) of the protection quantities in all irradiation geometries. For neutrons the results are

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[1] Members of the Joint Task Group were:

Full Members	
L.W. Brackenbush	PNWL, USA
J.-L. Chartier	IPSN, France
M.J. Clark	NRPB, UK
G. Dietze	PTB, Germany
G. Drexler	GSF, Germany
H.G. Menzel	CEC, Brussels
Chair:	
R.H. Thomas	University of California, USA
Corresponding Members	
R. Griffith	IAEA, Vienna
B.L. Grosswendt	PTB, Germany
N. Petoussi	GSF, Germany
B.R.L. Siebert	PTB, Germany
M. Zankl	GSF, Germany

more complicated. In most, but not all, irradiation geometries the situation is similar to that for photons and electrons. There is a particularly important exception for the AP geometry at low neutron energies (about 1 eV) where underestimates of the protection quantity by about 25 % can occur. Particular caution is also required when dealing with neutrons (and other radiations) having energies above about 20 MeV, because the operational quantities defined at a depth of 10 mm in tissue may not provide sufficient over-estimation of protection quantities. In those cases where the operational quantities prove to be unsatisfactory the data provided in the report provide a basis for the design of adequately protective measurement programmes, the correct interpretation of measurements results and their relation to the protection quantities. It is most important to stress that, if reliable results are to be obtained, knowledge of the neutron spectrum in which measurements are to be made is always highly recommended.

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GAMMA-RAY SPECTROMETRY IN THE ENVIRONMENT

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INTRODUCTION

The term gamma-ray spectrometry in the environment comprises methods based on an *in-situ* measurement of the spectral distribution of the photon fluence rate (1). The first main chapter of ICRU Report 53 (2) summarises the basic principles of gamma-ray spectrometry. If the measurement devices are coupled to computing devices with appropriate evaluation programs, they constitute rapid methods for an *in-situ* determination of environmental radioactive contaminations being capable to give radionuclide specific results. The methods considered mainly in ICRU Report 53 refer to gamma-ray spectrometry at ground-level, to gamma-ray spectrometry operated from aircraft, and to measurements of spectral distributions of the photon fluence rate and of absorbed dose rates in air.

Historically, the term *in-situ gamma-ray spectrometry* has been adopted for measurements that are performed close to the source, especially for ground-level gamma-ray spectrometry, mostly performed at a height of 1 m above ground. It is

mostly performed with high purity n-type germanium detectors. These detectors are capable to measure the spectral photon fluence in the energy range of 10 keV to 10 MeV, they have a high energy resolution (in the energy range of 100 keV to 1 MeV it is in the order of 1 keV), they are robust and need to be cooled only during the measurements. In ICRU 53, *in-situ* gamma-ray spectrometry is reviewed and updated by recent results of photon transport calculations. The sensitivity and the uncertainties are discussed. Methods for an *in-situ* determination of the attenuation of the radiation due to the surface roughness of the ground and the depth distribution in the soil are described. Some advanced methods in structured terrain (*e.g.*, in urban environments), with shielded and with unshielded detectors, and the application of on-ground gamma-ray spectrometry to the determination of radionuclide concentrations in air are reviewed.

Airborne gamma-ray spectrometry, *i.e.*, when spectrometers are operated from aircraft, is a uniquely rapid method, capable and economical for total mapping of large areas and for the location of lost radioactive sources. Standard equipment and methodologies based on geophysical survey techniques with scintillation detectors had been reviewed by the International Atomic Energy Agency (3) and an IEC standard (4) defines additionally technical and safety requirements for the equipment. ICRU Report 53 describes the types of equipment currently being used or developed for airborne gamma-ray spectrometry. Essential features of mapping natural and anthropogenic radionuclides are summarised. While it is recognized that the many years of practical application of airborne gamma-ray spectrometry have led to effective empirical approaches, some progress towards absolute methods involving photon transport calculations and towards systems incorporating high resolution detectors elements are discussed.

The last main chapter comprises methods to derive radionuclide specific gamma dose rates in air from the measured spectra. In addition, methods to determine the full spectral photon fluence in air, *i.e.*, not only the peaks, are reviewed, expressing the large potential of photon transport calculations in this field.

GROUND-LEVEL GAMMA-RAY SPECTROMETRY

Radionuclide distributions in the ground are parameterized by the mass per unit area, z (g cm^{-2}), of soil above a given source element. Photons with the same energy as source photons are called primary photons. A mathematical annex comprises relations between fluences of primary photons in air and radionuclide concentration in the ground and in the air for various kinds of source distributions. An important parameter in these relations is the total mass attenuation coefficient excluding coherent scattering, μ'/r (cm^2 g^{-1}), and its dependence on soil type is discussed.

One criterium for characterising n-type detectors is the ratio L/D of the length to the diameter of the sensitive detector region. The angular response to various radionuclide distributions in the ground is shown for detectors with different L/D ratios.

For monoenergetic sources distributed exponentially in the ground, primary photon fluences in air are given. Extensive calculation results are presented for anthropogenic radionuclides. Observations on depth distributions in dependence on the time after the deposition of anthropogenic radionuclides are summarised. Recommendations are given, which depth distributions might be assumed for an evaluation of *in-situ* spectra, if no site-specific information is available. Minimum detectable activities are given for characteristic cases and main sources of uncertainties are discussed. Methods for an *in-situ* determination of the attenuation by the ground are reviewed. For the first years after a cesium deposition, a method based on the simultaneous measurements of primary fluences of x-ray and gamma lines is shown to be applicable.

In-situ gamma-ray spectrometry in urban environments is reviewed. Shielded detectors are favourable for determinations of activities per unit area of urban surfaces like streets, walls or roofs, unshielded detectors are favourable for dose determinations. Fluences of primary photons in air due to horizontally finite radionuclide distributions in the ground are discussed.

Primary photon fluences in air at a height of one meter above ground are tabulated for natural radionuclides with uniform distributions in the ground and for radionuclides with uniform distributions in air. Over- and underestimations of radionuclide activities per unit air volume at the site of measurement due to applying these factors to finite clouds are discussed. Minimal detectable activities per unit volume are given for some common radionuclides in air after atmospheric releases of nuclear-technological plants.

AIRBORNE GAMMA-RAY SPECTROMETRY

The detector types discussed comprise standard geological systems, semiconductor detectors and combined systems. The main principles of various calibration approaches are discussed. Currently, the approach mainly used is to calibrate the detector system in small heights (about 1 m) over finite calibration pads. This type of approach needs approximations to take properly into account the scattering of the photons in the air between the source and the aircraft. For standard geological systems with NaI-detectors, this involves the derivation of the height dependence of the stripping ratios between the various spectral windows. The corresponding task is much simpler for germanium detectors with a high resolution. To support further developments on calibrations comparable to the ground-level gamma-ray spectrometry, tables with primary photon fluences in three flight heights above ground are supplied for various exponential source distributions in the ground.

Airborne systems equipped with air samplers are common in determining radionuclides in air. Nevertheless, airborne *in-situ* gamma-ray spectrometry has the advantage of measuring average concentrations over relatively large volumes without being subjected to local fluctuations. Also, in the case of accidental releases with high air concentrations, results may be obtained by overflying the highly contaminated cloud without flying directly through it. An approach of deriving radionuclide releases from spectra obtained by overflying a radioactive cloud is reviewed.

DETERMINATION OF GAMMA-RAY DOSE QUANTITIES

Assuming the shape of depth distributions of radionuclides in the ground, radionuclide-specific gamma dose rates in air may be derived from *in-situ* gamma ray spectra. The results for the dose rates depend less on the assumption on the depth distribution than corresponding results for radionuclide activities per unit area. The attenuation of the photon fluence is relatively simple to determine by *in-situ* gamma-ray spectrometry and laboratory measurements of the radionuclide active per unit area in soil samples. It is shown, under which conditions this information is sufficient to determine the radionuclide specific absorbed dose in air with good accuracy.

Extensive tables are given for gamma dose rates absorbed in air due to exponential distributions of anthropogenic radionuclides in the soil. The tables base on recent photon transport calculations (5) and a recent compilation of nuclear decay data (6). Further tables give corresponding results for uniform distributions of natural radionuclides in the ground and for uniform distributions of anthropogenic radionuclides in the air.

Measurement procedures of spectral photon fluence and absorbed dose rate distributions in air are reviewed. Methods to establish detector response functions by exposing them to monoenergetic or bienergetic gamma sources and applying appropriate interpolation functions are discussed. The high accuracy being obtainable by photon transport calculations of the response function of high purity n-type germanium detectors is demonstrated for applications to spectral photon fluences close to a ^{152}Eu source, in air over a contaminated free field and inside a house that is exposed to a ^{60}Co source.

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ELECTROMAGNETIC FIELDS AND CANCER: HOW ICNIRP HAS DEALT WITH THE ISSUE

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INTRODUCTION

Whether exposure to electromagnetic fields (EMFs) cause cancer has been vigorously debated for many years and has been the most vexing issue with which ICNIRP has had to deal during its short existence. There have been three parts of the electromagnetic spectrum that the issue of cancer has raised: static (0 Hz) magnetic fields, extremely low frequency (ELF) fields (defined as >0 - 300 Hz, but concerns have been raised almost exclusively at the power frequencies of 50/60 Hz), and radiofrequency (RF) fields (300 Hz - 300 GHz). By far the major problems have arisen during the contruction of new high voltage transmission lines and mobile telephone systems. Actions by protest groups concerned with possible health effects, especially with cancer in children, has now reached such a scale that it is costing electrical utilities and communications companies billions of dollars annually world-wide. With such high stakes, ICNIRP has had to be extremely careful in its evaluation of the scientific literature, use valid and defensible methods of literature review, and be completely independent of any special interest groups.

ICNIRP has had to continuously monitor the scientific literature to determine if any studies provide evidence strong enough to warrent a re-evaluation of the guidelines on exposure limits. This has been an increasing problem, particularly in this age of rapid communications (eg INTERNET), where information on new studies is quickly disseminated, evaluated and distributed to various special interest groups around the world who want to know what these studies will mean with respect to standards.

ICNIRP is a collaborator in a the International EMF Project with WHO, the International Agency for Research on Cancer, and many other international and national organizations having responsibility for EMF protection. The objectives of the Project are as follows:

(i) Pool resources of international and national agencies and key scientific institutions in the environmental health domain working on the biological effects of electromagnetic fields.

(ii) Identify gaps in scientific knowledge, provide protocols for the conduct of this research and encourage research in those areas that will lead to better health risk assessments.

(iii) Provide authoritative, independent, scientific, peer-review of the scientific literature, with fully substantiated recommendations.

(iv) Publish an updated EHC monograph giving a health risk assessment using results obtained during the Project on health effects of exposure to static, ELF and RF fields. This monograph would clearly differentiate between well established effects and those requiring further research.

(v) Publish an EHC monograph on risk perception, risk communication and risk management,

and public and occupational health policy.

(vi) Publish reports on appropriate topics to assist and support national health programs.

(vii) Use modern efficient means of communicating essential research information and findings which develop during the project.

It should be clearly understood that ICNIRP is a full partner in this Project and that the conclusions on health risk will assist ICNIRP to draft its own guidelines on exposure limits. A more complete description of the Project is available from WHO.

This paper summarises what criteria ICNIRP uses to review the literature, its response to EMF exposure and cancer, and its current position on static, ELF and RF fields.

HOW ICNIRP ASSESSES THE SCIENTIFIC LITERATURE

Scientific studies are in three categories:

(i) ***In-vitro studies*** conducted on isolated components of biological systems such as solutions of molecules (eg DNA), cultures of cells, or pieces of tissue. These studies are important for determining possible mechanisms by which EMF fields interact with biological systems and for identifying appropriate end-points and exposure conditions to be tested in whole animals. Determining mechanisms of interaction is important to give an understanding of how EMF fields act at the molecular or cellular level, and thus allow an extrapolation to the whole animal level. Studying simple systems allows interactions to be detected that may be masked in the complexity of interactions that occur normally at the whole animal level. It is because of this that biological effects found to occur at the molecular or cellular level cannot be assumed to occur at the whole animal level. Thus biological end-points found in-vitro must still be tested *in-vivo.*

(ii) ***In-vivo studies*** are conducted on complete biological systems such as laboratory animals. The great advantage of these studies is that they are conducted under carefully controlled laboratory conditions where all environmental and exposure parameters are kept constant. The only difference between exposed and unexposed animals should be the actual exposure to EMF fields they receive. Since experiments cannot normally be conducted on humans, animal studies are very useful for making health risk assessments related to human exposure. However, when evaluating animal studies, it is important to remember that the results of these studies are only applicable to humans if the effects observed occur in a number of different animal models. This is necessary because one particular animal model may be extremely sensitive to a particular end-point and have characteristics that are not observed in humans.

(iii) ***Human studies*** can be conducted on volunteers in the laboratory or on different populations of people in the living and working environment. Laboratory studies are conducted with the approval of the volunteer, and have the advantage of allowing exposures under strictly controlled conditions (as with animal studies). However the end-points that can be studied are limited. End-points such as cancer and mortality obviously cannot be studied on laboratory volunteers.

Studies on populations are called ***epidemiological studies*** and have the advantage of being non-intrusive. They compare the differences in the incidence of or mortality from some predetermined disease or diseases in populations. Generally one population is exposed to EMF fields and is compared with a population not exposed or at least having a much lower exposure to EMF fields. The major difficulty is to obtain two identical groups in sufficient numbers where the only difference is their exposure. This can become a significant problem when studying rare diseases such as cancer. However, these studies can indicate differences in the incidence of disease; the difficulty then being to attribute this difference to the EMF exposure and not some other factor in the living or working environment that is not detected as a difference between the two groups (eg chemical exposure in the workplace).

Criteria for evaluating scientific studies

When reviewing the scientific literature, certain criteria must be met if claims of a positive or negative effect are to be accepted into the database for conducting a human health risk assessment (Repacholi & Stolwijk 1991):

1) Experimental techniques, methods and conclusions should be as completely objective as possible, using biological systems appropriate to the end-points studied. To safeguard against bias, researchers should use double-blind techniques. Appropriate controls must be used for valid comparison of results. The sensitivity of the experiment must be such that there is a reasonable probability that an effect could be detected if it exists.

2) All data analyses should be fully and completely objective, no relevant data should be deleted from consideration and uniform analytical methods used. Data from experiments within the same protocol should be internally consistent.

3) The published descriptions of the methods should be given in sufficient detail that a critical reader would be convinced that all reasonable precautions have been taken to meet requirements 1 and 2.

4) Results should demonstrate an effect of the relevant variable at a high level of statistical significance using appropriate tests. The effects of interest should ordinarily be shown by a majority of test organisms and the responses found should be consistent.

5) Results should be quantifiable and susceptible to confirmation by independent researchers. Preferably the experiments should be repeated and the results confirmed independently; or the claimed effects should be consistent with results of similar experiments, where the biological systems are comparable.

6) Results should be viewed with respect to previously accepted scientific principles before ascribing them new ones.

While it will not be possible for all the above criteria to be applied to all experiments, these criteria provide a guide when determining what effects are established and can be used in a health risk assessment, and those that merely raise a hypothesis that needs to be tested, or those results which should be considered as preliminary and needing confirmation. Information from the

various types of laboratory and human studies, including the limitations in the amount of information they can provide, is also taken into account when conducting health risk assessments.

Assessment of epidemiological studies

Epidemiological demonstration of an association between two variables need not imply causality - both may be due to a common factor. However, establishment of causality is enhanced if (Miller 1986, 1989):

a) the presumed causal event precedes the effect,

b) one rather than multiple cancers are caused by a given exposure; giving specificity of effect,

c) a dose-response relationship exists,

d) there is consistency with other observations on cause and effect,

e) there is the exclusion of concomitant variables, or no alternative explanations,

f) the effect disappears when the cause is removed, and

g) the results are consistent with those from animal experimentation and other human observations.

Not all of these factors can be evaluated or will be true for even the most fully studied effects of an environmental exposure. It is known for example, that ionizing radiation induces various but not all forms of cancer. However, laboratory evidence is normally necessary to support the human studies. In other words, if there is truly a link between exposure to EMF and cancer, one would expect that laboratory studies on animals should show that EMF can cause cancer. If the laboratory studies do not show this and the human studies suggest only a weak association, then most likely there are problems with the human studies and they are not reaching valid conclusions.

What are the factors that must be taken into account when reviewing the epidemiological studies? What problems could affect the validity of associations found in epidemiological studies? There are four factors that could result in false associations in epidemiological studies:

- **inadequate dose assessment:** It is necessary to determine with some reasonable accurately a person's exposure to EMF. If these fields are associated with cancer, what aspect of the field is involved? To date it has been very difficult to gain incites into what should be considered as our concept of "dose" for EMFs.

- **confounders:** Other cancer risk factors could be causing a false association between exposure to EMFs and risk of cancer. Associations between things are not always evidence for causality. Exposure may be associated with a cancer risk other than EMFs. If such an associated

cancer risk were identified it would be called a "confounder" of the epidemiological study. An essential part of epidemiological studies is to identify and eliminate possible confounders. Possible confounders may be exposure to herbicides and chemicals such as PCBs, traffic density, socio-economic class and population mobility.

- **inappropriate controls:** The control groups must be selected to match as closely as possible all aspects (age, sex, socio-economic status, occupation etc) of the cancer cases (in a case-control study, which most of the epidemiological study reports are). An inherent problem with many epidemiological studies is the selection of a "control" group that is identical to the "exposed" group for all characteristics related to the disease except the exposure. This is especially difficult for rare disease such as leukaemia and brain cancer where the risk factors are poorly known. An additional complication is that people must consent to be in a "control" in a study, and participation in such studies is known to depend on many factors such as socio-economic class, race and occupation which are linked to differences in cancer rates.

- **publication bias:** Studies reporting positive results are much more likely to be accepted for publication than those with negative results. This would have the effect of skewing any analyses of a number of studies and severely bias meta-analyses that attempt to combine all studies (positive and negative) to investigate trends from a group of studies. Such study groups may erroneously contain more positive studies if negative studies are not published. This can happen because scientific journals are more interested in these for sales and publicity purposes or authors may not feel inclined to publish. Publication bias can thus increase apparent risks. This can effect both epidemiological as well as laboratory studies. Several specific examples of publication bias are known in studies of electrical occupations and cancer (see NRPB 1992). In their review Coleman and Beral (1988) reported the results of a Canadian study that found a RR of 2.4 for leukaemia in electrical workers. However, NRPB (1992) found that further follow-up of the same Canadian workers showed a deficiency of leukaemia (a RR of 0.6), but that this has never been published.

In reviewing the scientific literature, these factors must be taken into account to reach valid, supportable conclusions.

STATIC MAGNETIC FIELDS

There has been increasing concern among workers and the general public that exposure to static magnetic fields may be detrimental to health. This concern has been heightened by the increasing number of sources of exposure to these fields, and the ongoing debate about the possibility that exposure to 50/60 Hz magnetic fields may increase the incidence of cancer, with the implication that static magnetic fields may also be carcinogenic. Concerns have also been expressed because the next generation of train will use static magnetic fields (the magnetic levitation trains). The trains would expose very large numbers of people to quite strong static magnetic fields. In the field of medical diagnosis, magnetic resonance imaging units now expose patients and operators to magnetic fields as strong as 2 T. In the future, magnetic resonance spectroscopy could expose patients to fields up to 10 T.

Static magnetic fields are produced either by permanent magnets or whenever a direct current (DC) flows in a conductor. Direct currents are used to plate metals onto electrodes and

so static magnetic fields are produced around the electrolysis tanks. Magnetic fields are not appreciably distorted or attenuated by the human body. However they induce electrical currents in the body wherever there are moving parts (eg heart pumping or the body moving through the magnetic field). Thus, while adverse effects (such as involuntary nerve and muscle stimulation) could occur as a result of these induced currents, the cancer issue still had to be addressed.

ICNIRP completed a thorough review of the available literature on static magnetic fields in conjunction with the World Health Organization (UNEP/WHO/IRPA, 1987) and published a statement on magnetic resonance imaging (INIRC/IRPA, 1991). ICNIRP guidelines on exposure limits to static magnetic fields were published in 1994 (ICNIRP, 1994). No substantiated evidence was found to suggest that exposure to these fields is in anyway carcinogenic.

ELF FIELDS

Many studies have suggested that children exposed to 50/60 Hz magnetic fields, determined by residence near high current electrical installations (high voltage transmission lines, high current distribution lines, pole transformers etc) were at an increased risk of leukaemia or brain tumours. Later studies, using direct measurements and calculations of historical magnetic fields, have also suggested an increased risk of childhood cancer, and an increased carcinogenic risk in workers whose occupations have a higher exposure to magnetic fields.

Childhood epidemiological studies, when considered together, suggest there may be an association between cancer and exposure to 50/60 Hz magnetic fields. However, the average odds ratio for these studies is only of the order of 2.0. These odds ratios are quite small when one compares the odds ratios for cancer induction reported for smoking and X-ray exposure. When the odds ratios are so low, it would be more convincing that a true association exists if there were well conducted and independently confirmed laboratory studies indicating that 50/60 Hz field-exposure of animals or some recognised biological systems would either cause cancer directly or provide reasonable evidence that these fields influence the process of carcinogenesis. It is not necessary that a mechanism be found which would explain how such exposure had a carcinogenic effect, but merely that when the effect was found, it was statistically significant and independently replicated, and that a clear link with human cancer was established. Evidence from the animal studies to support the epidemiological evidence is lacking (Repacholi 1995).

Another major problem with these studies has been the lack of a clearly defined magnetic field measure (no concept of "dose" or the biologically active component of the magnetic field). In addition, many studies have had very low numbers of cases. To determine if a subtle effect exists on the incidence of rare cancers from magnetic field exposure, large numbers of cases and controls are needed. Of great concern has been the lack of consistency between study results. Some studies have suggested an increased leukaemia risk but not brain tumours while others have suggested exactly the opposite or reported no effect on cancer incidence at all. The problems with the 50/60 Hz epidemiological studies has lead to the necessity to conduct large animal studies to determine if indeed magnetic field exposure leads to an increased incidence of cancer.

ICNIRP has had to be very vigilent in this area because of the division of scientific opinion and more recently because of the unfortunate "leak" of a draft report from a National Commission on Radiological Protection and Measurement (NCRP) subcommittee responsible for identifying

health effects of ELF fields. Aside from the guidelines on exposure limits to 50/60 Hz fields issued in 1990 (INIRC/IRPA, 1990), ICNIRP reviewed the literature and reaffirmed the guidelines by issue a press release in 1993 (ICNIRP, 1993). ICNIRP is currently reviewing the scientific literature and drafting a set of guidelines that will provide exposure limits for electromagnetic fields from 0-300 GHz.

RADIOFREQUENCY FIELDS

It has also been suggected for many years that exposure to radiofrequency fields may also be associated with an increased incidence of cancer. Most recently ICNIRP has been confronted with concerns of people who find mobile telephone towers built near their home or schools. In addition there is the problem of the use of the mobile telephones themselves. They present an near-field exposure situation not anticipated by the RF guidelines published in 1988 (INIRC/IRPA 1988). In collaboration with WHO, INIRC/IRPA reviewed the RF literature and this was published in 1993 (UNEP/WHO/IRPA 1993). However, the issue of mobile telephones and their base stations resulted in the need to study possible effects from pulsed RF fields. ICNIRP has completed a review and is publishing the statement entitled "Health issues related to the use of hand-held radiotelephones and base transmitters" (ICNIRP 1996). A briefly summary of the reviewe is given below.

No consistent biological effect has been found in molecules or components of body cells exposed to RF fields, other than those effects caused by temperature increases. Researchers in the US reported that modulated microwave exposure of chromosomally abnormal cells which were treated with X-rays and a chemical promoter showed an accelerated rate of change from normal to cancer cells (Balcer-Kubiczek and Harrison 1991). While inconsistencies have been noted between various of these studies, the results are important. However, their implications for carcinogenesis in humans are not clear. This type of study tends to be susceptible to a variety of experimental confounding factors and needs independent confirmation.

Possible effects on DNA and chromosome structure are an important consideration in somatic cells, where such changes could lead to cell death or the development of cancers. If these changes occur in the male or female germ cells, surviving mutations might be passed on to the next generation. A large number of studies have been conducted in various somatic cells (Saunders et al 1991; NRPB 1992, 1993; UNEP/WHO/IRPA 1993) and most have reported a lack of effect on chromosome aberrations and single or double strand breaks in the DNA. Studies on the germ cells also suggests that acute or chronic exposure to RF does not result in increases in mutation or chromosome aberration frequency when the temperature is maintained within physiological limits. Where increased frequencies of chromosomal aberrations have been reported, these studies have not been successfully replicated. Chronic exposure experiments, which are relevant to long term RF exposure from base stations, have not produced any evidence of chromosomal aberrations in rodents exposed to SARs from 1-5 W/kg (UNEP/WHO/IRPA 1993).

In two recent rodent studies, there is the suggestion that RF fields may affect DNA directly. When mice were exposed to 2.45 GHz fields at 1 mW/cm^2 (SAR 1.18 W/kg) for 2 h/d for 120, 150 and 200 days, there was an indication of structural genomic rearrangement in brain and testes cells (Sarkar et al 1994). Lai and Singh (1995) report that rats exposed to pulsed (2 μs pulses, 500 pps) or continuous wave 2.45 GHz fields with SARs of 0.6 or 1.2 W/kg for 2 h

increased the number of single stand breaks in brain DNA. Both these papers produce quantitative data subject to sources of inter-trial variation and experimental error such as incomplete DNA digestion (Sarkar et al) or unusually high levels of background DNA fragmentation (Lai and Singh). These experiments should be replicated before the results can be used in any health risk assessment, especially given the weight of evidence suggesting that RF fields are not genotoxic (UNEP/WHO/IRPA1993).

A review of the laboratory studies conducted with WHO (UNEP/WHO/IRPA1993) concluded that RF field exposure is not mutagenic and is therefore unlikely to initiate damage directly to the DNA which would lead to cancer. The evidence for RF exposure causing tumour promotion or progression is not convincing but deserves further investigation.

Epidemiological studies have been conducted to assess the general health patterns and cancer risk among several groups of workers that were occupationally exposed to RF fields. These studies involved radar workers (Robinette et al 1980) and Moscow embassy workers chronically exposed to low-intensity microwaves for surveillance purposes (Lillienfeld et al. 1978). One of the more important studies conducted so far were the large epidemiological studies on Korean War radar technicians exposed to various levels of RF. No adverse health effects were established.

CONCLUSIONS

There is very little evidence to suggest that static fields have an effect on any stage of carcinogenesis. The major animal studies provide little convincing evidence that 50 Hz fields can promote cancer. There is some evidence that 50/60 Hz magnetic fields may be cancer co-promoters in tissues that are in the process of carcinogenesis, but the health impact of such a finding remains unclear.

With respect to whole-body RF exposure, the conclusions of the 1988 IRPA/INIRC guidelines remain valid. However, additional guidance has been given for mobile telephones. The conclusions of the ICNIRP (1996) statement are given below.

1. The results of published epidemiological studies do not form a basis for health hazard assessments of exposure to RF fields, neither can they be used for setting quantitative restrictions on human exposure. They do not provide a basis for hazard assessments in relation to the use of hand-held radiotelephones and base transmitters.

2. Data from laboratory studies relevant to cancer do not provide a basis for limiting exposure to the fields associated with the use of hand-held radiotelephones and base transmitters.

3. Limits for human exposure to the fields associated with the use of hand-held radiotelephones and base transmitters should be those of the INIRC (IRPA/INIRC 1988) for whole body average SAR and those of ICNIRP for localised SAR set out in this document.

4. There is no substantive evidence that adverse health effects, including cancer, can occur

in people exposed to levels at or below the limits on whole body average SAR recommended by INIRC (IRPA/INIRC 1988), or, at or below the ICNIRP limits for localised SAR set out in this document.

5. At the frequencies and power levels involved in the use of hand-held radiotelephones there will be no concern about shocks and burns.

6. The localised SARs in the head associated with the use of hand-held radiotelephones must be assessed for each frequency and configuration used.

7. For hand-held radiotelephones used in occupational situations, ICNIRP recommends that the localised SAR in the head be limited to 10 W kg^{-1} averaged over any 10 g mass of tissue in the head (0.1 W absorbed in any 10 g mass of tissue in the head).

8. For hand-held radiotelephones used by the general public, ICNIRP recommends that the localised SAR in the head be limited to 2 W kg^{-1} averaged over any 10 g mass of tissue in the head (0.02 W absorbed in any 10 g mass of tissue in the head).

9. The use of radiotelephones should be restricted to areas where interference effects are unlikely to occur (for example, well away from hospital intensive care departments and similar locations). Manufacturers of electrical equipment are encouraged to design and manufacture equipment that is insensitive to RF interference.

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Basic Principles for Intervention after a Nuclear or Radiological Emergency

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Abstract. The current status of internationally agreed principles for intervention after a nuclear accident or radiological emergency and the international development of intervention guidance since the Chernobyl accident are reviewed. The experience gained after the Chernobyl accident indicates that the international advice on intervention existing at the time of the Chernobyl accident was not fully understood by decision makers neither in Western Europe nor in the former USSR and that the guidance failed to address adequately the difficult social problems which can arise after a serious nuclear accident. The radiation protection philosophy of today distinguishes between practices and interventions. The radiological protection system of intervention includes justification of the protective action and optimization of the level of protection achieved by that action. Dose limits do not apply in intervention situations. The inputs to justification and optimization studies include factors that are related to radiological protection, whereas the final decisions on introduction of countermeasures would also depend on other factors. The basic principles for intervention as recommended by international organisations are discussed in detail and the application of the principles on a generic basis is illustrated for long-term protective actions. The concepts of intervention level, operational intervention level and action level are presented and the relation between these quantities is illustrated. The numerical guidance on intervention in a nuclear accident or radiological emergency or a chronic exposure situation given by ICRP, IAEA and in the Basic Safety Standards is presented.

1 Practices and interventions

The latest recommendations from the International Commission on Radiological Protection [1] outline the systems of protection for *practices* and *interventions*. Human activities that *add* radiation exposure to that which people normally incur due to background radiation, or that increase the likelihood of their incurring exposure, are termed *practices*. The human activities that seek to reduce the existing exposure, or the existing likelihood of incurring exposure which is not part of a controlled practice, are termed *interventions*. The concepts of *dose addition* in a practice and *dose reduction* in an intervention are shown in Fig. 1.

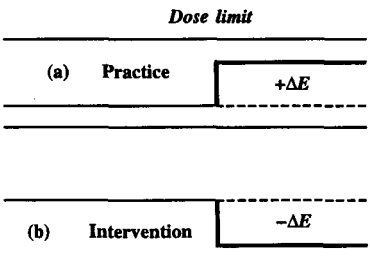

Fig. 1. Exposure situations for practices (a) and interventions (b). In a practice the dose increments $+\Delta E$ are controlled within the dose limit. In an intervention situation the dose decrement $-\Delta E$ or avertable dose is the result of the protective action.

For practices where the occurrence of the exposure is foreseen, the recommendations include the control of the source and limitation of exposure. Such situations include medical exposure, occupational exposure and exposure of the general public, and the system of protection includes the justification of the practice, optimisation of protection and imposition of overall dose limits.

In some situations the sources, the pathways and the exposed individuals are already in place when the decisions about control measures are being considered, and protection can therefore only be achieved by interventions, which always have some disadvantages. The system of protection for interventions include the justification of the intervention and the optimisation of the form, scale and duration of the intervention so as to maximize the net benefit.

The dose limits recommended by the Commission are intended for use in the control of practices. The use of these dose limits, or of any other pre-determined dose limits, as the basis for deciding on intervention might involve measures that would be out of all proportion to the benefit obtained and would then conflict with the principle of justification. The two different protection systems for practices and interventions are shown in Fig. 2.

Practice	**Intervention**
Each practice should be justified	**Each protective measure should be justified**
The doses *adding up* in a practice should be kept as low as reasonably achievable	**The level of protective measures resulting in *dose subtraction* should be optimized**
The *sum* of doses in a practice should be kept below specified dose limits	

Fig. 2. Systems of radiation protection for practices and interventions. The major differences between the two systems are that dose limits do not apply in the system of interventions and that different dose quantities apply.

2 Basic principles for intervention

An intervention should be *justified* in the sense that the introduction of protective measures should achieve *more good than harm*, and the level at which an intervention is introduced, and the level at which it is later withdrawn, should be *optimized* so that it will produce the maximum net benefit, i.e. *do the most good.*

A. JUSTIFICATION AND OPTIMIZATION OF PROTECTIVE ACTIONS

The avertable dose by the protective action, ΔE, can be found as the difference between the dose *without* any actions, and the dose *after implementation* of a protective action. The process of justification and optimization *both* apply to the protective action, so it is necessary to consider them *together* when reaching a decision.

Justification is the process of deciding that the disadvantages of each component of intervention, i.e. of each protective action or, in the case of accidents, each countermeasure, are more than offset by the reductions in the dose (avertable dose) likely to be achieved.

Optimization is the process of deciding on the method, scale and duration of the action so as to obtain the *maximum net benefit.* In simple terms, the difference between the disadvantages and the benefits, expressed in the same terms, e.g. monetary terms, should be positive for each countermeasure adopted and should be *maximised* by setting the details of that countermeasure.

Each of the factors describing the net benefit achieved by the protective measure have to be expressed in the same units. These units can be dimensionless quantities as used in multiattribute analysis, or values could be expressed in equivalent years of lost life. Normally in cost-benefit methods values are expressed in monetary units. However, it is the relative values placed on the components and their weighting one to another that is important, rather than the absolute unit. The use of a particular currency is relatively unimportant, as all terms could be evaluated as fractions of a country's gross national product (GNP) per head to allow for differences per head between countries.

If a protective measure were introduced at time t_1 and lifted at time t_2, the avertable dose, ΔE would be equal to the time-integral of the dose per unit time, $\dot{E}(t)$, over this time interval, τ.

The avertable effective dose, ΔE, and the effective dose per unit time, $\dot{E}$, are shown in Fig. 3.

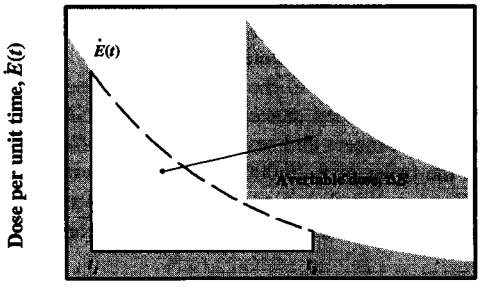

Fig. 3. Avertable dose, ΔE, when the protective measure is introduced at time t_1 and lifted again at time t_2.

B. INTERVENTION LEVELS

Intervention levels refer to the dose that is expected to be averted (avertable dose) by a *specific countermeasure* over the period it is in action. If an intervention level is exceeded, *ie*, if the expected avertable individual dose is greater than the intervention level, then it is indicated that the specific protective action is likely to be appropriate for that situation. Intervention levels (ILs) are specific to accident situations. The intervention level is defined as [5]:

> *Intervention level* is the level of *avertable dose* at which *a specific protective action* or *remedial action* is taken in an *emergency exposure* situation or a *chronic exposure* situation

Recommended intervention levels from the IAEA and ICRP for the major protective actions in a nuclear emergency are shown in Table 1 [2,3,5]. These recommendations are based on generic optimization as shown below.

Table 1. Summary of Recommended intervention levels from ICRP Publication 63 and IAEA Safety Series No. 109.

Protective measure	IAEA generically optimized intervention levels	ICRP range of optimized values
Sheltering (less than 1 day)	10 mSv	5 - 50 mSv
Administration of stable iodine	100 mGy to thyroid	50 - 500 mSv to thyroid
Evacuation (less than 1 week)	50 mSv	50 - 500 mSv
Temporary relocation	initiate at 30 mSv in a month and suspend at 10 mSv in a month	almost always justified at a dose level of 1 Sv 5 - 15 mSv/month as an optimized range
Permanent resettlement	if lifetime dose would exceed 1 Sv	

The optimized intervention level could either be expressed as an *individual dose, $\dot{E}$*, averted per unit time or as a *collective dose, S*, averted per unit mass of a given foodstuff, both given at the end of the interval, τ, for which the countermeasure, introduced at time t_1, would have to be in action [2,3,4]:

$$\dot{E}(t_1+\tau) = \frac{c}{\alpha} \quad \text{or} \quad S(t_1+\tau) = \frac{b}{\alpha}$$

The costs parameters in the above expressions, c (cost per unit time of implementing a given

countermeasure), b (cost per unit mass of restricting a given foodstuff) and α (monetary value of a unit dose averted) are likely to be similarly correlated to national wealth and thus susceptible to a relatively large variation between countries. However, their ratios would in general be much less sensitive to geographical location than either of the cost parameters alone.

The monetary value of a unit dose, α, would be related to the risk per unit dose, R, and the statistical loss of life expectancy, ΔL, per radiation induced cancer as $\alpha = R\ \Delta L\ GNP$, with some allowance for loss of quality of life for non-fatal cancers and severe hereditary effects. For rich developed countries the value of *GNP* is of the order of \$ 25,000 year^{-1} which would give a reference value of α of \$ 25,000 per sievert. For less developed countries α-values would be correspondingly lower. Generically optimized intervention levels would, however, be more or less country independent.

C. OPERATIONAL INTERVENTION LEVELS

Because of the inherent difficulty of forecasting doses that could be averted, there is a merit in establishing surrogate quantities such as *dose rate in air*, *surface contamination density* and *activity concentration in air*. The relationship between these quantities and the avertable dose will vary considerably with the circumstances of the accident and nature of contamination. The operational quantities would, therefore, be both accident and site specific but would still be inextricably linked to the avertable dose.

The term "*operational intervention level (OIL)*" is reserved for quantities that can be more easily assessed at the time of decision on intervention such as dose rate, activity concentration, surface contamination density, etc. OILs are related to the dose that could be averted by a *specific protective action* like evacuation, relocation and banning of foodstuffs.

In general terms, the avertable dose, ΔE, from *all exposure pathways* by implementing a given countermeasure can be expressed by a measurable quantity, q. The operational intervention level can be determined from the intervention level (IL) as follows:

$$OIL = \frac{IL}{\sum_{pathways} \Delta E(q=1)}$$

It should be recognised that in the calculation of $\Sigma\Delta E(q=1)$, summed over all pathways, p, site specific parameters like location/filtration factors and indoor/outdoor occupancy have to be used. For long-lived γ-emitting radionuclides like $^{137}Cs/^{134}Cs$ the value of $\Sigma\Delta E(q=1)$ would be approximately 200 mSv·month^{-1}/mSv·h^{-1} for a time-averaged location factor of 0.3, q being the outdoor dose rate. The OIL_{rel} for relocation corresponding to an IL of 10 mSv·month^{-1} can thus be calculated from the above formula to be about 50 μSv·h^{-1}. When the outdoor dose rate from long-lived γ-emitting radionuclides therefore exceeds 50 μSv·h^{-1}, it is indicated that temporary relocation might be needed.

D. ACTION LEVELS

Action levels refer to different protective measures or strategies of actions like agricultural countermeasures or radon reducing measures in houses and they relate to the residual dose without remedial actions. The action level is defined as [5]:

> *Action level* is the level of *dose rate* or *activity concentration* above which *remedial actions* or *protective actions* should be carried out in *chronic* or *emergency exposure* situations

Action levels are levels above which remedial actions are taken and below which they are not. An action level is set such that the dose averted by taking the remedial action is always worthwhile in terms of the costs and other disadvantages involved. Justified action levels would

begin at the minimum value of the avertable individual dose at which the remedial action is just beginning to do more good than harm.

The action level can thus be defined as the lowest level at which remedial actions to reduce doses are justified and optimized. The equivalent definition would be that the action level is equal to the maximum acceptable level of dose attributable to the contamination without any protective actions taken. If an action level is exceeded, it is indicated that some form of remedial action specific to the situation considered is likely to be appropriate. Action levels have, therefore, the same character as operational intervention levels. The concept of an action level is illustrated in Fig. 4.

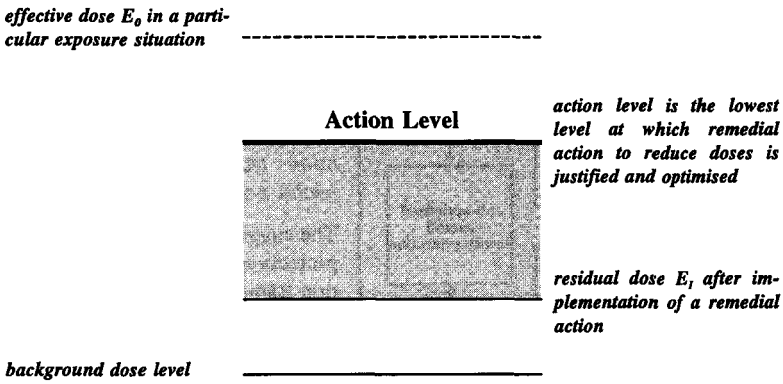

Fig. 4. The action level (AL) is the maximum acceptable residual dose without a remedial action which is equivalent to the minimum avertable dose by that action. The avertable dose in the figure is given as $\Delta E = E_0 - E_I$.

Action levels would also be used for the reduction of radon in dwellings. Recommended action levels from ICRP for radon in dwellings are shown in Table 2 [6]. The action level for remedial actions relating to chronic exposure situations involving radon in workplaces is a yearly average concentration of ^{222}Rn of 1,000 $Bq \cdot m^{-3}$ in air.

Annual dose (mSv)	3 - 10
Concentration ($Bq \cdot m^{-3}$)	200 - 600

Table 2. Action levels for radon in dwellings.

Bone marrow	0.4
Lens of the eye	0.1
Gonads	0.2

Table 3. Action levels for chronic exposure of single organs (Gy y^{-1}*).*

Different tissues have a wide range of sensitivity for radiation induced deterministic effects. Threshold dose rates for deterministic effects under conditions of prolonged exposure over many years have been used in the Basic Safety Standards (BSS) [5] as action levels for chronic exposure. The values for permanent sterility, for clinically significant depression of the blood-forming process and for opacities sufficient to cause impairment of vision are shown as action levels in Table 3 [1,5].

3 Unresolved issues

Following a nuclear or radiological emergency, especially in the later phases, many complex human, social and economic considerations will have to be taken into account by the responsible authorities. The decisions and protective actions taken may themselves induce social and

psychological impacts. Internationally, the application of different intervention levels in similar circumstances resulting from a single accident would cause much confusion in the public mind. At the national level, taking decisions about lines of demarcation between those areas where protective measures are applied and those where they are not, might create anxiety or even fear by people living on the 'safe' side of the demarcation line.

From the experience in CIS following the Chernobyl accident, countermeasures to mitigate social-psychological impacts have obviously been needed and they have been identified to be a new category of action. It has even been argued that radiation protection philosophy has not yet been developed to fully include these countermeasures, as being reflected in Fig. 5, but that the two 'lines' of optimization should be merged into one system of 'radiation protection'.

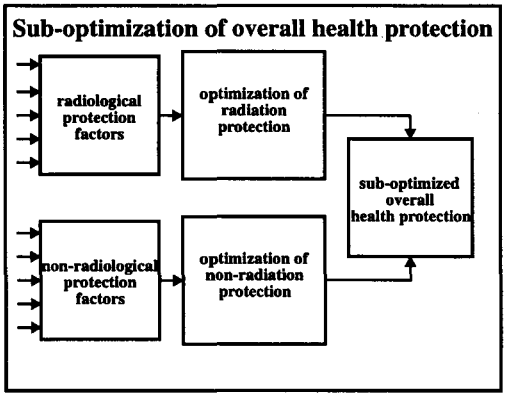

Figure 5. Sub-optimization of overall health protection where radiological protection and non-radiological protection factors each are used for separate optimizations leading to a sub-optimized overall health protection.

The suggestion to include non-radiological protection factors in the radiation protection framework seems awkward. Radiation protection factors are related to the level of protection achieved including those factors describing the dose distribution averted and costs and other disadvantages incurred in averting doses. The level of socio-psychological impact would depend not only on the presence of radiation but to a large extent on other non-radiological protection factors, such as the attitude of the mass media, the political climate and the general level of information in the population. Non-radiological protection factors would also be different from country to country, and would probably be highly dependent on the existing political situation. To include socio-psychological factors in the radiation protection framework would thus give very random levels of 'radiation protection'.

Therefore, to achieve an optimized overall health protection, non-radiological protection factors should enter the optimization process in parallel with radiological protection factors to form an optimized countermeasure strategy as shown in Fig. 6.

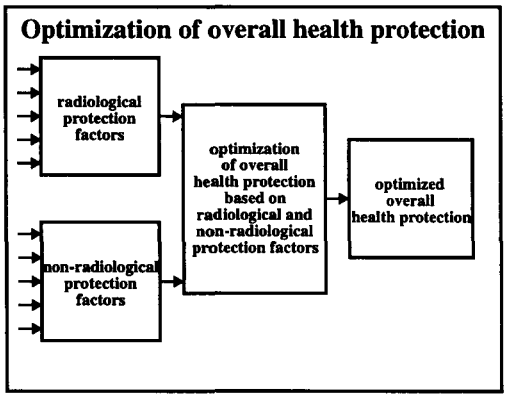

Figure 6. Optimization of overall health protection where radiological and non-radiological protection factors are included simultaneously in the optimization process.

Optimization of overall health protection would thus be the responsibility of the decision maker with guidance from radiation protection experts as well as experts in the fields of social and psychological sciences. Optimization of overall health protection is therefore a discipline for decision makers and not an extension of the radiation protection philosophy. It is also of great importance that decision makers present the optimized protection strategy to the public in a transparent way so that all the factors and their relative importance in reaching the optimized strategy are revealed.

4 Summary

The distinction between practices and interventions has created some confusion. The fact that dose levels for introduction of protective measures have been interpreted as doses *received* and not as doses *averted* is, however, awkward and prone to suggest to people that the levels are dose limits. The experience after the Chernobyl accident was that many actions taken led to an unnecessarily large expenditure of national resources, and many instances occurred of contradictory national responses. Therefore, there was a strong need for a simple set of internationally consistent intervention levels and action levels and for clear guidance on application of the principles in planning and preparedness for response to nuclear accidents or radiological emergencies.

Intervention levels have therefore been established by ICRP and IAEA for both short-term and longer-term countermeasures. These intervention levels are expressed in terms of dose *averted* by the specific action. Since the dose that will be averted cannot easily be estimated in the period immediately after the accident, derived quantities of *operational intervention levels* (*OILs*) or *action levels* (*ALs*) can be established (dose rate or activity concentration, for example). The appropriate protective action will be invoked if these levels are exceeded, but the action will normally not be taken if they are not.

There are still unresolved issues for different intervention situations, especially for long-term exposure situations; these issues would involve both optimization of radiation protection and optimization of overall health protection:

- So far, no international agreed numerical guidance have been established on action levels for protection of populations against different chronic or semi-chronic exposure situations from natural and artificial radionuclides in the environment (except for radon in dwellings).
- Clean-up levels for residues from previous events or past practices, *eg*, uranium mining and milling, might - due to their origin from intervention optimization - result in a lower level of protection than for still operating practices of a similar type because of the dose constraints for practices. It would be peculiar to operate with different levels of protection in two similar situations and, obviously, optimized action levels for such intervention situations might have to be constrained to an equal protection level as in the similar practice.
- Interaction between radiological and non-radiological factors in decision making has been identified to be an important issue. Both radiological and non-radiological factors will influence the level of protective actions being introduced. From the experience in CIS following the Chernobyl accident, the need for social-psychological countermeasures is obvious. However, the quantification of non-radiological protection factors needs further development.
- It has been suggested that the inclusion of socio-psychological factors into the intervention decision making framework should be as a part of the radiation protection framework. This suggestion seems awkward as the level of socio-psychological impact would depend not only on the presence of radiation but to a large extent on other non-radiological protection factors, such as the attitude of the mass media, the political climate and the general level of information in the population. Inclusion of socio-psychological factors in the radiation protection framework would therefore give very random levels of 'radiation protection'.
- Non-radiological protection factors such as public anxiety and risk perception will play a legitimate role in the decision making. These factors should be addressed by the decision maker and *not* by the radiation protection community. The optimization of overall health protection is thus the responsibility of the decision maker. It is of great importance that the decision maker present the protection strategy to the public in a transparent way so that all factors and their relative importance in reaching an optimized strategy are revealed.

There are at least two areas where the radiation protection community should clarify its

recommendations in the years to come. Firstly, convincing guidance needs to be prepared for the 'grey area' of similar intervention/practice situations for which the optimized level of radiation protection of populations might be different. Secondly, only radiological protection factors should be included in generically optimized intervention levels. The socio-psychological and political factors should be included by *the decision maker* in the optimization of overall health protection based on inputs from radiation protection experts and from experts in the fields of social and psychological sciences.

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COUNTERMEASURES IN AGRICULTURE AND FORESTRY AFTER A NUCLEAR ACCIDENT

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INTRODUCTION

During the past 35 years, three major accidents at nuclear facilities have given cause for concern and required the implementation of countermeasures in agriculture. These accidents occurred at Windscale (United Kingdom) in 1957, Kyshtym (former Soviet Union) in 1957 and Chernobyl (former Soviet Union) in 1986. Despite the increased knowledge on the safety of nuclear power plants and the implementation of improved procedures, it would be naive to exclude the possibility of other accidents occurring in the future and the consequent deposition of radioactive material into the environment.

As nuclear facilities are not normally sited in densely population areas, the adjacent land is likely to be agricultural or at least rural. Furthermore, the accident at Chernobyl, albeit a severe one, showed that agricultural practices can be affected hundreds and even thousands of kilometres from the accident site. Therefore contingency plans are needed to initiate countermeasures that can be used to reduce contamination of agricultural produce, even in countries with no nuclear power of their own.

This paper presents an overview of the principles for intervention to protect the public from the perspective of agricultural countermeasures, the development of generic and specific intervention and action levels, the range of countermeasures available at various times after an aciident and some of the factors involved in developing an overall strategy for decision making and implementation. For further information on these subjects the reader is referred to an IAEA/FAO Technical Report(1).

PROTECTIVE MEASURES

In the event of a nuclear accident, the ICRP principles relating to intervention situations apply (2). In these circumstances, three major principles have evolved to protect and maintain human health against the effects associated with radiation exposure (3); these principles appear to have almost universal acceptance:

- a) All possible efforts should be made to prevent serious deterministic health effects;
- b) The intervention should be justified, in the sense that introduction of the protective measure should achieve more good than harm.
- c) The levels at which the intervention is introduced and at which it is later withdrawn should be optmised, so that the protective measure will produce a maximum net benefit.

In practice, agricultural countermeasures normally address stochastic health effects in the human population, although the more immediate impact of possible deterministic effects, particularly on farm animals, should be considered. In general, countermeasures to protect foodstuffs will normally be taken at levels much lower than those at which such effects occur (4).

In the event of a nuclear accident, the effectiveness of measures taken to protect the agricultural sector (people, land, crops and livestock) will depend heavily upon the adequacy of emergency plans prepared in advance. In these emergency plans, criteria are specified for taking particular prompt actions. Even after the immediate emergency, having predefined criteria for longer term actions will do much to counter any loss of confidence in the competence and integrity of the authorities on the part of the public. Such criteria for intervention are based primarily on radiological protection principles.

There are two distinct phases in which optimisation of protective measures should be considered:

- In the phase of planning and preparation, prior to accidents, a *generic* optimisation of protective actions should be established, based on *generic* accident scenario calculations. This should result, for each protective measure and each selected scenario, in an optimised *generic* intervention level, which is meant to be the first criterion for action to be used immediately and for a short time after the occurrence of an accident.
- Some time after the beginning of a real accident, specific information on its nature and likely consequences and evolution would be expected to be available. in this case a more precise and *specific* optimisation analysis should be carried out in the basis of actual data and the actual efficiency of protective measures. This should result in a *specific* intervention level for each protective measure, to be used as a criteria in the medium and long term. However, in many cases the optimisation will be constrained by sociopolitical factors, which may make it difficult to alter the generic intervention levels unless there are overriding reasons to do so.

Radionuclide Group	*Optimized intervention level (Bq/kg)*				
	Specifically for withdrawal of foodstuff		Specifically for substitution of clean fodder		
	Fresh milk, vegetables, grain, fruit	Meat, milk products	Milk		Meat
Group 1	~1000 - ~10,000	~10,000 - ~100,000	~100 - ~1000	Cs-137	~few 100 - several 1000
				I-131	~100 or so - ~1000 or so
Group 2	~100 - ~1000	~1000 - ~10,000	~10 - ~100	Sr-90	Several 10s - several 1000s
Group 3	~10 - ~100	~100 - ~1000	~1 - ~10	Pu-239	Few 100 - few 1000

Table 1 Optimised intervention levels for food. Optimised intervention levels for various food categories and nuclide groups have been shown to be of the order given in the table. The nuclide groups are delineated according to the effective dose per unit activity ingested: Group 1 - ~10^{-8} Sv/Bq; Group 2 - ~10^{-7} Sv/Bq; Group 3 - ~10^{-6} Sv/Bq.

Commonly, situations will arise where contaminated food and water can be withdrawn and uncontaminated replacements can be readily substituted, at costs approximately equal to the value of the foodstuff without seriously distorting supply and demand and without introducing major disruption of lifestyles. In these cases it is reasonable as a first approximation to apply simple methods to estimate optimized intervention levels specifically for withdrawing food (1). Ranges of values are given in Table 1. Whilst the optimised values have been derived on the

assumption of withdrawing contaminated foodstuffs and substituting uncontaminated ones, there exist a wide range of other less extreme measures that are often more cost effective. Also given in Table 1 are ranges for optimised intervention levels for providing clean feed to cattle otherwise grazing contaminated land. The latter set of values are significantly lower than the set for a complete food ban.

GENERIC ACTION LEVELS FOR FOODS

Whilst it is possible to develop intervention levels for each agricultural countermeasure of interest, it is more practical to establish action levels, which are not specified for any particular countermeasure, but are defined as levels above which some action is needed.

There are a number of advantages in using internationally recognized values for the action levels: 1) maintaining credibility, confidence and trust in the authorities; 2) preventing anomalies that might otherwise exist along borders of neighbouring countries; 3) strong arguments can be made to adopt international values as national intervention levels for the control of food.

The Codex Alimentarius Commission of the FAO and WHO has addressed the situation of international standards in order to maintain widespread international trading in food. Guideline levels for radionuclides in international trade following accidental nuclear contamination have been agreed (5). It should be recognised that these levels are a compromise between what is appropriate on radiological protection grounds (which would often give rise to significantly higher values) and the natural wish of countries unaffected by an accident to avoid importing produce with any contamination at all, no matter how small, even compared with natural radiation. The values have been adopted by the FAO, IAEA, ILO, OECD/NEA, PAHO and WHO in developing the new International Basic Safety Standards for Radiation Protection (6) and are presented in Table 2.

Radionuclides	*Foods destined for general consumption (kBq/kg)*	*Milk, infant foods and drinking water (kBq/kg)*
Cs-134, Cs-137, Ru-103, Ru-106, Sr-89	1	1
I-131		0.1
Sr-90	0.1	
Am-241, Pu-238, Pu-239	0.01	0.001

Table 2 Generic Action Levels for Foodstuffs. For practical reasons, the criteria for separate radionuclide groups shall be applied independently to the sum of the activities of the radionuclides in each group.

It should always be recognised that accident and site specific conditions, as well as political considerations, might lead to different *specific* action levels. In particular, where the numbers of people and area of land potentially affected by a protective action become extremely large, the cost of resources and societal disruption become more and more important in comparison with available national resources, and relaxation of the intervention levels may be necessary. On the other hand, when the numbers of people and area of land potentially affected are very small, additional costs incurred in order to gain public confidence can be readily absorbed by society.

An indication of how the costs and doses associated with a countermeasure depend on an intervention/action level is given in Figure 1. This figure shows for example, that increasing an action level by a factor of two also increases the dose to the hypothetical critical group for that pathway by a factor of two, assuming it is the dominant pathway (7). However the collective dose saved is lower by only 20% while the associated costs fall by 60%. Similarly reducing an intervention/action level by a factor of ten will decrease the dose to the hypothetical critical group for that pathway also by a factor of ten (assuming it is the dominant pathway). However, the collective dose saved is increased by only a factor of two, while the costs increase by a factor of 20.

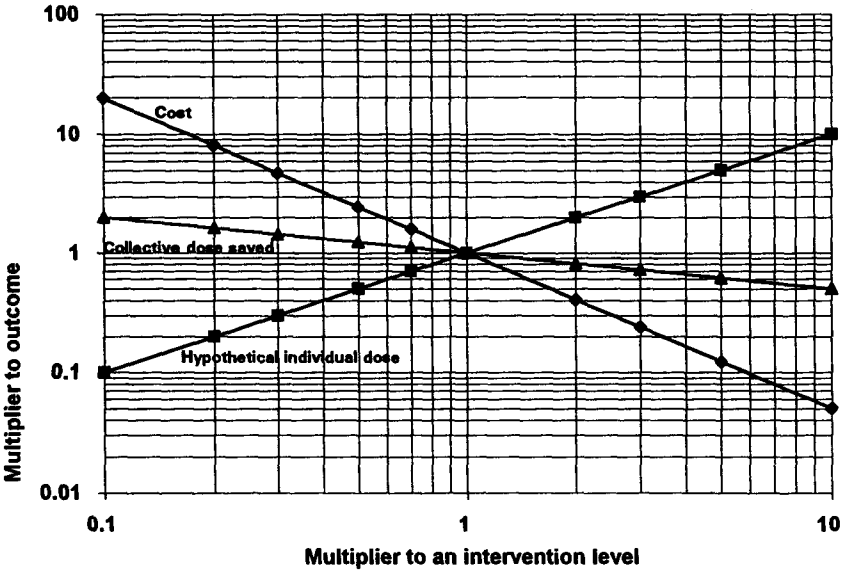

Figure 1. The effects of modifications to an intervention/action level on the cost and collective dose saved by a countermeasure and the residual dose to a hypothetical critical individual.

COUNTERMEASURES STRATEGY AND DECISION MAKING

The main considerations for government (from central down to local) in preparing an agricultural countermeasures strategy is to :

- define measures to be applied before and during fallout, and in the consequent medium and long term;
- protect human health by reducing radioactive contamination of agricultural products;
- return the land to normal use as far as possible;
- ensure that the countermeasures applied balance health protection measures, cost, disruption to daily life and the well-being of communities.

For the early phase of an accident, authorities will rely in the main on their own emergency plans. These plans will have been drawn up prior to an accident; the need for elaborate decision making in real time on the introduction of countermeasures will to a large extent be obviated by the existence of plans and there will anyway be a lack of time in which to react. After the early phase is over and as more detailed information becomes available and the potential for large expenditure becomes more apparent, more considered decisions will need to be made about the need for specific longer term countermeasures and/or the relaxation of countermeasures. These may well involve the use of decision aiding tools.

There can be no single generic strategy for the introduction of agricultural countermeasures because environmental conditions vary from place to place, and the nature and size of accidental release, the size of the affected population and the social, economic and agricultural conditions will all differ. As discussed earlier, the main objective of a countermeasure should be to effectively reduce radiation doses to the population and hence diminish health risks. In practice, this translates into selecting a countermeasure strategy that will allow production of food with activity levels below the action levels in as cost effective a manner as possible, with minimum side effects. On this basis the simplest way to compare the cost effectiveness of countermeasure options is to calculate the monetary cost per unit collective dose saved. However, the side effects of some countermeasures, which can include ecological, economic and social factors can be substantial. For example, agriculture could become unsustainable, farmers and their dependent communities could be made redundant, and the psychological consequences could be large, even leading to clinical health problems. There may be practical reasons why countermeasures must be adopted in a particular area in order to limit the spread of contaminated produce outside that area. Markets could be severely distorted in the short to medium term by the introduction of a particular strategy. Subsidies and compensation to farmers may introduce their own market distortions. These additional factors will have to be taken into account in the drawing up of recommendations to farmers. Some sophisticated techniques exist that can be used to aid decisions between different alternatives. Nevertheless, these methods can only be used if the problem is relatively simple and well structured, and persons with specialist knowledge are available. Such methods have not as yet been used in practice for deciding on agricultural countermeasures; decisions have to date been based on the cost effectiveness of the particular countermeasure and a subjective weighing of other factors.

AGRICULTURAL COUNTERMEASURES

The timing and nature of countermeasures depend on the quantity and composition of the radionuclides in the release, the season of the year and site specific factors. The post-accident period can be conveniently divided into three overlapping phases. Examples are given below of the measures available. For further details the reader is referred to Ref. (1).

Early Phase

The early phase may begin before fallout arrives if sufficient warning is available, and ends when the 'cloud' has passed and the acute deposition finished. It is self-evident that countermeasures taken during this phase must be applied promptly to be effective. However, since they will be based on predictions derived from incomplete information there is a high degree of uncertainty. There will be a tendency to be cautious and therefore countermeasures invoked should be inexpensive and with only short term implications in case they turn out to have been unwarranted.

Countermeasures that might be invoked in this phase include:

- Generic ban on collection of milk and other products pending monitoring;
- Housing animals that are grazing outdoors (or preventing the return of stalled animals to pasture) and providing uncontaminated forage prevents the contamination of milk, meat and offal. Moreover, it may, in the event of a large radioiodine release protect against acute deterministic effects to the animals' thyroids;
- Covering uncovered feed/food stores and open sources of water with impermeable sheeting. Covering haystacks, heaps of potatoes etc. with plastic sheeting is effective and relatively easy to arrange where suitable covering is commonly available;
- Harvesting ripe crops before contamination occurs: this is only possible where there is sufficient warning available. Priority would normally be given to high value crops.

Intermediate Phase

The intermediate phase lasts several weeks or months until contamination from short-lived radionuclides or from the direct deposition is no longer of concern or the first crop is harvested. Countermeasures may be needed to deal with the direct deposition of activity onto crops or pastures, and the subsequent contamination of food products.

Typical countermeasures that can be efficient in this phase include:

- Prohibiting hunting, fishing, mushroom collection, and consumption of vegetables and water derived from surface water or precipitation;
- Harvesting crops and grass for disposal or storage; or
- Delaying the harvesting of forage/crops;
- Processing milk contaminated with I-131 to storable products, e.g. butter, skimmed milk powder, cheese etc.

Late Phase

In the late phase the radioactive contamination is mainly due to long lived radionuclides and contamination of foodstuffs occurs primarily through root uptake of radionuclides by plants. The need to implement long term agricultural countermeasures needs to be carefully assessed since any actions are likely to have to last for long periods with often high associated costs.

Typical countermeasures in the late phase include:

- Ploughing to dilute the radionuclides in the rooting layer;
- Removing a shallow surface layer of the contaminated soil;
- Selecting suitable varieties of a crop, or alternative crops that accumulate lower levels of the contaminating radionculide than the variety normally grown in the area;
- Growing crops such as flax, cotton, ornamental plants or oil seed rape where the product is non-edible;
- Using contaminated land for non-dairy animals or for animals non intended for immediate slaughter;
- Changing land use to forestry;
- Applying potassium fertilizers;
- Changing slaughter time to reduce contamination levels in animals;

- Administering chemical binders to reduce radiocaesium absorption in the gut of grazing animals;
- Prohibiting fishing;
- Restricting access to forests;
- Changing forest management, including hunting schedules, reforestation etc.;
- Optimising food processing to minimise the contamination of final products.

LEGAL CONSIDERATIONS

A clear legal basis should be established for intervention in the event of an accident. Prevailing food law and consumer protection legislation should be sufficiently comprehensive so that it is not necessary to pass any new laws in the period immediately following an accident.

Specific numerical action levels for radioactive contamination in food in an emergency should be laid down in advance. The internationally agreed FAO/WHO *Codex Alimentarius* Commission Guidelines for foods moving in international trade mentioned above will form the basis for these limits. Such levels may have to be reviewed in the light of the specific conditions of an accident within a predefined period of time (e.g. 3 months).

Legal aspects specifically related to agricultural countermeasures that should be considered include:

- any legal obstacles to the implementation of countermeasures
- the need to license chemicals planned for soil treatment or feed additives (e.g. Prussian Blue)
- legal action that may be necessary to ensure the stockpiling of uncontaminated feed, chemical agents, etc.
- the legal basis for compensation for losses in agriculture following accidental contaminate of farmlands
- the liability of an operator responsible for an accident.

FUTURE DIRECTIONS

There is a need for authorities to develop secondary reference levels (so-called 'operational intervention levels') for quantities such as animal feeds and levels of contamination of pasture on which animals should normally graze. These working quantities are specified in their own appropriate units (e.g. Bq/m^2). International guidance on these is to be expected.

Also, whereas parameter values for the prediction of radionuclide transfers exist for the soil-plant system in temperate environments (8), such values are not currently available for warmer climates and sub-tropical soils and plants. The IAEA has recently initiated a Coordinated Research Programme to predict radionuclide transfer in these environments.

The application of formal decision aiding techniques in various scenarios that can encountered following an accident will likely lead to the development of more generic guidance on countermeasures strategies for various scenarios and environments in the future. The experience gained in the former Soviet Union following the Chernobyl accident will be invaluable in the validation of these techniques.

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Decision Making on Remedial Actions after a Nuclear or Radiological Emergency

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Abstract. Guidance developed at an international level for longer term countermeasures (eg, relocation, decontamination, restoration of contaminated areas, removal of countermeasures, etc) is largely outline in nature. The considered view is that, for such countermeasures, sufficient time will be available for the benefits and disadvantages of different options to be carefully weighed and balanced in the circumstances then prevailing. The wide range of circumstances that might be encountered after an accident is also judged to constrain the degree to which detailed, practicable guidance can be prepared in advance. The availability of more time, however, is unlikely to make the process of deciding what remedial measures to adopt easier, indeed probably the reverse. Various approaches and techniques are available to aid the development of more detailed guidance in advance of an accident and/or decisions on intervention following an accident. These are summarised with particular reference to their relative strengths and weaknesses. Consideration is also given to whether these techniques should find greater application in advance of an accident with a view to facilitating the development of more detailed, practical guidance should the need ever arise.

1. Introduction

In general, for longer term countermeasures (eg, relocation, decontamination, restoration of contaminated areas, removal of countermeasures, etc) only outline guidance has been developed for use in the event of an accident. The considered view is that, for such countermeasures, sufficient time will be available for the benefits and disadvantages of different options to be carefully weighed and balanced taking account of the particular circumstances then prevailing. The wide range of circumstances that might be encountered after an accident is judged to constrain the degree to which detailed, practicable guidance can be prepared in advance. The availability of more time, however, is unlikely to make the process of deciding what remedial measures to adopt easier, indeed probably the reverse. Pressures will be exerted by various interest groups with differing and probably conflicting objectives that will be difficult to reconcile, in particular as the issues would be exposed to wide public debate and media attention.

Two issues are addressed in this paper: firstly, the merits of developing more detailed, practical guidance for long term countermeasures in advance of an accident and, secondly, techniques that could be used to facilitate the development of such guidance, irrespective of whether this was prior to or after an accident had occurred.

2. Development of more detailed guidance

Only limited guidance is available internationally on longer term countermeasures and that which does exist is largely outline in nature and constrained in the range of factors that have been considered in its development. Various international bodies have issued quantitative guidance on relocation [1-3] but little yet exists on the restoration of contaminated land or the so called "return to normality" when all remedial measures taken after an accident are removed. These latter topics are, however, currently being addressed by the IAEA, ICRP and others. The factors taken into account in developing the current quantitative guidance on relocation were essentially limited to the monetary cost of the intervention and the resulting decrease in radiation health detriment. Such guidance must, therefore, be qualified with respect to its practical application. The qualifications placed on the guidance developed in reference [1] include the following: *"... The guidance developed here has been kept deliberately simple and is intended to be broadly applicable. Of necessity, it will need to be refined in light of the particular circumstances of an accident and to take account of other factors of a more social nature that have not been addressed here. The guidance is intended to provide a basic*

framework for decision making within which other factors or considerations could be readily integrated ..... These levels (ie, the quantitative guidance) *provide a useful starting point for making decisions on relocation"*

In practice a number of other factors or considerations will need to be taken into account in determining both the nature and extent of longer term interventions following an accident. In their absence, both the credibility and acceptability of the approach or criteria established are likely to be strongly challenged with consequential implications. This is exemplified by experience following the Chernobyl accident and the contamination of land by other carcinogenic or toxic agents. It is difficult to be prescriptive or complete with respect to the other factors to be considered since much will depend on the circumstances prevailing at the time of the accident, in particular the social, economic and political situation. Some of the more obvious factors that will need to be addressed, in the context of relocation, include the following:

- the health detriment resulting from relocation itself (ie, there is some evidence that the imposed movement of people leads to a reduced life expectancy)
- anxiety and stress consequent upon living in a "contaminated" environment
- pressures for compensation for damage caused by an accident and from continuing to live in a "contaminated" area over the long term
- the apparent conflict between radiological standards used for practices and intervention (notwithstanding their intellectual soundness, these differences are perceived very differently by the layman compared to the radiation protection community and are a continuing source of difficulty in practical policy formulation)
- utilisation of resources and constraints, particularly achieving a balance in terms of expenditure on other health and environmental risks
- use of the issue as part of a broader political objective.

These, together with other potentially important factors, are considered more fully in another of the papers presented in this IRPA Symposium [4].

Because of the diversity of these factors and their potential variability (ie, depending on the scale and extent of an accident and the prevailing social, economic and political conditions), there are clearly limits to the level to which detailed planning for relocation can be made or, indeed, would be sensible. At an international level [1-3] there is a broad consensus on both the level of detail that is appropriate in developing guidance on relocation (ie, outline guidance taking account essentially of only two factors, monetary cost and the reduction in radiation health detriment from the intervention) and on the quantitative guidance itself. Whether more detailed guidance should be developed in advance of an accident at a national level, however, is less clear and subject to judgement. This issue is finely balanced and judgements on it may be influenced, *inter alia*, by the following considerations:

- the degree to which it is possible to foresee those additional factors that may significantly influence relocation policy and the extent to which their influence can be evaluated (ie, quantitatively or qualitatively) with reasonable confidence in advance of an accident
- the scope for developing generically applicable guidance at a national level (ie, would it be precluded because of marked sensitivity to the nature and scale of the accident scenario, to variations in socio-economic conditions within a country, etc)
- would the resources needed for the development of more detailed guidance (in particular the significant demands it would place on those at a decision or policy making level) be justified in relation to the very low probability claimed for an accident of magnitude sufficient to warrant relocation.

Clearly, there is scope for very different judgements on each of these issues (ie, depending on national conditions and the level of risk foreseen from various types of nuclear installation). This has consequential implications for judgements on whether resources should be allocated to developing more detailed guidance in advance of an accident or whether, on balance, these matters would best be dealt with at the time an accident occurred.

Notwithstanding what judgement is reached on the above issue, there can be little or no dispute over the need to have in place, in advance of an accident, effective procedures to transform outline guidance into a practicable and generally acceptable policy taking due account of other influential factors. These procedures should, *inter alia*, address organisational and methodological issues and also identify the key information needed as an input to this process; to the extent practicable, this information should be assembled in advance. Ideally, these procedures should be exercised to demonstrate their efficacy and limitations; such exercises would provide useful insights into the degree to which more detailed guidance could be developed in advance and also a useful forum for identifying other factors that may need to be taken into account in the development of practical policies.

The methodological approaches or techniques that can be used to assist the development of more detailed guidance are essentially the same irrespective of whether the guidance is developed in advance of or in response to an accident. Some of these techniques are summarised in the next Section, with particular reference to the use of multi-attribute decision analysis techniques.

3. Decision aiding methods

Many techniques are available to assist in the resolution of complex decision problems with each having different strengths and weaknesses. No simple generally applicable guidance can be given on which approach is best: much will depend on the nature of the application and the preferences and experience of the analyst and/or those responsible for the decision. Cost benefit or cost effectiveness analysis is perhaps the simplest form of decision aid and has found widescale use. Inevitably it requires the values of all attributes relevant to a particular decision problem to be expressed in monetary terms. In principle this can be done but, in some cases, it can be difficult where "non tangible" attributes (eg, aesthetics, public acceptability, etc) play a major role in the decision. Strong opposition to using the approach is often experienced on ethical grounds, in particular in those cases where a trade-off has to be made between monetary cost and human life and suffering (ie, an explicit or implicit monetary cost is placed on human life). Such criticism, however, is not unique to cost benefit analysis; it can equally be levelled against all decision analytic techniques where trade-offs have to be made involving such matters; however, by its very nature, the trade off in cost benefit analysis is often more explicit and visible and, therefore, more open to criticism.

Cost benefit or cost effectiveness analysis is most suited for use in decision problems which involve only tangible attributes which, in general, can be readily transformed into monetary terms. Guidance on the use of the technique in radiation protection can be found, *inter alia*, in [5, 6] and is not discussed further here. Cost benefit analysis has found increasing use over the past decade or so as a useful aid to the optimisation of protection and resource allocation more generally. Where non-tangible attributes (eg, socio-political considerations, etc) are important, techniques such as multi attribute value or utility analysis (MAVA) are more appropriate. There are many forms of multi-attribute decision aiding techniques (eg, see [7-9]) and a judgement has to be made on which technique to use and in what manner it should be applied. Much again will depend on the nature of the problem to be addressed and in what forum it is to be resolved. These aspects, however, are largely outside the scope of this paper, as is the theory of multi-attribute analysis itself and limitations on its applicability. The focus here is on the generalities of the MAVA approach and how it can be applied, with relative simplicity, using commercially available software to aid the resolution of complex decision problems. To demonstrate the potential of the technique, particular reference is made here to its use within the framework of the International Chernobyl Project [10] to evaluate and clarify the policy options for relocation then being considered in the former Soviet Union in response to the Chernobyl accident.

3.1 Multi-attribute value analysis

The first stage in the use of MAVA (or for that matter in any decision problem) is to identify the set of attributes which describe the success or otherwise of each option or alternative that is being considered. In other words, the first stage is to answer the question: what factors matter to the analyst or decision maker/s in considering the relative merits of the alternatives? The second stage is to establish an attribute hierarchy to facilitate the analysis of the decision problem.

This process is most easily illustrated by reference to a specific example. The example used here are the Decision Conferences (for further information see [10-12]) carried out within the International Chernobyl Project, in particular as they are directly relevant to the topic of this Symposium, that of countermeasures and remediation following an accident. The attributes that were identified to be of most importance by those participating in the Decision Conferences (essentially those responsible for formulating policy in this area together with their technical advisers) are illustrated in Figure 1 in the form of an attribute hierarchy. The main objective of these Decision Conferences was to evaluate and clarify a number of policy

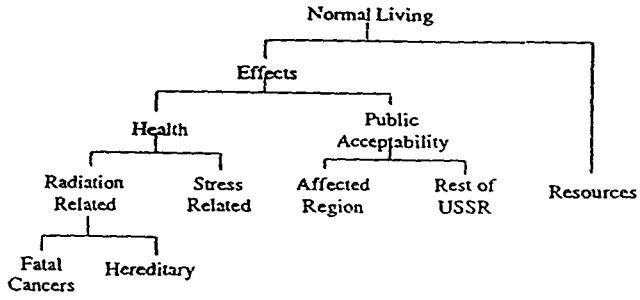


Figure 1: The Attribute Hierarchy

options that could be adopted to restore "normal living" conditions in those areas significantly contaminated as a result of the accident; these options involved taking various remedial measures (eg, relocation of some settlements, agricultural countermeasures, etc), to improve the situation of those affected. The two major attributes identified were the resources required to achieve a policy option (principally but not solely monetary costs) and the effects on the population. The effects on the population were broken down into impacts on health and the public acceptability of the policy which was deemed to be an important consideration. The health impact was further broken down into those resulting from radiation exposure from continuing to live in the contaminated areas and from stress and anxiety which could arise either as a result of remaining in the affected areas or of being relocated. Public acceptability was broken down further into the impact on those directly affected by the policy and on those in the rest of the former Soviet Union who would be indirectly affected through having to meet its costs.

It should be recognised that no simple or universally applicable procedure exists for generating attribute hierarchies - they result from careful analysis and/or discussion of the issues and emerge as much by the art of the analyst as application of any formal procedure. Similarly, there is no right or wrong hierarchy nor is a hierarchy "objective" in the sense that any analyst or group of decision makers would construct precisely the same hierarchy for the same problem. Inevitably, it will be based on the perception or perceptions of the analyst or decision maker/s charged with the particular decision. Notwithstanding this, it is common for different analysts or groups of decision makers facing similar problems to construct very similar hierarchies.

Having established an attribute hierarchy it remains for the analyst or decision maker/s to assign values to each attribute for each of the policy options or alternatives under consideration and to then assign relative weights between the different attributes reflecting the values or preferences of the analyst. Simple linear arithmetic is then involved in determining the "value" of the various options and differentiating between them. Assigning values to attributes is, in principle, simple but invariably it involves much judgement and various techniques are available to help with this. The process is simplest where the attribute is a tangible quantity (eg, monetary cost, etc) but, even here, matters may not be straightforward as the relationship between such a physical quantity and the value placed on it is not necessarily linear. Where the attribute is less tangible (eg, public acceptability) the valuation becomes even more judgemental and must reflect the preferences and beliefs of the analyst or decision maker/s and, inevitably, will be associated with much uncertainty. Assessing the robustness or sensitivity of the conclusions of a MAVA is, therefore, essential. Fortunately, one of the strengths of the approach is that the sensitivity of different judgements or input data can be readily and efficiently evaluated,

thus enabling the analyst or decision maker/s to focus attention on key issues and not be distracted by detail or matters of secondary concern.

Notwithstanding the potential of these techniques, their use is greatly under-exploited both within the radiological protection community and elsewhere. There may be several reasons for this but probably the greatest is a widely held, but misconceived, view that the techniques are complex (indeed even the terminology, MAVA, appears complex) and must, therefore, be very difficult to use other than by a decision analyst. Nothing could be further from the truth. Commercially available software (eg, [13-14] is widely available at a modest cost for carrying out MAVA; these software packages, which are user friendly, require few resources to become proficient in their use and place little or no demands on users in terms of their knowledge or understanding of multi attribute or utility theory. The software facilitates the construction of attribute hierarchies, provides an effective framework for assigning values to attributes for different decision alternatives and relative weights to attributes and, most importantly, is able to display directly the impact of adopting a range of differing judgements or preferences expressed by the analyst or decision maker. What it does **not** do, however, is remove from the analyst or decision maker/s the need to make difficult and complex judgements. Through structuring the problem and breaking it into its component parts it does, however, provide greater clarity and transparency to the decision process and, above all, enables the analyst or decision maker/s to better understand the problem and thus focus on the key issues.

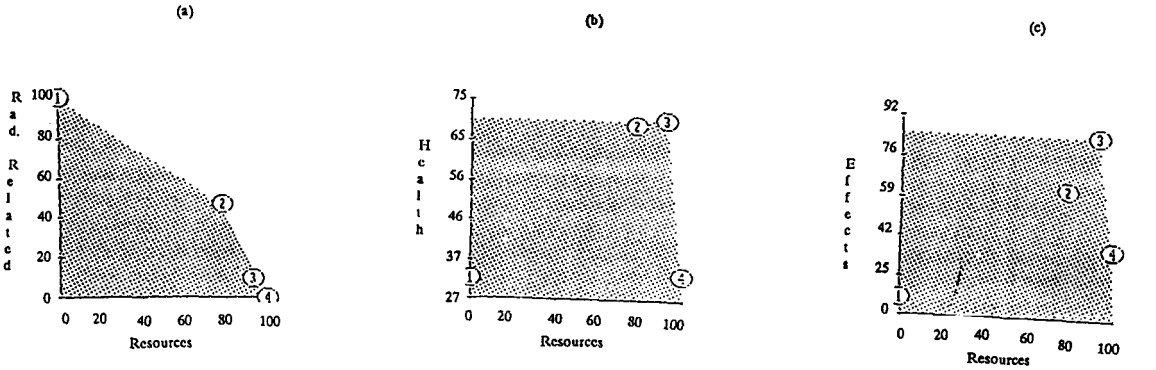
3.2 Illustration of the potential of MAVA

The potential of the techniques is most easily illustrated by an example of an application; selected results from the Decision Conferences referred to above are used here for this purpose. It should be recognised that the emphasis in the following is on demonstrating the potential of MAVA rather than on the results themselves which, within the present context, are largely of academic interest. Consequently, only very limited information is presented here on the Conferences themselves and the quantitative data or judgements that were input into the model in Figure 1 (ie, the values and weights assigned to the various attributes); such information has been kept to an absolute minimum commensurate with providing enough background to comprehend the generality of the results presented. For those with a deeper interest in the Decision Conferences and their detailed content and results, further information can be found in [10-12].

In the last of the five Decision Conferences that were held, the group decided to evaluate four strategies or options for achieving "normal living" conditions which were designated **SL(a-b)** where **a** refers to a lower level of dose below which no restrictions would be imposed, and **b** refers to an upper level of dose above which people would be relocated. In the intermediate region of dose (ie, a to b), various protective measures (eg, agricultural countermeasures, etc) would be taken. The quantities, a and b (expressed in rem), were the additional dose that would be received over a lifetime as a result of remaining in the contaminated areas assuming no protective measures were taken. Quantitative input to the model was provided on the expected reduction in radiation induced health effects and costs consequent upon different options. Values assigned to the other attributes and to their relative weights represented judgements of the group. Extensive sensitivity analyses were carried out with the model to evaluate the sensitivity of the results obtained to uncertainties in the judgements expressed. The following selection of illustrative results shows that the conclusions reached were remarkably robust to changes in the values and weights within the ranges judged reasonable by the group.

Figure 2 shows a sequence of plots of "effects" against the costs of the strategies or options. In Figure 2(a) the **Radiation Related** effects are plotted against **Resources**, ie, the costs of the strategies. Note first that the **Resources** scale is a preference scale (as are all the scales in the model). This means that greater scores correspond to increasing preference and, in this case, decreasing cost. Moreover, conventionally all scales are re-normalised to have a range 0-100. Thus on the **Resources** scale, here the horizonal axis, **SL2-40** has a score of 100 since it is the cheapest, whereas **SL2-2** has a score of 0 since it is the most expensive. The vertical axis gives the scores for the strategies on the **Radiation Related** effects, again re-normalised to a 0-100 scale with the most preferred strategy **SL2-2** under this attribute scoring 100.

Figure 2(b) gives a similar plot in which the vertical axis now represents the overall **Health Effects**, given by combining **Radiation Related** and **Stress** scales. Figure 2(c) again differs in the vertical scale. In this case the attribute is **Effects** which arises from combining the **Radiation Related, Stress** and **Public Acceptability** scales.

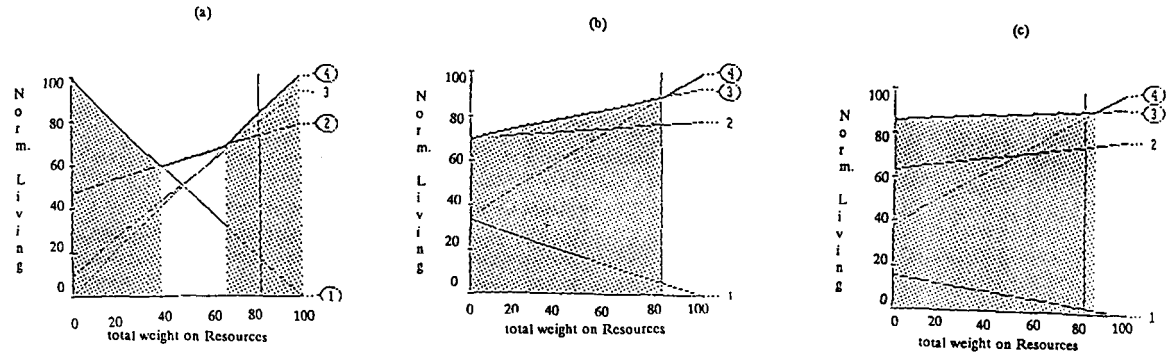


(a) Plot of Radiation Related effects against Resources.

(b) Plot of Health effects against Resources.

(c) Plot of Effects against Resources.

Figure 2: Plots of effects against resources

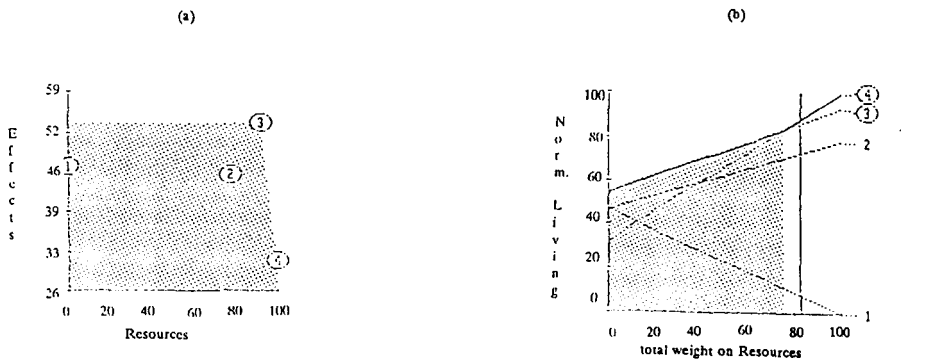


(a) Sensitivity plot when neither Stress nor Public Acceptability included.

(b) Sensitivity plot when Public Acceptability not included

(c) Sensitivity plot when all Effects attributes included.

Figure 3: Sensitivity plots when different attributes included in model



(a) Plot of Effects against Resources.

(b) Sensitivity plot when all Effects attributes included.

Figure 4: Plots for the model when SL2-2 is most acceptable to those living in the Affected Region

Key 1: - SL2-2 2: - SL2-10 3: - SL2-20 4: - SL2-40

In each of Figures 2(a-c) "ideal" strategies would lie at the top right hand corner, because such strategies would score 100 on both the horizontal and vertical preference scales. When **Stress** and **Public Acceptability** are included, as in Figure 2(c), it may be seen that **SL2-20** (Strategy 3) moves towards the top right hand corner.

Figure 3(a-c) shows the results of a number of sensitivity analyses which illustrate the influence of varying the weights assigned to different attributes. The vertical line in each plot is set to a weight on **Resources** which corresponds approximately to a cost of 100 Roubles (1990 prices) per man rem saved.
Figure 3(a) gives the sensitivity plot when zero weight is applied to both the **Stress** and **Public Acceptability** attributes. With the weight on **Resources** equivalent to assigning a value of 100 Rouble for saving 1 man rem, **SL2-40** is optimal. As the weight on **Resources** is decreased (ie, as the number of Roubles per man rem is increased), **SL2-10** then **SL2-2** become optimal. In a very rough way, Figure 3(a) corresponds to a simple cost benefit analysis of the strategies in which trade offs are made between the radiological protection benefits and the resources required.

Figure 3(b) shows the same sensitivity plot when **Stress** is given its assessed weight, but **Public Acceptability** is still given zero weight. **SL2-20** is now optimal, but only just so. If the weight on **Resources** is increased very slightly, then **SL2-40** becomes optimal. This can be seen by the fact that the vertical line representing the weight on **Resources** given by the equivalence of 1 man rem with 100 Roubles is right on the boundary between the range of values for which **SL2-20** is optimal and the range for which **SL2-40** is optimal. Note that with the weight on **Stress** included, **SL2-2** and **SL2-10** are not optimal for any value of the weight on **Resources**.

Figure 3(c) shows the same sensitivity plot when **Public Acceptability** is given its assessed weight. **SL2-20** remains optimal over a wider range of weights on the **Resources** attribute. Furthermore, with the weight on **Resources** set at a value equivalent to 100 Rouble per man rem saved, **SL2-20** is optimal and remains optimal if the weight is slightly increased.

From Figures 2 and 3, it is clear that the introduction of the attributes **Stress** and **Public Acceptability** into the model suggests that **SL2-20** is the preferred strategy in place of **SL2-40**, which is the strategy that a simple trade-off between costs and radiation effects would suggest.

Much the same conclusion may be drawn if the scores on the public acceptability in the **Affected Region** are changed to make **SL2-2** the most preferred strategy. Figure 4 gives plots corresponding to Figure 2(c) and 3(c) for this model. Now it is not the case that **SL2-20** is the preferred strategy when **Stress** and **Public Acceptability** are included in the evaluation; the model suggests that **SL2-40** is. However, this observation is relatively less important than the observation that **SL2-2** is not suggested as the most preferred strategy even when it receives a score of 100 on the **Public Acceptability** scale and **SL2-20** and **SL2-40** are both scored at 0, as they are in the model on which Figure 4 is based. In other words, with the weight currently assigned to **Public Acceptability**, neither **SL2-2** nor **SL2-10** can be made optimal simply by giving them a high **Public Acceptability** score. Sensitivity analysis of the weight on **Public Acceptability** confirmed that this observation holds for all values of this weight which the group felt reasonable.

The main conclusion drawn from the above analysis was that, if only **Radiation Related** and **Resources** attributes were included in the model, then **SL2-40** would be the strategy that provides the best balance. However, once **Stress** and **Public Acceptability** attributes are included, it became quite clear that **SL2-20** was the strategy that provided the best balance. This is shown particularly clearly by the sequence of plots shown in Figures 2 and 3. These conclusions must of course be qualified in the sense that they are particular to the values and weights assigned by the group to the respective attributes and to what they judged were reasonable variations in them.

The above results illustrate the potential of MAVA for effectively structuring and evaluating complex decision problems. A particular strength of MAVA is the ease with which it enables the sensitivity or robustness of conclusions reached to be evaluated. It is equally amenable for use by an individual analyst or a group but is particularly effective in aiding group decision making, especially where the group comprises differing levels of skill and experience and possibly divergent opinions. Further information on the use of MAVA in structured Decision Conferences can be found in [15-18]. The shared understanding and clarification of the issues which MAVA can provide, together with the ease with which the implications of different viewpoints within a group can be evaluated, greatly facilitate the decision process. The process itself encourages shared ownership of a

problem and, as a result, can lead to greater commitment and accountability from the various stakeholders in any decision reached.

4. Summary

For longer term countermeasures only outline guidance is currently available at an international level. Arguments can be made both in favour of and against developing more detailed guidance in advance of an accident and judgements on this issue are finely balanced. Notwithstanding what judgement is made on this issue, as a minimum, there is a need to have in place in advance of an accident adequate procedures to transform outline guidance into a practical and defensible policy which takes account of all influential factors. Techniques that could be used to aid this process have been reviewed, notably multi-attribute value analysis which is particularly suited for such applications.

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International Congress on Radiation Protection-IRPA9, Vienna, Austria, 14-19 April 1996

Psychological and Social Impacts of Post-Accident Situations : Lessons from the Chernobyl Accident

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Abstract. *This paper presents the main features, from the psychological and social points of view, of the post-accident situation in the contaminated areas around Chernobyl. This is based on a series of surveys performed in the concerned territories of the CIS republics. The high level of stress affecting a large segment of the population is related to the perception of the situation by those living in a durably contaminated environment but also to the side-effects of some of the countermeasures adopted to mitigate the radiological consequences or to compensate the affected population. The distinction between the accident and the post-accident phase is enlarged to take into account the various phases characterizing the dynamics of the social response. Although the size of the catastrophe as well as the economic and political conditions that were prevailing at the time and after the accident have resulted in a maximal intensity of the reactions of the population, many lessons can be drawn for the management of potential post-accident situations.*

INTRODUCTION

The knowledge of the psychological and social impacts of severe accidents involving radioactivity as well as the comprehension of the mechanisms driving the acceptability of post-accident situations characterized by large contaminated areas were rather limited before the Chernobyl accident. The first attempts in the late eighties to explain the reaction of the population in the affected territories after the accident were superficial. They were founded on the observation that the general public was generally ignorant about the potential effect of radiation, rather suspicious with regard to any official information, largely influenced by media having a strong propensity to spread alarmist news or perturbed by the on-going controversy between experts on the nature and the scope of the measures to be adopted to improve the situation.

Although these various aspects have certainly played a role, they were not sufficient to explain how a high level of stress was dominant in a large fraction of the population living in the contaminated territories in the three affected Republics and leading to a profound social crisis many years after the accident. This paper is first presenting the psychological situation as well as some of the factors that have been identified at the origin of the chronic and acute stress which is still characterizing the situation and preventing a return to normal living conditions from the psychological and social points of view. It is based on the results of a series of studies carried out from 1990 to 1995 in the contaminated areas in Belarus, Russia and Ukraine, first in the framework of the International Chernobyl Project initiated by the International Atomic Energy Agency [1, 2] then within the Evaluation Programme of the Consequences of the Chernobyl Nuclear Accident (Joint Study Project n°2) from the European Commission [3, 4, 5]. The second part of the paper describes the successive steps reflecting the social response during the emergency and the post-accident phases following the accident. This dynamics allows to better take into account the symbolization process driving the social acceptability of the post-accident situation. Finally some considerations for improving post-accident management and restoring social trust in the affected population in case of future accidents are presented in conclusion.

THE PSYCHOLOGICAL SITUATION

Ten years after the Chernobyl accident, any well informed observer can notice that a large anxiety is still existing among the population living in the contaminated territories around Chernobyl. Quantitative studies on the psychological and social effects carried out in 1992 and 1993 [3, 4] highlight an acute feeling of lack of control over individual living conditions clearly linked to a high level of psychological distress in the affected population. This anxiety is essentially related to the potential effect of radioactivity on health and particularly the one of children. Without being able to verify the accuracy of their allegations, interviewed persons are mentioning an abnormal number of somatic effects which are all attributed to the ambient contamination. This last one is a permanent concern for the population who seems in the impossibility to forget the accident and its real or supposed consequences.

As early as 1987, a few Soviet experts have put forward the rather confusing notion of radiophobia to try to explain the psychological reaction of the population. This notion was a convenient explanation for the fears noticed by many individuals inside the population, fears being considered groundless by the experts taking into account the prevailing radiological conditions. The investigations carried out in the contaminated areas have demonstrated that questioned people were expressing fears always supported by rational speeches on the basis of their observations related to the situation [5]. It is difficult to conceive how the concept of phobia, which in psychiatric terms means an unfounded and irrational fear concerning objects and situations that are not in themselves hazardous, could apply to a situation where on the contrary, there were objective reasons to be anxious. It is also interesting to note that this hypothesis of radiophobia, which is now definitively abandoned, received initially a large echo at the international level among experts dealing with the consequences of the Chernobyl accident to explain the attitude of the population in the contaminated areas.

Over the recent years, the concept of chronic and acute stress has progressively emerged to explain some of the symptoms to be observed in the affected population. This stress, which meaning is different from the one given in media and everyday language, can be described as a «continuing inability to adapt at the biological, psychological and social levels. The body, the nervous system and the psychology of those affected enter a state of alert or excitation involving a permanent expenditure of energy. The result is bodily dysfunction and pathological effects manifesting themselves both organically and psychologically (extreme fatigue, insomnia and depression)... A characteristic of the stress situation is that habitual, regular and familiar responses are dislocated, and that the subject is unable to find satisfactory solutions to the problems with which he or she is confronted» [6]. In the post-accident context of Chernobyl, this stress is observable in all groups affected i.e. the re-settled population, the one living in the contaminated areas and of course the liquidators. The anxiety mainly focuses on health, and particularly on the health of children, with a constant reference to somatic disorders. There is also a great concern of the population about its future with the conviction by many people that the situation is not going to improve with time but on the contrary can only get worse thus reinforcing their fatalist and passive attitude.

Many stress factors have been identified which are closely linked with the management of the accident and post-accident phases of the Chernobyl accident. To better understand some of these factors it is useful to distinguish between the Soviet period of the accident management from 1986 to 1990 and the national period with the takeover, from 1991, of the post-accident management directly by Belarus, Russia and Ukraine.

The first period is characterized by the relative failure of the protective measures adopted. Evacuations and iodine distributions were often organized too late and the population affected by the radioactive releases received in some circumstances very high doses. This is in part explaining the growing number of health effects over years that have been observed reinforcing the fear of the population. The context of secrecy and censure which prevailed during this period also largely contributed to the development of distrust towards the authorities. Globally the accompanying symbolization process which normally follows an

accident did not take place and the population has attributed all the negative consequences of the accident to the remaining contamination. The occultation of the importance of the accident and of its potential delayed effects by the authorities associated with the fact that the population was permanently confronted to lasting countermeasures led to over-amplify the risk of the post-accident phase. Furthermore, the presence of about 800,000 liquidators disseminated among the general population was a supplementary factor of disturbance. Their greater vulnerability from the health point of view reinforced the distorted perception of the situation as regard to the potential effects of the residual contamination and contributed to maintain or even increase the level of stress in the general population.

The second period has been fundamentally characterized by the willingness of the national authorities to differentiate their management from the previous centralized one by adopting a more realistic and democratic perspective. However, the emphasize put on the defense of the victims of Chernobyl has mainly focused the attention of the new authorities on the post-accident situation. The various national laws adopted by the Republics in the early nineties have established rather generous compensation systems which were supposed to be financed by the Russian Federation. Besides the fact these systems have never been properly financed, they have established a close link between the level of contamination of the environment and the level of compensation thus reinforcing the idea that the risk is first associated with living in contaminated territories. As a consequence, the population many years after the accident tends to attribute all the problems it is facing to the residual contamination and the general feeling is that difficulties will even grow with time. It is evident that such a perception of the situation is a powerful factor for reinforcing the general stress of the population as well as a mental trap to which it is difficult to escape. One can also understand the devastating impact of a lack of objective information on the potential consequences of a large nuclear accident in the early phase and even before.

Another important stress factor is the loss of trust of the population in the scientific, medical and political authorities. Beyond the negative impact of the lack of transparency of the authorities at the time of the accident, the on-going debates on the criteria for establishing countermeasures as well as on the potential health effects associated with the remaining contamination have slowly turned the situation into a very complex one where individuals did not trust anymore experts and felt totally insecure and unable to contribute to resolve the problems themselves. A general feeling of loss of control is thus reinforcing the climate of social distrust.

Observations in the contaminated areas have also shown the paradoxical effects, from a psychological point of view, of many of the countermeasures adopted. The zoning of the contaminated areas has induced a ghetto effect leading to a loss of identity of the population and some forms of exclusion. The definitive resettlement was a strong factor of stress because in most cases the area of relocation was imposed without taking into account the social and cultural affinities of the relocated persons. Moreover, the conditions of relocation into blocks of flats in urban areas was also creating a ghetto effect installing the relocated population into a social status of «Chernobylites». Finally, the compensation system also contributes to develop jealousy in the non compensated population and a reinforcement of the segregation with those affected who are feeling more and more isolated.

It is worthwhile, to complete this rapid overview of the psychological situation, to mention the situation of the few hundreds of persons called the «samossiols» who were initially evacuated at the time of the accident and despite administrative interdictions, came back in the thirty kilometer zone to stay again in their own houses. These generally old persons who are living in an hostile environment and in poor material conditions seem to be in a better psychological situation than the population living in the other contaminated areas. Their willingness to live as before without the financial support of the state can be interpreted as a positive factor to restore their autonomy and thus could explain a lower level of distress than the one observed in the territories affected by the accident.

THE SOCIAL DYNAMICS OF ACCIDENTS

From the accident management point of view, it is usual to distinguish two main phases: the emergency one which is following immediately after the accident and during which a series of actions are implemented according to plans prepared in advance, and the post-accident phase during which countermeasures are adopted along the justification and the optimization principles taking into account the specificity of the situation resulting from the accident. This simple scheme does not reflect properly the dynamics of the accident from the point of view of the reaction of the population. In the perspective of better taking into account the psychological and social dimensions, it is worthwhile to envisage a more detailed sequence characterized by the collective behavior of the population directly affected by the accident as well as the rest of the population which can be observed each time a large accident or a catastrophe occurs [7].

The sequence generally starts with the implementation of emergency measures planned in advance. The level of preparation including the information of the public is an essential element for the success of this phase which perception is largely influencing the next ones. This reflex phase is followed by a strong reaction expressed by both the affected population and the general public. Besides an inevitable feeling of revolt which normally turns rapidly into a social debate about negligence and responsibilities about the causes of the accident, heroic actions to save persons or to eliminate remaining acute dangers reinforce a strong feeling of solidarity by identification to the victims especially among the non directly affected population. Past experiences with large industrial accident or natural disasters show that during this rather short period, the affected people as well as the society as a whole are able to endure exceptional constraints and particularly to face much higher levels of risk than in normal situations.

The heroism phase is normally followed by a transition period during which a set of technical and administrative measures are implemented with the objectives to control the residual risk for the population, to repair if necessary the physical and health damages or to restore the general environment and to reorganize the social activities. This phase can be compared to a convalescence during which society is recovering progressively and return to a situation considered as normal. Parallel to this evolution, the symbolization process around victims is engaged to allow the construction of a collective memory indispensable to reaching social acceptance of the accident and its consequences. This last dimension is very similar to a mourning process involving both the affected population and the society as a whole. It is important to note that the compensation of victims for the damages they supported is part of this general process of symbolization and it is therefore important to properly assess the various losses of each affected individual. In this perspective, the reconstitution of doses received by each individual at the time of a nuclear accident is a key element to objectify the risk and to set up an effective compensation system.

The return to normal living conditions after a large accident affecting a society is a complex process involving a rearrangement of the mental representations of most of the affected people and in some severe cases the establishment of a new social regulation to overcome the discredit of the old system which failed. The success of this process is not only depending on technical measures but heavily rely on the ability of authorities to restore as soon as possible social trust in the society and particularly some confidence with regard to experts and authorities. The return to normality expression must be well understood and used properly. By no means it is a return to the situation that was prevailing before the accident. This is impossible in most cases and particularly after a large nuclear accident followed by a durable contamination of the environment. The objective is to restore a situation where those involved can again focus their attention to their individual day-to-day activities and plans in a context of vigilance towards the ambient contamination. This does not mean that society behaves like nothing has happened, but on the contrary that the consequences of the accident have been fully integrated at both the rationale and symbolic levels, are taken into account in the day-to-day behavior of the population and finally serve for the whole society as a learning experience for the future.

The above described evolution is ideal. In reality some of the key factors for restoring the situation cannot play their curative role and it is even possible that, as it was the case with the Chernobyl accident, a post-accident crisis is taking place due to the rejection of the situation by the affected population. The mechanisms leading after the solidarity phase to a non-acceptance of the situation and then a revolt are not completely identified. At Chernobyl, the high level of stress in the population has certainly played an important role but there was also a conjunction of other factors going far beyond the psychological dimensions. It is evident that the social, economic and political context was extremely favorable as well as the very poor level of information and preparation of the population with regard to a potential nuclear accident. Interviews in the population around Chernobyl and in the contaminated areas have shown that the accident was simply not possible in the mind of people.

It is evident that the intensity of the accident and the extent of its consequences in terms of the number of persons having received high individual doses, the surface of the severely contaminated territories, the number of people to be relocated and the economic cost to try to restore a tolerable situation have contributed to rapidly build a collective representation closer to a disaster and a catastrophe than a large accident. It is not by chance that the population in the contaminated areas is speaking about the accident and the post-accident phase with a lot of words coming from the war vocabulary.

PERSPECTIVES

Although still limited, the analysis of the post-accident situation at Chernobyl allows to delineate some elements first for designing better communication with the public in case of an accident and, secondly, for establishing criteria and strategies for setting up protection objectives and selecting countermeasures which should be compatible with the progressive restoration of social trust among the affected population.

The Chernobyl experience has demonstrated that beyond a certain level of contamination of the environment, the situation is not manageable and it is necessary to relocate the population and to abandon the concerned territories. This is of course true when the level of exposure of the population may lead to the appearance of deterministic effects but also when it is impossible to maintain the integrity of the basic social functions. At the opposite of these unacceptable situations, it is possible to define a level of residual contamination below which the risk can be considered as negligible. This situation must be characterized by the total absence of restrictions on the day-to-day life to allow the population to slowly forget the accident and its consequences after the process of mourning and symbolization leading to a certain form of return to normality.

Between these two situations (unacceptable and negligible), the population is facing a large spectrum of situations that can be qualified as tolerable. They are characterized by the existence of more or less severe restrictions on normal living conditions affecting the production, the distribution and the consumption of goods as well as the social organization related to education, the health care system or the collective leisure activities. Some of these constraints (countermeasures) which are implemented to last on rather long periods need to be adjusted with time and must be accompanied with an effective and independent structure of surveillance and control of the contamination to insure the transparency of the measurements of the residual contamination. This is a very important element to objectify the level of risk as well as to decide about the actions to be adopted. Transparency must also prevail as far as the effectiveness and the cost of countermeasures are concerned.

The main difficulty with the management of post-accident situations concerns the establishment of criteria expressed as numerical values that could serve as reference to define the negligible, the tolerable and unacceptable levels. Various considerations have to be taken into account to elaborate these criteria. The Chernobyl experience is clearly demonstrating that it is impossible to set up reference values for structuring decision making processes related to the post-accident management without referring to those which are universally adopted for the protection of the public and workers in normal situations. Furthermore, the control of exposure

in the contaminated areas necessarily involves to take into account the diversity of sources and in this perspective it is extremely difficult to envisage centralized and generic measures that could be applied to large areas. The implementation of a decentralized approach relying on the commitment of those directly affected by the accident seems to be the most effective way of attacking the problem.

The Chernobyl experience has also revealed the complexity of managing post-accident situations with the need to reconcile the radiological protection imperative with the various psychological and social dimensions. Taking only into account the radiological considerations may lead to a difficult situation because of the perverse effects of many countermeasures which are reinforcing the feeling of exclusion of the affected population and maintaining the doubt on the risk really at stake. At the opposite, giving priority to the psychological factors may involve serious consequences from the radiation protection point of view if the population is not fully aware about the exposure pathways and the mechanisms through which it is possible to control exposures. Observations have been made over the recent years confirming that the doses received by some part of the population living in the contaminated areas were increasing because of a lack of precaution or even in some cases of a denial of the potential impact of the contamination.

The main difficulty for restoring normal living conditions in contaminated territories is to build collective modalities to cope with the radiological residual risk in order to lead progressively to an acceptable situation from the psychological and social points of view. The success of this approach largely depends on the capability to set up an individual and collective management of residual risk, in spite of the implementation of long lasting countermeasures. Anticipating a benefit, the involved persons must be in a position to actively and voluntarily manage the level of their exposure taking into account the specific situation they are facing. From a collective point of view, adequate means to control the risk must be available: concepts, assessment tools, equipment, budget... Such an approach is at the opposite of the one which inspired the various systems implemented so far in the affected Republics where the persons exposed to the residual contamination are compensated for the risk they take and not for the one they contribute to avoid for themselves and the others. Furthermore, these compensation schemes reinforce the existing risk by objectifying it in financial terms and perquisites.

In practice, the adoption of a pro-active approach involving the commitment of the concerned population should result in the implementation of an ALARA type process close to the day-to-day activities of the population (at schools, in the plants, within families, at the farm level...). This decentralized approach should be implemented in the framework of a more global system ensuring the coherence and the rationale use of the material and financial support indispensable to restore satisfactory living conditions. An independent system for controlling the situation, particularly the contamination of food products and the close environment of the population, should also allow to facilitate the restoration of a climate of social trust within the population with regard to the experts and the authorities. Such mechanisms must rely on solid ethical basis clearly displayed in view to establish the transparency of the decision making processes and the responsibilization of all concerned actors and primarily the affected population. To respond to the question: Is it possible to live here? it is inevitable to begin with a true characterization of the situation in term of risk which is the first element to develop a commitment to the situation with a protection culture based on the responsibility of each individual to the risk he or she is going to take freely and voluntarily.

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RADIATION SAFETY PROBLEMS AROUND HEAVY ION ACCELERATORS

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INTRODUCTION

Heavy ion accelerators are finding increased application in both the fundamental and applied sciences around the world. While the use of these accelerators in the applied sciences and industry has perhaps been slower, and is less extensive, than has been the case for proton accelerators, it is now rapidly increasing. Ion species and their energies used in these facilities spread in wide ranges. Heavy ion beam intensities can vary over a wide range from hundreds of microamperes downwards. Some of these beams have the potential to produce radiation, principally neutrons, which may expose prsonnel or be detrimental to sensitive detectors or equipment. The paper gives a brief overview of the radiological safety aspects of heavy ion accelerators. Information available on radiation environments of the accelerators, angular distribution and neutron spectra around thick targets bombarded by ion beams, induced radioactivity in irradiated targets and shielding problems are considered. Operational health physics of a superconducting heavy ion collider and associated design criteria for prompt radiation are also discussed.

THICK TARGETS PARTICLE YIELDS

It is sometimes necessary for personnel to work in areas where heavy ions interacting with targets, target holders or beam transport equipment produce prompt radiation levels significantly above background. Since neutrons are responsible for a major part of this background and, it would be helpful to be able to predict neutron radiation levels as a function of ion mass and energy. For a shielding design of heavy ion accelerators, inclusive neutron yield data are fundamental to estimate source terms of the calculation. For a detailed calculation of accelerator shielding more information than the total number of neutrons produced by heavy ions on thick targets is required: in particular the energy and angular distribution of neutrons must be known.

Experimental information on neutron production by heavy ions on thick targets of various materials over the energy range from 0.9 to 3650 MeV/A is given in Table 1, together with references to the original sources. The data presented in Table 1 were obtained for the most part with targets thick enough to completely stop the incident ions.

TABLE 1. Information on neutron production by heavy ions on thick targets.

Ions	Energy, Mev/A	Targets	Yield	Angular distrib.	Energy distrib.	Ref.
^{6}Li, ^{7}Li, ^{9}Be, ^{10}B, ^{11}B, ^{12}C,^{13}C,^{16}O, ^{18}O, ^{24}Mg, ^{27}Al, ^{28}Si, ^{40}Ca, ^{58}Ni	0.9 - 5.84	Li, Be, C, Mg, Al,Ca, Ti, Ni, Cu, Mo, Au	Dose rate	Yes	No	[1,2]
Ne, Si	670•A	Cu	Yes	Yes	No	[3]
^{12}C,^{14}N, ^{16}O, ^{20}Ne	3 -16	Cu, Fe, Ni	Yes[a)]	No	No	[4]
^{58}Ni	6.6	Cu	Yes	Yes	Yes	[5]
^{48}Ti	4.2	Cu	Yes	Yes	Yes	[6]
^{12}C	86	C, Al, Fe	Yes	Yes	No	[7]
^{6}Li, ^{7}Li	6	Li, Be, C, Cu	Yes	Yes	Yes	[8]
^{12}C,	3650	Cu	No	Yes	Yes	[9]
^{12}C,	3650	Cu	Yes	No	No	[10]

[a)] Dose equivalent rate at 1 m, 90° from beam direction

Measurements of the neutron yield were made for variety of ions with thin or thick targets. However, the measurements are still not enough in the number to estimate the systematics in the neutron yields.

A theoretical model for estimating the potential radiation hazards associated with low energy heavy ion accelerators was proposed by B.B.Back [11]. The model is based on procedures for estimating the fusion cross section, fission competition, neutron evaporation and stopping powers. The range of applicability of the model is limited to energies less than about 15 MeV/nucleon.

A number of empirical formulae to estimate yield and angular distribution of neutrons produced by heavy ions on thick targets of various materials were proposed.

Based on the existing in the beginning of eightieth data the formula for the neutron dose equivalent rate caused by an intermediate energy heavy ion beam impinging on a thick target was proposed by F.Clapier and C.S.Zaidins [12]:

$$R(\theta,r,i,Z,w) = Y(Z,w)\, i\, f(\theta,\gamma) / 7.2\, r^2 \quad \text{mrem/h}, \qquad (1)$$

where $f(\theta,\gamma) = \frac{1}{4\pi} \frac{1}{\ln(1+1/\gamma)} \frac{1}{\gamma + \sin^2(\theta/2)}$, $Y(Z,w) = C(Z) w^{\beta(Z)}$,

i is heavy ion current in particles/s, r is distance from the target in meters, w is beam energy in MeV/A, Z is atomic number of the projectile and the empirical parameters C, β and γ can be interpolated or approximately calculated from table 2 of ref. [12]. This formula allows to predict the rate within a factor 2 for values of w up to 10 - 20 MeV/A.

The thick target neutron dose for a range of heavy ion reactions covering the projectile mass range 6<A<58 and the target mass range 6<A<197 have been measured at five different angles and over the energy range $0.9 \leq E/A \leq 6.0$ MeV/A and represented by simple empirical formulae[1]:

$$D = D_0\, e^{mP} \text{ when } P < 0 \text{ and } D = \alpha^2 (P+\beta)^2 \text{ when } P > 0, \qquad (2)$$

where D is the mean dose rate at a distance of 1 m from the target, $P = (E_b - CB)/A_b$, where E_b is the beam energy, A_b is the mass of ion in beam, CB the Coulomb barrier for a specific beam-target combination and α, β, D_0 and m are fitting parameters tabulated in table 1 of ref. [1].

The dose rate $D(\theta_L)$ at angle θ_L to the direction of the incoming particle is defined as follows:

$$D(\theta_L) = DK_m\, e^{S\theta_L}, \qquad (3)$$

the quantities $D(\theta_L)$, D, S and K_m for a specific beam-target combination are given in ref. [2].

Differential thick target neutron yields by several heavy ions with energies of about 10 MeV/amu for several thick targets were measured and fitted by two component moving source model and simple expressions which represented the systematic dependence of the yields on the target, incident ions and their energies were proposed [13]:

$$\phi(E_n,\theta) = \sum_{i=1}^{2} M_i \frac{E_n}{2(\pi\tau_i)^{3/2}} \exp(-\frac{E_s}{\tau_i}), \qquad (4)$$

$E_s = E_n - 2\varepsilon_i E_n \cos\theta + \varepsilon_i$, where M_i is the source intensity, τ_i is the nuclear temperature, ε_i is the source moving energy in terms of the kinetic energy of nucleon, E_n is the neutron energy and θ is the neutron emission angle in the laboratory frame. To extend these formulations to a much higher energy range inclusive neutron production cross sections were systematically calculated [14] for heavy ions of the energy of 337 MeV/amu by the intranuclear cascade evaporation code HIC [15]. Using the calculation results, the above expressions were modified [14] so that the systematics of the calculated neutron production cross sections were described by the formulae.

L.G.Beskrovnaja and M.M.Komochkov [16,17] have summarized neutron yields for ^{12}C and ^{238}U reactions in thick iron target in the energy range 2.5 MeV/nucleon to 3.65 GeV/ nucleon. The yield, $Y(E_i)$, for ^{12}C in the energy range 2.5 MeV/nucleon to 6 MeV/nucleon was calculated based on formula (3) and in the energy range 4 MeV/nucleon to 100 MeV/nucleon the yield was taken form ref.[12]. At energies more than 60 MeV/nucleon the yield, $Y(E_i)$, was calculated by formula:

$$Y(E_i) = N(\sigma_n \eta_n / \sigma_p \eta_p)\, f\, Y_p(E_i), \qquad (5)$$

where N is number of equivalent protons [18]: $N = (A_i)^{0.25} + (A_i - 1)^{0.6}\, 0.078 [\ln A_i - 1.85]$, E_i is energy of the projectile in MeV/nucleon, σ_n and σ_p are inelastic interaction cross section of nuclear-nuclear and proton-nuclear, respectively; η_n and η_p are number of inelastic interactions of nuclei and protons in the target, respectively; $Y_p(E_i)$ is yield of neutrons from the same target bombarded by protons with the energy E_i, $f = 1 + 0.19\eta_n \sqrt{A_n}$. The yield, $Y(E_i)$, for ^{238}U in the energy range 7 MeV/nucleon to 20 MeV/nucleon was calculated based on calculation of dose equivalent taken from ref. [11]. At energies more than 60 MeV/nucleon the yield, $Y(E_i)$, was calculated by formula (5).

A series of radiation measurements were carried out with high energy ^{12}C ion beams in CERN [7] (86 MeV/nucleon) and JINR [9,10,18,19] (3.65 GeV/nucleon) to determine secondary - particles production on thick targets. Three different targets (Fe, Al and C) in CERN and two targets (Cu, Pb) in JINR were used. Various threshold detectors were used to obtain information about angular neutron energy distribution and hadron yields[7,10]. Scintillation counter telescope was used to measure angular charged particles distribution from Cu [19] and Pb [20] targets. Neutron energy distribution at 71° and 105° was measured by time-of-flight spectrometer [11]. The calculated [15] angular flux density distribution around the Fe target is compared [7] with the measured flux densities. The difference between the different target results is small. The the theoretical data never differ by more than factor 2 from the experimental results. Experimental results [11] together with calculated [20] neutron spectra are shown in Fig. 1.

RADIATION SHIELDING

The estimation of shielding for accelerators requires a detailed understanding of the production of particles by the interaction of the primary ions, their transport through the shield and determination of the energy spectrum of the radiation that penetrate the shield. The vast literature on the shielding of proton accelerators was reviewed by R.H.Thomas and G.R.Stevenson [23]. In principle, the radiation shielding problems at proton and at heavy ion accelerators are equal ones with the exception of source terms. Consideration of the source term was given in previous section. In this section experimental and calculated data on neutron attenuation in shield, neutron dose equivalent and spectra outside heavy ion accelerators shield is considered.

Experimental attenuation length in heavy concrete (density 3.4 g/cm^2) have been obtained by J.W.N.Tuyn et al. [7] with various threshold detectors under 0° and 45° with the incident 86 MeV/nucleon ^{12}C ion beam on Fe, Al and C thick targets. The attenuation length under 0° derived from the S detectors is 78±9 g/cm^2 while derived from ^{11}C detectors an attenuation length of 104±2 g/cm^2 is found as an average for all three targets. Under 45° an attenuation length of 80±4 g/cm^2 is found for the S detectors and 91±12 g/cm^2 for the ^{11}C detectors. These values shall be considered as upper limits due to the unknown contribution of scattered neutrons.

Attenuation of neutrons through a concrete shield was measured by J.B.McCaslin et al. [3] at 7°, 54° and 72° to the beams direction. Beams of neon and silicon at 670 MeV•A were stopped in copper target. The attenuation curves are shown in Fig. 2. Values of the effective attenuation length observed at 7°, λ_{conc}= 114 and 109 g cm^{-2} for Ne and Si, respectively, are consistent with those at high energy proton accelerators and represent the maximum value. For the range of concrete shielding thickness used, the effective attenuation length decreases with increasing angle. Values obtained are 64 and 56 g cm^{-2}, for secondaries from Ne at 54° and 72°, respectively, and 61 g cm^{-2} for Si at 72°. For a beam of Ne ions on Cu, $\lambda_{conc}(\theta)$ may be represented by the equations [3]:

$$\text{for } \theta<75°: \qquad \lambda_{conc}(\theta) = 115\, e^{-0.01\theta} \text{ g cm}^{-2} \qquad (6a)$$

$$\text{for } \theta>75°: \qquad \lambda_{conc}(\theta) = 55 \text{ g cm}^{-2}, \qquad (6b)$$

where the angle θ is in degrees. The second of these expressions (6b) is a conservative assumption and is used because of inadequate data at larger angles.

On the basis of the measured attenuation lengths and angular distributions [3], an empirical model was devised to fit measured dose equivalents, H(θ), outside thin shield [3] :

$$H(\theta)=8.67\times10^{-12} N C_1 \theta^{-1} r^{-2} e^{-l(\theta)/\lambda(\theta)} \qquad (7)$$

where H is in Sv and θ is in degrees, N is the number of Ne ions stopped in Cu target, C_1 accounts for that fraction of the dose equivalent due to neutrons of energy greater than 20 MeV, r(θ) is the distance from the target to the detector (m), l(θ) is the shield thickness in the direction θ (g cm^{-2}), λ(θ) is the attenuation length (g cm^{-2}). For the estimation of dose equivalent outside thick shields where particle equilibrium is achieved, following expressions were proposed [3]:

$$\text{for } 0°<\theta<20°: \qquad H(\theta) = 12\times10^{-12} N e^{-0.20\theta} r^{-2} e^{-l(\theta)/115} \qquad (8a)$$

$$\text{for } 20°<\theta<120°: \qquad H(\theta) = 12\times10^{-12} N e^{-0.038\theta} r^{-2} e^{-l(\theta)/115} \qquad (8b)$$

where H is in Sv, r is in meters and l is in g cm^{-2}

L.G.Beskrovnaja and M.M.Komochkov [16] proposed a phenomenological model for calculation of differential yields of neutrons, $d^2 Y/dEd\Omega$, produced by heavy ions on thick targets and for estimation of heavy

ion accelerators shielding. The model covers the projectile mass range $4 \leq A \leq 238$ and energy range from Coulomb barrier to 10 GeV/nucleon. Comparison of attenuation curve calculated for heavy concrete [16] with measurements [7] shows reasonable agreement.

T.A.Gabriel et al.[24] presented a heavy ion transport code, HIT, and the results of calculations which compare the number and total energy of neutrons ($E_n \geq 100$ MeV) emitted external to a central shielding block (radius = 20 cm, height = 60 cm) composed of iron by incident heavy ion beams to those expected from incident proton beams. The heavy ion collision model used in HIT is the HIC [25] code. The number and total energy of neutrons ($E_n \geq 100$ MeV) per source particle energy emerging form the side and back of the central shielding block composed of iron as a function of the atomic number of the projectile are calculated for 3.0 GeV/nucleon incident ions.

RADIATION ENVIRONMENT OUTSIDE HEAVY ION ACCELERATORS SHIELD

Theoretical[26] and experimental[27] studies of neutron fields outside shield of the JINR synchrophasotron were carried out in two geometries: 1) Outside the lateral concrete shield of 1 m thickness irradiated with particles generated in thick copper target by beam of ^{12}C at 3.65 GeV/nucleon and 2) Outside the 2 m concrete beam stopper of ^{12}C at 3.65 GeV/nucleon beam. Neutron fluences, dose equivalents and high-energy neutron spectra were measured and calculated. Measured[26] and calculated[26] lateral distribution of high-energy hadron fluence and measured[26] high-energy neutron spectra outside the stopper are given in Fig. 3 and Fig. 4, respectively.

The calculation[28] of the shielding at the SIS (heavy ion synchrotron) of the GSI was checked experimentally by J. G. Festag [29]. The measurements of the radiation outside the shielding of SIS have shown that in general more than 90% of the equivalent dose are due to neutrons. These measurements are done with ionization chambers, proportional counters and different models of REM counters. There are also used REM counters with an extended range[30]. At the same place and during the same time intervals measurements with a pair of REM counters (one original and one modified) have been taken. They show that sometimes only 50% of the equivalent dose are recorded by a normal REM counter. The proportion of the registered doses (modified /unmodified Rem counter) seems to diminish in reactions of very heavy ions at very high energies[31,32].

INDUCED RADIOACTIVITY

In addition to the prompt radiation, a residual radiation field is also produced that remains after shutdown of the accelerator beam. This radiation field results from the decay of radioactivity induced in the accelerator structure and its ancillary components by the interaction of particles of the prompt radiation field. The induced activity depends on many factors such as the type and energy of the ions accelerated, the mean value of the losses of accelerated ions in the accelerator or in the beam pipes and targets, the duration of these losses and materials irradiated.

The basic information needed for the calculation of dose rates from irradiated materials are radionuclide production rates for the production of a given radionuclide from a target material bombarded by specified ions. An expression for estimating radionucleus production rate, R, in target irradiated by heavy ions with energy less than about 15 MeV/nucleon was proposed by B.B.Back [11]:

$$R = \frac{3761I}{A_{tgt}} \int_0^{R_e} \left(\sigma_{er} + 2\sigma_f\right) dx, \qquad (9)$$

where I is beam intensity measured in particle nanoampheres, A_{tgt} is the atomic weight of the target material, σ_{er} and σ_f are cross section of evaporation residue and fission-like reactions, respectively, R_e is the total effective range of beam particles penetrating the target material. R_e may be calculated as follows

$$R_e = \int_{E_0}^{E_1} \left(\frac{dE}{dx}\right)^{-1} dE,$$

where E_1 is the original beam energy and dE/dx is the energy dependent stopping power, E_0 is the lowest energy at which heavy ion fusion reactions occur.

M.M.Komochkov[18] suggested simple expression for estimation of specific radioactivity induced by light high energy ions with the energy E_i (MeV/nucleon) depending on the specific radioactivity induced by protons with the energy $E_p = E_i \ A_i$ (MeV) in the same target:

$$q_n \cong N \, q_p, \qquad (10)$$

where $N = (A_i)^{0.25} + (A_i - 1)^{0.6} 0.078[\ln A_i - 1.85]$, A_i is atomic mass of the ion.

DESIGN CRITERIA FOR PROMPT RADIATION ON THE RHIC SITE

A worst-case fault in the Relativistic Heavy Ion Collider (RHIC) would be the loss of the full beam at full energy at an arbitrary point (any magnet or device which intrudes into the physical aperture). Harrison and Stevens[33] concluded that the maximum credible fault would be full beam loss at points which are near the limiting aperture of the collider and loss of one half of the full beam at other locations, and that such occurrences should be allowed for at a rate of once in several years[33]. For the purpose of evaluating necessary shielding and access restrictions as applied to a specific location, the design-basis accident (DBA) will be assumed to be the maximum credible fault once per year.

Design criteria for dose limits for the Collider are shown below. They were chosen to be consistent with the Laboratory design goal of 0.25 mSv y^{-1}. Low occupancy is defined as 1/16 [34]. Both normal loss limitations and DBA fault limitations are considered. The annual dose fault limitation applies to both the Injection Line and the Collider.

I. Classification "A": Radiation workers; high occupancy
   1. Normal loss
      0.002 mSv h^{-1}
   2. DBA Fault
      5 mSv y^{-1} limit

II. Classification "B": Radiation workers; low occupancy
   1. Normal loss
      0.032 mSv h^{-1}
   2. DBA Fault
      10 mSv y^{-1} limit

III. Classification "C": Non-radiation workers; high occupancy
   1. Normal loss
      0.15 mSv y^{-1}
   2. DBA Fault
      0.1 mSv y^{-1} limit

IV. Classification "D": Non-radiation workers; low occupancy
   1. Normal loss
      2.4 mSv y^{-1}
   2. DBA Fault
      1.6 mSv y^{-1} limit

Existing regulatory requirements do not explicitly consider low probability fault situations for accelerators[35, 36, 37]. These criteria use the International Commission on Radiological Protection (ICRP) concept of dose averaging[38] and adopts the philosophy that both low occupancy and low probability of faults mitigate allowable dose in a single year if a multi-year average dose for a given individual is acceptably low. With this in mind, this approach asserts that the allowable dose from normal loss should be well within any existing or proposed regulatory requirement, but the DBA scenario, given the fault assumptions made herein, be allowed to exceed yearly dose limitations by a small amount comparable to either: (a) the annual dose from normal losses for radiation workers; or (b) an amount comparable to the natural background for non-radiation workers. In either case, it is important to realize that an actual DBA fault occurrence would most likely represent a once-in-a-lifetime event for a given individual.

Comparisons with known requirements are as follows:

1. Radiation Workers

A. The 0.002 mSv h^{-1} criteria is below the 0.005 mSv h^{-1} design goal specified in DOE Order 5480.11 [35].

B. The annual dose from losses of 0.002 mSv h^{-1} × 2000 h y^{-1} = 4.0 mSv y^{-1} is well below the Federal Dose Limit of 50 mSv y^{-1} [39], the Department of Energy (DOE) Administrative Control Level (ACL) of 20 mSv y^{-1} [37], the Laboratory ACL of 12.5 mSv y^{-1} [40].

C. A worst-case scenario, given the assumptions specified above, would pertain to an individual who spent 15/16 of a year in a Class A area and 1/16 of a year in a Class B area during which he/she was exposed to 10 mSv during fault conditions. The upper limit for annual dose to this individual would be (15/16) × 2000 h × 0.002 mSv h^{-1} + (1/16) × 2000 h × 0.032 mSv h^{-1} + 10 mSv = 18.69 mSv. For the vast majority of the worker population on the RHIC site, this upper limit is not credible because shielding designed for the possibility of a fault in the region of the collider limits the normal-loss hourly dose to values far lower than specified by the criteria[41]. The most likely dose

to a radiation worker exposed to an accident in a Class B area would, therefore, be the fault value itself of 10 mSv. Of the total dose in this extremely improbable situation, 14 mSv would occur in the low-occupancy region which exceeds the design goal of 10 mSv y^{-1} in low occupancy areas in DOE Order 5480.11 Section 9.J.1.b [35].

2. Non-radiation workers

A. The annual normal loss dose of 0.15 mSv is below the Laboratory design goal of 0.25 mSv y^{-1} per facility. With allowance of a worst case fault, the goal is exactly met.

B. A worst-case scenario would pertain to an individual who spent 15/16 of a year in a Class C area and 1/16 of a year in a Class D area during which he/she was exposed to 1.6 mSv during fault conditions. The maximum annual dose here would be ~1.89 mSv which is both comparable with natural background and is expected to represent a very rare occurrence for a given individual. In addition it is reasonable to assume that the exposed population for this case is one person or at most a very few people.

3. Regulatory limits and guidance for the general public

The following are the existing regulatory requirements and guidance for the general public which do not explicitly address low probability faults[42]:

ICRP-60
- Limit annual effective dose to 1.0 mSv averaged over any 5 consecutive years

NCRP-91
- Limit annual effective dose to 1.0 mSv for continuous or repeated exposure
- Limit annual effective dose to 5.0 mSv for infrequent exposure

10 CFR 20
- Limit annual effective dose to 1.0 mSv
- Limit dose in 1 h to 2.0 mSv

DOE Order 5400.5
- Limit annual effective dose to 1.0 mSv
- If avoidance of higher exposure is impractical, temporary limit not to exceed 5.0 mSv may be authorized

Four area classifications are proposed for the RHIC site for regions accessible without restriction by physical barriers. The classifications are distinguished by occupancy and by whether radiation worker training is required for entry. Each classification is specified by limits on dose equivalent resulting from both anticipated beam loss and from design basis accident faults. Although no explicit regulatory requirements exist for low probability faults, the highest proposed fault limits, 10 mSv y^{-1} in low occupancy regions restricted to radiation workers and 1.6 mSv y^{-1} in low occupancy uncontrolled regions, are compatible with several recommendations[42] that consider infrequent exposures and multi-year dose averaging for given individuals.

Note: A more complete explanation of the Criteria can be found in Health Physics, Vol. 66, No. 3, (1994).

CONCLUSIONS

The data presented here demonstrate the need for more detailed measurements. Neutron yields, spectral and angular distributions produced by ions in thick target as function of ion energy, ion species and target material are required. Detailed investigation of the radiation environment at heavy ion accelerators as function of ion energy, ion species as well as attenuation measurements in shielding are needed.

Radiological safety aspects of the operation of heavy ion accelerators does not essentially differ from proton accelerators, but a heavy ion accelerator is able to accelerate a number of ions species with different energies, therefor the operational health physics of heavy ion accelerators is more difficult than the operational health physics of proton or electron accelerators, since the radiation fields is varying to some extent with the different accelerated ions.

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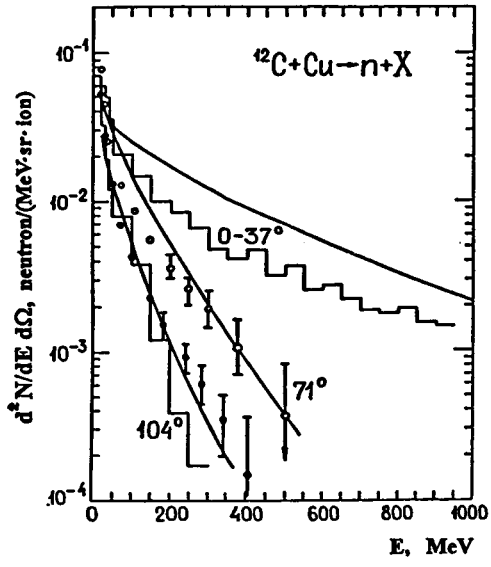

Fig. 1 Experimental (points[9]) and calculated (solid curves[21], histogram[22]) spectra of high energy neutrons.

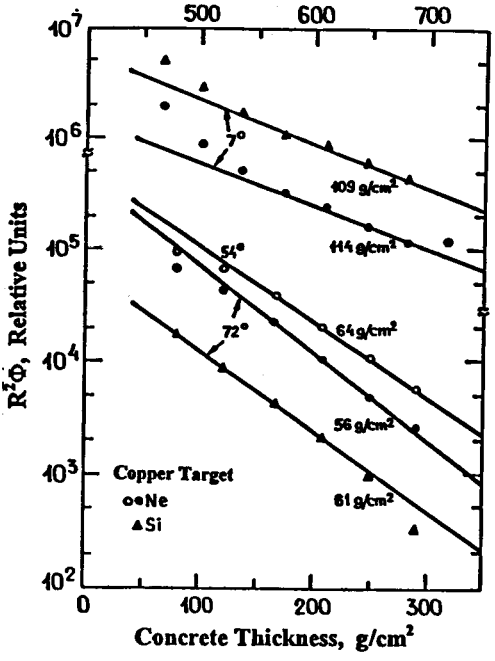

Fig. 2 Attenuation curves in concrete[3].

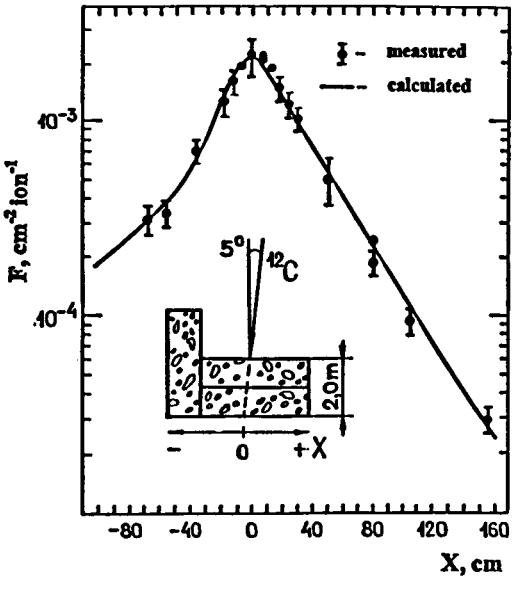

Fig. 3. Measured[27] and calculated[26] lateral distribution of high energy hadron fluence outside the beam stopper of ^{12}C 3.65 GeV/nucleon.

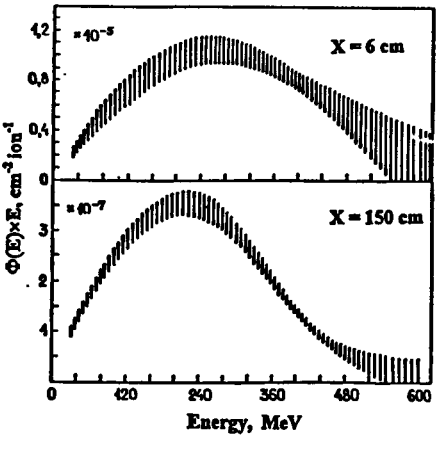

Fig. 4 Neutron spectra outside the beam stopper of ^{12}C 3.65 GeV/nucleon[27] (position of measurements see in Fig. 3).

INSTRUMENTATION AND DOSIMETRY AT ACCELERATOR FACILITIES

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INTRODUCTION

The increasing use of high energy and intense particle accelerators has brought the importance of detection and monitoring of high energy neutrons which are most penetrating. The spectrometry and dosimetry of high energy neutrons at accelerator facilities are however very difficult and many works have ever been done, but no established instrumentation still exists.

We have newly developed two types of high energy neutron spectrometers, Bi spallation detector and organic liquid scintillation detector with plastic veto counter, and also two types of wide-energy range personal neutron dosimeters, real-time Si semiconductor detector and CR39 track detector.

The 12.5-cm diam by 12.5-cm long organic liquid scintillation detector, such as NE-213 and BC-501A, was used to measure neutron spectra in the energy range from about 1 MeV up to about 130 MeV, by coupling with the n-γ discrimination technique. The thin (5 mm thick) plastic scintillator (NE-102A) was set in front of the neutron detector as a veto counter to discriminate the incoming charged particles. The Bi spallation detector utilizes the Bi(n,xn) reactions up to Bi(n,14n), which have different threshold energies. This detector is very small and light, and can give the gross neutron spectrum over the energy range beyond 100 MeV.

A real time personal dosimeter consists of three types of Si detectors, thermal neutron sensor, fast neutron sensor and gamma-ray sensor (1,2). The thermal neutron sensor which is B-10 doped n-type silicon with a polyethylene radiator detects alpha particles from $^{10}B(n,\alpha)$ and protons from H(n,n) reactions, which is mainly sensitive to neutrons from thermal to 1 MeV. The fast neutron sensor which is p-type silicon with polyethylene is sensitive above 1 MeV, and the gamma-ray sensor is the same as the fast neutron sensor without polyethylene. By taking the weighted sum of two neutron sensor counts, this dosimeter can give the neutron dose equivalent within about 50% errors in wide energy range from thermal to several tens MeV.

A passive personal neutron dosimeter using CR39 was also developed (3). The dosimeter has two pieces of CR39 which are contacted with boron nitride and polyethylene radiators. The former is sensitive to lower energy neutrons and the latter to higher energy neutrons. The weighted sum of two CR39 etch-pits gives neutron dose equivalent within about 30% errors in the energy range from thermal to 15 MeV.

The response functions of these four detectors as a function of neutron energy were measured using the monoenergetic neutron field from 8 keV to 15 MeV at the Dynamitron facility, Department of Nuclear Engineering of Tohoku University (FNL), and the quasi-monoenergetic neutron fields produced by $^{7}Li(p,n)$ reaction at four cyclotron facilities; Cyclotron and Radioisotope Center of Tohoku University (CYRIC), Institute for Nuclear Study of University of Tokyo (INS), Takasaki Research Establishment of Japan Atomic Energy Research Institute (JAERI) and Institute of Physical and Chemical Research (RIKEN).

ESTABLISHMENT OF MONOENERGETIC AND QUASI-MONOENERGETIC NEUTRON CALIBRATION FIELDS

Dynamitron Neutron Field

The monoenergetic neutron field was developed at the Dynamitron accelerator facility, FNL. Monoenergetic neutrons of 8 and 27 keV, 0.25 and 0.55 MeV, 1.0 and 2.0 MeV, 5 MeV, 15 MeV were

obtained in the forward direction to the incident beam axis by Sc(p,n), ^{7}Li(p,n), T(p,n), D(d,n), T(d,n) reactions, respectively. The absolute neutron fluences were measured with ^{6}LiF-SSD for 8 and 27 keV neutrons and with ^{235}U fission chamber for other neutrons. The neutron fluence during the experiment was monitored simultaneously with the proton-recoil proportional counter. The neutron fluences were 1.0×10^{2} n cm^{-2} μC^{-1} for 8 keV, 2.4×10^{1} n cm^{-2} μC^{-1} for 24 keV, 1.0 - 4.0×10^{4} n cm^{-2} μC^{-1} for others.

CYRIC Neutron Field

The CYRIC neutron field has the 45 m long neutron TOF facility coupled with the beam chopping system and the beam swinger system. The quasi-monoenergetic neutrons of 22.0 and 32.5 MeV having 1.7 and 1.4 MeV FWHM were obtained from 2 mm thick ^{7}Li target bombarded by 25 and 35 MeV protons, respectively, and the proton beam hit the target at 10 deg through the swinger magnet and was fully stopped at the Faraday cup. The neutrons were extracted in the TOF facility through the 50 cm thick iron-polyethylene collimator of 30 cm x 20 cm aperture settled in the 280 cm thick concrete wall of 100 cm x 50 cm aperture. The absolute neutron fluence of this field was measured with the proton recoil counter telescope (PRT) and the neutron spectrum was measured with the TOF method. The 22.0 and 32.5 MeV peak neutron fluences were 1.1×10^{3} and 1.7×10^{3} n cm^{-2} μC^{-1} at the collimator exit behind 8.6 m from the target. The neutron fluence during the experiment was monitored simultaneously with the ^{238}U fission chamber fixed closely to the target.

INS Neutron Field

The INS neutron field was used only for neutron irradiation and the irradiation samples were placed 10 cm away from the Li target in the forward direction, in order to get high neutron fluence and to depress the contribution of room-scattered neutrons, since the irradiation room is small in space. The neutron spectra are the same as those in the CYRIC neutron field. This field is now not available.

TIARA Neutron Field

The TIARA neutron field was established in the neutron beam line collimated into 10 cm diameter. The 2 to 5 mm thick ^{7}Li target settled in the cyclotron room was bombarded by the proton beam of 20 to 90 MeV at 0 deg and the protons passed through the target were bent down to the beam dump by a clearing magnet, and the neutrons produced at 0 deg were extracted through the 220 cm thick concrete wall. The absolute fluence of source neutrons was determined with PRT and the neutron fluence during the experiment was monitored simultaneously with the ^{238}U and ^{232}Th fission chambers fixed closely to the target. The FWHM of 40 and 65 MeV monoenergetic peak and the peak neutron yield have the respective values of 2.0 MeV and 2.1×10^{4} n cm^{-2} μC^{-1}, 2.1 MeV and 3.2×10^{4} n cm$^{-2}\mu$C^{-1}, at the collimator exit behind 4 m from the target, for 43 and 67 MeV proton incidence. The neutron spectra measured with the TOF method.

RIKEN Neutron Field

The RIKEN neutron field is now being established at the E4 experimental room of the separate sector ring cyclotron. The proton beam having energies of 80, 90, 100, 110, 120, 135, 150 and 210 MeV were injected on a 10 mm thick ^{7}Li target through the beam swinger. Protons passed through the target were cleared out by the magnet and absorbed in the spectrograph. Neutrons produced at 0 deg were transported through the iron-concrete collimator of 20 cm by 20 cm aperture and 120 cm length. The neutron spectra were measured with the TOF method using BC501A and the absolute neutron fluence with the Li activation method using the ^{7}Be activity from the ^{7}Li(p,n)^{7}Be reaction.

MATERIALS AND METHODS

Bi Spallation Detector

Two sizes of natural bismuth samples were prepared. Thick samples are 80 mm diam by 10 mm thick and thin samples are 30 mm diam by 2 mm thick, which have chemical purities of 99.999 %.

The neutron reaction cross sections of Bi in the energy region from 20 to 210 MeV were measured by irradiating these samples in the p-Li quasi-monoenergetic neutron fields at INS, TIARA and RIKEN. The gamma-ray activities of the irradiated samples were counted by using a Ge detector and the reaction rates of identified radioisotopes were obtained after correction of self absorption and sum-coincidence effects.

By using the neutron energy spectrum $\Phi(E)$ and the reaction rate, A, the activation cross section $\sigma(E)$ can be estimated as follows. The reaction rate, A is divided into two parts; one is induced by the peak energy neutrons and the other by the low energy continuum neutrons, as

$$A = N \int_{Eth}^{Emin} \sigma(E)\,\Phi(E)\,dE + N\,\sigma(Ep)\,\Phi(Ep)$$

where N : number of target atoms relating to the relevant reaction,
E_{th} : threshold energy,
E_{min}: lowest energy of monoenergetic peak neutrons,
$\sigma(Ep)$: cross section at peak neutron energy,
$\Phi(Ep)$: monoenergetic peak neutron flux.

If the threshold energy E_{th} is higher than E_{min}, the first integration term must be zero. Otherwise, this term can be estimated by successive subtraction method using the neutron flux $\Phi(E)$ having lower peak energy. The $\sigma(E)$ values in lower energy region were cited from the evaluated data files, ENDF/B-VI (4), McLane et al. (5) and so on.

Organic Liquid Scintillation Detector

The cylindrical organic liquid scintillator of 12.7 cm diam by 12.7 cm long NE-213 or BC501A is coupled to a R4144 photomultiplier connected to the E1458 base (Hamamatsu Photonics Co. Ltd.) which is specially designed to expand the dynamic range of output pulses for high energy neutron measurements. The detector size which stops the recoil proton up to about 120 MeV was selected in order to to be able to detect as higher energy neutrons as possible with keeping a good n-γ pulse-shape discrimination capability.

The response functions of the scintillator were measured at CYRIC, TIARA and RIKEN (6). At CYRIC, 35-MeV protons and 50- and 65-MeV ^{3}He ions were injected into a 10 mm thick Be target. At TIARA, 67 MeV proton beam was injected into a 7.6 mm thick copper target, and at RIKEN, 135 MeV proton beam was injected into the Be (70 mm thick) + C (20 mm thick) target. These targets produced white spectral neutrons and the response functions down to a few MeV neutron energy were obtained by grouping them into monoenergetic interval with the TOF method coupled with the beam chopping system.

Si Semiconductor Dosimeter

The dosimeter installs two neutron sensors and one gamma-ray sensor, which enables us to give neutron and gamma-ray doses at the same time (1,2). One type, slow neutron sensor, is an n-type silicon crystal on which a p+ layer of elementary boron enriched 94% ^{10}B is deposited in about 1 mm thickness and the other type, fast neutron sensor, is an p-type silicon crystal without boron coating. Both crystals are contacted with 0.08 mm thick polyethylene radiators and in some cases only the slow neutron sensor is covered with thermal neutron filter of 0.5mm thick cadmium to improve its energy response. Figure 1 shows the schematic cross sectional view of two neutron sensors. The gamma-ray sensor is the p-type silicon detector of 3mm x 3mm without any radiator.

These three sensors are encapsulated in a metal package together with the charge sensitive preamplifier and the linear amplifier circuits. The output pulses from the sensors are counted through the pulse height discriminator and fed to the CPU for computing and displaying the dose equivalent values. The external size of the dosimeter is 100mm x 60mm x 20mm and its weight is about 170g.

The neutron detection efficiencies of these two sensors were measured in the monoenergetic neutron field at FNL. Monoenergetic neutrons of 0.2, 0.55, 1, 2, 5 and 15 MeV were produced using the Dynamitron accelerator. The dosimeter was placed in the forward direction to the beam axis. The efficiency measurement of the slow neutron sensor to thermal neutrons was done in the experimental hole of the TRIGA-II type reactor of Rikkyo University and in the thermal neutron field leaked from a graphite pile of the Institute of Radiation Measurements (IRM). The dosimeter was placed in front of a commercially available ellipsoidal water phantom, 45 cm high and 30 cm wide. The output pulses due to alpha particles produced by the ^{10}B(n,α) reaction and protons recoiled from the elastic collision in the polyethylene radiator were measured with a multi-channel analyser, in this measurement.

CR39 Track Dosimeter

The CR39 plastic has higher sensitivity than the NTA film for lower energy neutron, however many background etch-pits appear on the surface of nonirradiated CR39 plastic. In order to depress this background contamination, we diminished it by dyeing the CR39 plastic before etching with a KOH solution. The number of background etch-pits per cm^2 decreased about a factor of 10 after dyeing. As shown in Fig. 2, the CR39 dosimeter is contacted with two kinds of radiators, one is a 0.2 mm thick boron nitride for measuring low energy neutrons and the other is a 1 mm thick polyethylene for measuring fast neutrons.

The energy response of the CR39 etch-pits behind the two radiators were measured with the monoenergetic neutron field in the energy range of 8 keV to 15 MeV at FNL, graphite-moderated thermal neutrion fields at the Electro-Technical Laboratory and IRM, and the quasi-monoenergetic neutron field of 22 and 33 MeV at CYRIC. In these experiments, the dosimeter was also attached on a ellipsoidal water phantom.

RESULTS AND DISCUSSIONS

Bi spalltion detector

From the above experiments, we could identify the radionuclides of ^{207}Bi to ^{196}Bi produced by ^{209}Bi(n,2n) to ^{209}Bi(n,14n) reactions, respectively. Figure 3 indicates the cross section data of ^{209}Bi(n,2n) to ^{209}Bi(n,10n) reactions which have already been analyzed, in comparison with other experimental data (5) and the ENDF/B-VI high energy file data calculated by the ALICE code(4). Our data are generally in good agreement with them, but a large discrepancy can be found for Bi(n,9n) reaction, which might indicate a possibility of the inaccurate data of decay scheme and/or branching ratio of ^{201}Bi . This figure clearly shows that the excitation functions of Bi(n,xn) reactions have simple and similar shapes, and the threhold energies differ at about 8MeV interval, which indicates that these reactions are quite useful for high energy neutron spectrometry.

The Bi detector has been used by our group to measure neutrons leaked through bulk shield of 3m thick iron and 1m thick concrete at ISIS of Rutherford Appleton Laboratory.

Organic Liquid Scintillator

We obtained the neutron response functions of the scintillator by sampling neutron events into neutron energy intervals of 1, 2, 4, 5 MeV for neutron energy range from 0 to 44, 44 to 70, 70 to 90, 90 to 130 MeV, respectively (6). The response matrix of 68 neutron bin x 70 light output bin in the neutron energy range of 0 to 120 MeV was formed by using the measured response functions from 5 to 120 MeV neutrons and the SCINFUL Monte Carlo calculations (7) below 5 MeV neutron.

This detector has being used for neutron shielding experiment at TIARA and for thick-target yield experiment with heavy ions at the heavy ion medical synchrotron facility, National Institute of Radiological Sciences (HIMAC).

Figure 4 shows the cross-sectional view of the shielding experiment at TIARA. Inside the 340-cm-thick concrete shielding wall between the cyclotron room and the experimental room, an empty space of 120 cm x 120 cm x120 cm is equipped for shielding experiments. The concrete shield of 25 to 200 cm thickness and the iron shield of 10 to 130 cm thickness were fixed in contact with the 10.9-cm-diam collimator exit located at 4 m from the neutron target. The neutron energy spectra transmitted through shields were measured with the BC501A scintillator, using 40 and 65 MeV quasi-monoenergetic neutron sources which were produced by 43 and 68 MeV p-^{7}Li reactions. In Fig. 5, neutron spectra penetrated through concrete which were unfolded from the measured light output distributions with the FERDOU code (8) are compared with the MORSE Monte Carlo calculation (9) using the DLC-119/HILO86 group cross section library (10), as examples. This figure revealed in general the good agreement between experiment and calculation.

Figure 6 shows the schematic diagram of the experiment at HIMAC. Neutrons produced from a target bombarded by heavy ion beams were measured with three sets of the 12.7 cm diam by 12. 7cm long NE213 scintillators by the TOF method. A veto counter of 5 mm thick NE102A plastic scintillator was set in front of the NE213 to discriminate the secondary charged particles. A 0.5 mm thick NE102A scintillator fixed upstream to the target was used to produce beam pick-up signals for the TOF starting. The neutron energy spectra produced at 0 degree from 20 mm thick C, Al, and 10 mm thick Cu, Pb targets which were bombarded by 100 MeV/nucleon ^{12}C ions are shown in Fig. 7, as examples.

Si Semiconductor Dosimeter

Figure 8 shows the neutron detection efficiency of the dosimeter as a function of neutron energy. The measured results are the sum of the integrated counts given by the slow neutron sensor and the integrated counts of the fast neutron sensor multiplied by a factor of 20, in order to get the detection efficiency as close as possible to the fluence-to-dose-equivalent conversion factor given by ICRP-51 (11) which is drawn in a dotted line. In Fig. 8, the results calculated with the Monte Carlo method are also shown to compare with the measured results. Very Good agreement between experiment and calculation can be seen in the energy below 5MeV. Above 5MeV, the calculation underestimates the measured results due to the neglection of the contribution of charged particles which are produced in the silicon crystal itself and the surrounding material by various neutron reactions.

This dosimeter which combines two silicon sensors has neutron sensitivity over a wide energy range from thermal to 15 MeV and also has good energy response, excluding a large deviation from the ICRP-51 response curve in the energy range from 50 keV to 1 MeV, as seen in Fig. 8. Considering the energy response of the dosimeter shown in Fig. 8, the neutron dose equivalent H can be given by adding the neutron dose equivalent of energy higher than 1 MeV, H_f and that of energy lower than 1 MeV, H_s. H_f and H_s are given by

$$H_f = K_f C_f \qquad \text{for En} \geq 1 \text{ MeV},$$
$$H_s = K_s C_s \qquad \text{for En} \leq 1 \text{ MeV},$$

where C_f, C_s are the respective counts measured with fast sensor and slow sensor, K_f, K_s are the respective conversion factor in units of μSv/count. The K_f and K_s values were determined as 0.5 and 10 μSv/count, respectively, from the estimation in the fission and 1/E slowing-down spectral fields.

In order to investigate the accuracy of this dose estimation method in a wide variety of actual neutron fields, we have done the field test of the dosimeter calibration in the following typical neutron fields having known neutron energy spectra; 1) moderated ^{252}Cf neutron source calibration field, 2) a beam extraction hole of the fast neutron source reactor of University of Tokyo, 3) labyrinth from the 40 MeV cyclotron room of Tohoku University, 4) MOX (Mixed Oxide) fuel handling room of Power Reactor and Nuclear Fuel Development Corporation, 5) 14MeV d-T neutron field penetrated through concrete shield, and 6) On the outer surface of the concrete shield surrounding several hundreds MeV electron synchrotron. In these field tests, the dosimeter was fixed on the water phantom faced to the neutron beam direction. The measured counts were compared with the dose equivalent values obtained from the dose equivalent counters (rem counters) of Studsvik Co. 2202D and Fuji Electric Co. NSN1 which were used as neutron dose monitors.

Figure 9 summarizes the results of these field tests. The ratio of neutron dose equivalent value measured with this personal dosimeter to that with the rem counter is shown as a function of neutron energy in the test field averaged by weighting with the dose equivalent. The ratio must be equal to 1 for the ideal dosimeter, and our personal dosimeter gives neutron dose equivalent within a factor of 2 margin of accuracy, excluding a special field in which thermal neutron fluence occupies more than 50% of total neutron fluence, such as at some positions in the labyrinth from the cyclotron room. Even in such a field, we can estimate the neutron dose equivalent within a factor of 3 by using a thermal neutron sensor with Cd cover and moreover with much better accuracy by adjusting the conversion factor K_s slightly.

CR39 Track Dosimeter

Figure 10 shows the weighted sum of the energy response of the dosimeter with two radiators, boron nitride (BN) and polyethylene (poly), in order to give a best fit to the ICRP51 response curve. The data at thermal, and 8 keV up to 15 MeV energies are the measured results (shown in white circles and triangles), and the data between thermal and 8 keV energies are the calculated results with the Monte Carlo method (in white squares). Good agreement between experiment and calculation can be seen and the energy response of this dosimeter well fits the ICRP51 response curve in the energy range lower than about 10 MeV. In order to improve the energy response above 10 MeV, the weighting factor was further increased only for the etch-pits having a diameter larger than 20 μm, considering that the diameter of etch-pits increases with neutron energy. The thus-adjusted energy response of the dosimeter is also shown as white square signs in Fig. 10 and indicates much better agreement with the ICRP51 response curve above 10 MeV. The dose equivalent in mSv is then given by the following formula,

$$H\,(\text{mSv}) = \{BN/8 + [poly1 + (poly2 - 100) \times 8] \times 8\} \times 10^{-4}$$

where BN, poly1 and poly2 are the etch-pits per cm^2 of CR39 with BN and those having a diameter smaller and larger than 20 μm with polyethylene radiators, respectively. If poly2 is less than 100, the third term of the above formula is neglected.

CONCLUSION

We have developed two types of high energy neutron spectrometers, and active and passive personal neutron dosimeters. The two spectrometers are now in use for neutron spectrometry at accelerator facilities and these two dosimeters will be commercially available very soon.

ACKNOWLEDGMENTS

The authors wish to thank Dr. Y. Kumamoto and Mr. A. Fukumura for their kindful cooperation during the HIMAC experiment and the machine group members for their four cyclotron operations during our experiments.

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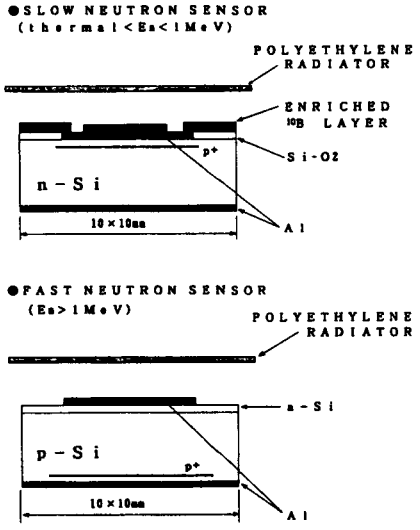

Fig. 1 Schematic cross sectional view of two silicon neutron sensors

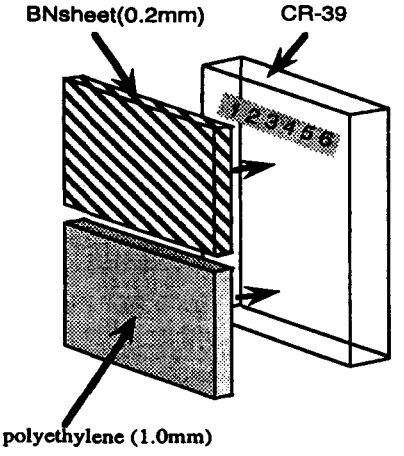

Fig. 2 Schematic view of CR39 plastic with two radiators

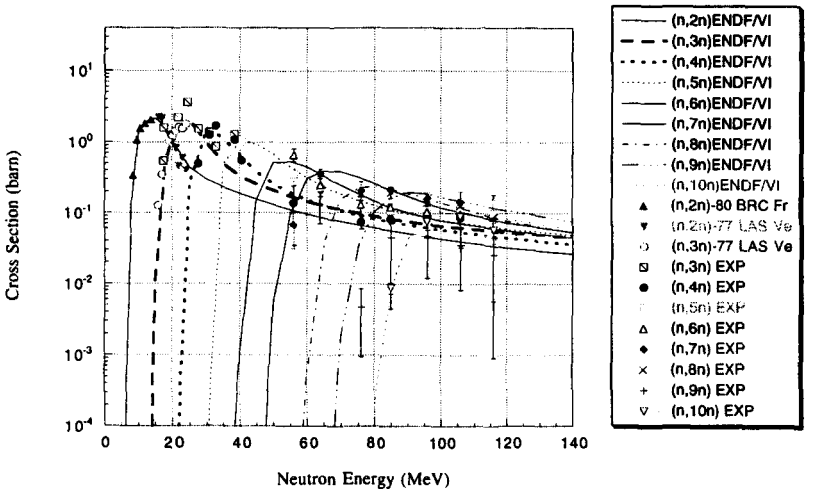

Fig. 3 Measured cross section data of $^{209}Bi(n,xn)$ reactions compared with ENDF/B-VI (4) and other experimantal Data (5)

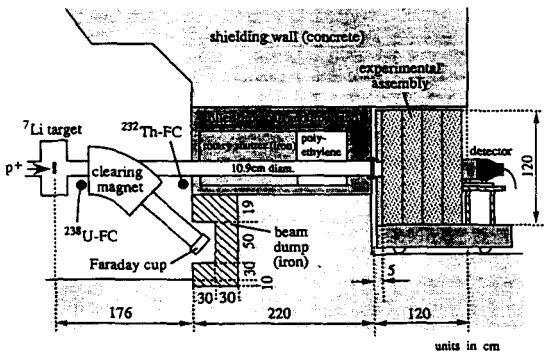

Fig. 4 Cross sectional view of the shielding experiment at TIARA

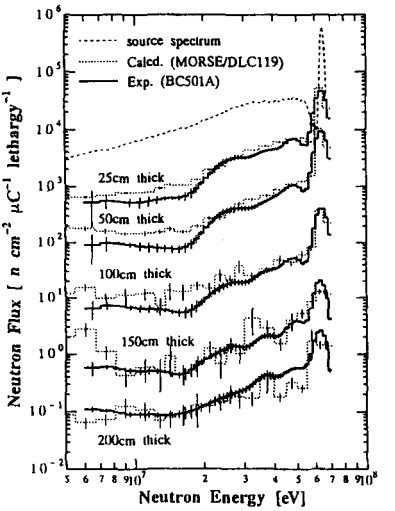

Fig. 5 Comparison of neutron spectra transmitted through concrete of various thicknesses between experiment and MORSE calculation for 65 MeV quasi-monoenergetic neutron source

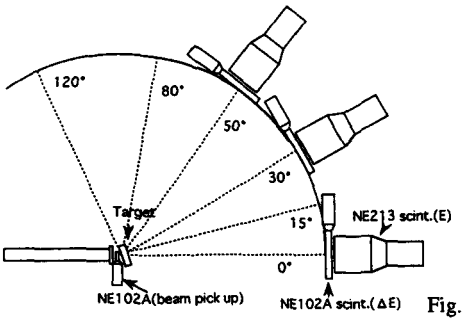

Fig. 6 Schematic diagram of the thick-target yield experiment at HIMAC

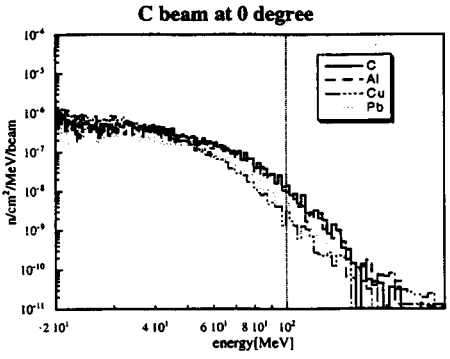

Fig. 7 Neutron spectra produced at 0 degree from 20-mm thick C, Al and 10-mm thick Cu, Pb targets bombarded by 100 MeV/nucleon C ions

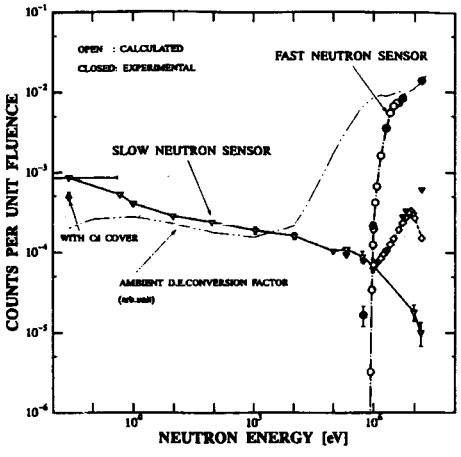

Fig. 8 Comparison of measured and calculated neutron detection efficiencies of the Si dosimeter, together with the ICRP-51 fluence-to-dose-equivalent conversion factor

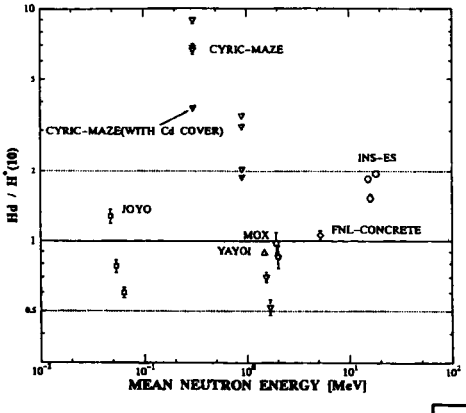

Fig. 9 Comparison of dose equivalent given by the Si dosimeter to that by the rem counter for various neutron fields having different mean neutron energies

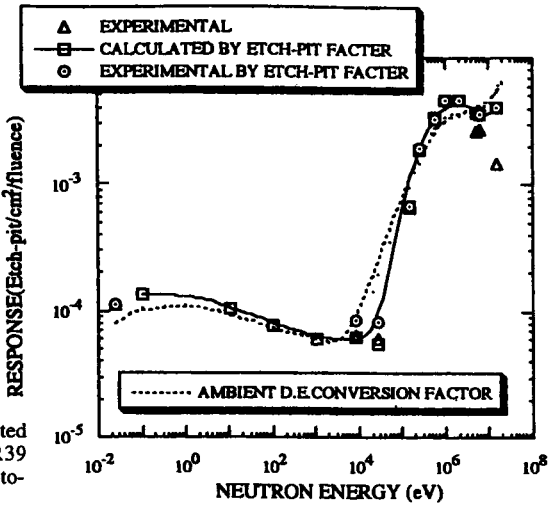

Fig.10 Comparison of measured and calculated neutron detection efficiencies of the CR39 dosimeter, together with the ICRP-51 fluence-to-dose-equivalent conversion factor

RADIATION PROTECTION AT THE LHC, CERN'S LARGE HADRON COLLIDER

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ABSTRACT

After a brief description of the Large Hadron Collider (LHC), which will produce 7 TeV on 7 TeV proton collisions, some of the radiological questions it raises will be discussed. The machine will be built in the 27 km circumference ring-tunnel of an existing collider at CERN. It aims to achieve collision rates of 10^9 per second in two of its high-energy particle detectors. This requires two high-intensity beams of more than 10^{14} protons each. Shielding, access control and activation in addition to the high power in the proton-proton collisions must be taken into account. The detectors and local electronics of the particle physics experiments, which will surround these collisions, will have to be radiation resistant. Some of the environmental issues raised by the project will be discussed.

THE DESIGN CONCEPT OF THE LHC

The Large Hadron Collider (LHC) of CERN will be a synchrotron-collider which accelerates and stores two intense beams of particles circulating in opposite directions and collides them head-on at two or more points where particle physics detectors can study the interactions. While CERN's present collider, LEP, collides electrons and positrons at energies up to 100 GeV, the LHC will collide two beams of protons at energies of up to 7 TeV. The LHC will be installed in the same underground tunnel which at present houses LEP alone (see Figure 1). Since the circumference of the LHC is given by the existing LEP tunnel, the maximum beam energy depends only on the magnetic field which can be reached in the high quality dipole magnets needed to guide the protons around the 27 km of the tunnel. These magnets, which will use superconducting coils of NbTi cooled to 1.9 K are the subject of a vigorous R and D programme as they need to provide a field of 8.4 T, almost 50% higher than that presently foreseen for any other accelerator.

The LHC is designed to study quark-quark interactions whose point-like nature implies that cross-sections decrease as the energy E increases. To maintain a constant detection probability for a given type of event, the luminosity, which is proportional to collision rate, should increase at least as E^2. To explore rare processes such as Higgs$\rightarrow 2Z^0 \rightarrow 4\mu$ the LHC must be able to provide luminosities of the order of $10^{34}\,\mathrm{cm}^{-2}\mathrm{s}^{-1}$ (interaction rates of approximately 10^9 per second). To achieve this, high intensity circulating beams are needed with very small cross-sections at the collision point. The LHC will circulate two opposing proton beams in ingenious "two-in-one" magnets which provide the necessary twin magnetic channels with opposite sign fields in the same yoke and cryostat.

The layout of the LHC, indicated in Figure 2, is given by the form of the LEP ring which consists of eight arcs with a bending radius close to 3.5 km linked together with 550 m long straight sections to form a regular structure. The two beams of the LHC will lie side by side in the horizontal plane in the arcs, 194 mm apart, and will cross over in the centre of each straight section where collisions are required. Two such regions, P1 and P5, will be used for large general purpose LHC detectors, called ATLAS and CMS. The Technical Proposal for each of these has been prepared by collaborations of some 1500 particle physicists from all over the world. Two smaller and more specialised experiments, ALICE which will study lead-ion collisions and LHC-B which will study particles with the fifth "beauty" quark and CP

violation, will be installed at P2 and P8, respectively. The remaining straight sections will be used for LHC machine utilities, one to provide safe external beam dumps for the beams at the end of each run, one for the accelerating system of Radio Frequency cavities and the other two for the beam cleaning systems which will be needed to ensure that halo particles cannot reach the vacuum chamber walls and deposit their energy in the superconducting magnets.

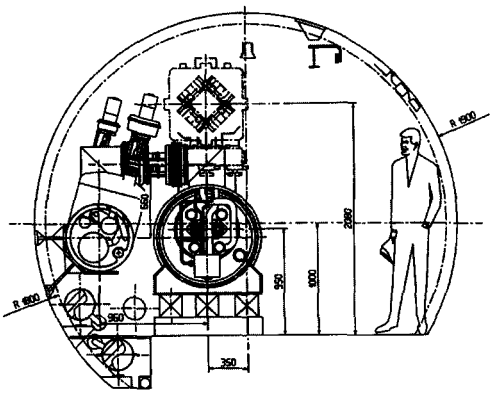

Figure 1: A cross-section of the machine tunnel with the twin aperture super-conducting magnet of the LHC installed below a LEP quadrupole. The separated cryogenic feed-line and connecting valve box can be seen on the left.

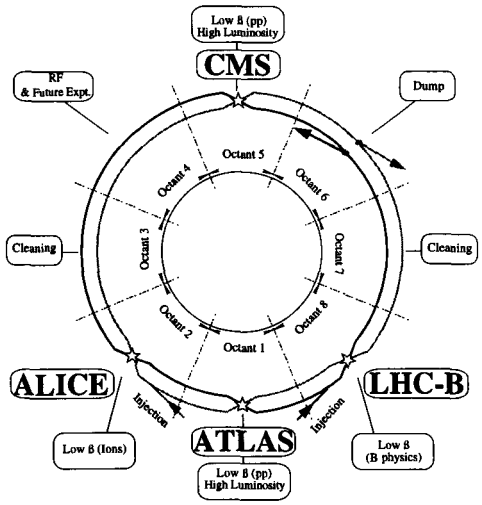

Figure 2: A schematic layout showing the assignment of the eight long straight sections of the LHC to experiments and utilities

The LHC will obtain its protons from the existing CERN accelerator chain of 50 MeV Linac, 1.4 GeV Booster, 26 GeV PS (Proton Synchrotron) and the 450 GeV SPS (Super Proton Synchrotron). The requirements of the LHC, large numbers of high intensity proton bunches with the correct small emittance, are only slightly beyond the routine performance of this injector chain and the necessary development is well underway. As these same injectors frequently operate with other particles, notably heavy ions such as lead, the LHC will also be able to accelerate and collide heavy ions. Luminosities of up to $10^{27}\,\mathrm{cm}^{-2}\mathrm{s}^{-1}$ should be reached with lead ions in the LHC with the total energy of the collisions between these nuclei being about 1150 TeV.

The luminosity, $\mathcal{L}$, per collision point with two equal beams is given by:

$$\mathcal{L} = N^2 k f \gamma / 4\pi\epsilon\beta^*$$

where N is the number of protons in each of k circulating bunches, f is the revolution frequency, β^* is the value of the betatron function at the collision point and ϵ is the emittance corresponding to the 1σ contour of the beam, normalized by multiplying by the Lorentz factor $\gamma = (E/m_0c^2)$.

It is easy to see that high luminosity requires as many high density bunches as possible with the smallest possible transverse cross-section at the collision point (N and k large, $\epsilon\beta^*$ small). The main limit to such a simple approach comes from the so called beam-beam effect which results from the extremely non-linear fields seen by the particles of one beam as they pass through the bunches of the other. The long range

interaction between the bunches of each beam which occurs on either side of the crossing points where the beams pass through a common length of vacuum pipe, about 100 m long, is reduced by introducing a small but finite crossing angle. A careful cost optimization of performance has led to the main parameters of the LHC given in Table 1.

Table 1: Main LHC parameters

	Protons	Pb-Ions
Centre of mass total energy (TeV)	14	1148
Magnetic field in bending magnets (T)	8.4	8.4
Initial luminosity per collision region ($cm^{-2}s^{-1}$)	10^{34}	2×10^{27}
Bunch spacing (m/ns)	7.5/25	37.4/124.8
No. of particles per bunch	10^{11}	9.4×10^{7}
Number of collision regions assumed	2	1
Beta parameter at interaction point (m)	0.5	0.5
r.m.s. beam radius at collision point (mm)	16	15
r.m.s. collision region length (mm)	54	53
r.m.s. energy spread $\delta E/E$	1.1×10^{-4}	1.1×10^{-4}
Beam crossing angle (μrad)	200	100
Luminosity lifetime (h)	10.0	6.7
Stored energy per beam (MJ)	334	4.8
Synchrotron radiation per beam (kW)	3.6	–

For the reasons given above the LHC requires the highest possible magnetic field, which makes the twin aperture superconducting dipoles the most technologically challenging components. The development of these magnets was started in 1985, in close collaboration with European Industry and other Accelerator Laboratories, with the aim of achieving 10 T fields. Progress has been impressive and five 10 m prototypes built in industry have been delivered and tested. Three of these magnets were assembled in a "string-test" during 1995 to verify the cryogenic system needed to bathe the coils of all 1344 dipoles and some 2000 smaller magnets in superfluid helium and maintain them at 1.9 K. As a result of a year's successful testing, the basic concepts of cooling and powering the LHC magnets have been well validated and a number of changes have been incorporated into the latest design report (1).

The massive cryogenic system needed will be based on that already installed to cool the superconducting accelerating cavities of the LEP 200 project. The system will need a considerable upgrade, however, to lower the temperature from the 4.5 K used for the LEP cavities to the 1.9 K needed for the LHC magnets and to provide the increased cooling power needed to cool down 31,000 tons of cold mass in a reasonable time. A total of about 400,000 litres of liquid helium will be needed, but nonetheless the LHC will be extremely energy-efficient, providing twenty times the collision energy and ten thousand times the luminosity of the SPS p-pbar collider which operated in the 1980s, for a similar energy consumption.

During the whole LHC operation cycle of particle injection, acceleration, colliding and finally extracting and dumping the beams, one of the most critical systems will be the beam cleaning system. A series of collimators must intercept the beam halo particles and prevent them from depositing energy in the coils of the magnets and causing the superconducting strands of the coils to become resistive *i.e.* to quench. If this should happen, for whatever reason, the stored energy in the coils of the magnets will be rapidly extracted to avoid destroying them. It is estimated that a point loss of as few as 10^6 protons per second could quench a magnet, while the losses from small angle elastic scattering in the collision regions alone will scatter of the order of 10^9 protons per second into the halo region.

At the end of a colliding beam run, the two beams will be deflected by fast kicker magnets, providing 18 Tm with a 3 μsec rise time, into extraction channels leading to dump blocks about 750 m away. An appropriate gap will be left in the normal bunch structure during injection so that with proper time synchronization the kicker magnet can extract the beam without excessive losses. The extraction channels will have to be equipped with sweeping magnets to spread the beams over the dump blocks. Even so the inner core of the dumps will have to be made of graphite in order to withstand the very high ($\sim$ 2000°C) temperatures generated by the intense proton beams during the dumping operation. By diluting the beam in this way, the 330 MJ of each beam can be safely absorbed without incurring a risk of destroying the absorber block by the thermal shock. The absorbers will be 14 m long blocks of iron with central cores of graphite encased in aluminium. The iron may have to be water-cooled to allow repeated use during setting up and machine studies and the assemblies will be installed in caverns, designed to cope with the induced radioactivity. Like most of the rest of the LHC, these caverns will be some 100 m underground.

A Conceptual Design Study of the Large Hadron Collider (1) was published in October 1995, less than 12 months after the CERN Council approved the project in December 1994. Technical studies suggest that, with adequate funding, the LHC could be built and installed by the year 2004, but the exact time scale will depend on how many non-member countries accept CERN Council's invitation to join the project. If funding is inadequate, a two stage programme with first operation in 2004 at a lower energy and luminosity is envisaged. In that case full performance might not be reached before 2008.

LHC SHIELDING

Shielding requirements for high-energy proton storage rings are normally not been based on the estimates of loss that will occur around the ring under standard operating conditions. As explained above, these losses must be kept to a minimal level for the storage ring with superconducting magnets to work at all. Estimates of shield thicknesses are based more on the potential exposure in the case of an unexpected loss of the circulating beam (or beams) at a single point. The damage caused to the accelerator by such a loss would be dramatic, and every effort will be made to ensure that a full beam loss will not occur. However, although the probability of such an event is extremely small, a full beam loss cannot be excluded from consideration. This shielding philosophy is explored in more detail in (2). The design constraint chosen for the LHC was that the loss of one circulating beam at full intensity should not give rise to an *ambient* dose equivalent of more than 50 mSv at the outer surface of the shield. This is expected in the real exposure situation to lead to an *effective* dose equivalent of less than 20 mSv. This would not then involve any declaration of a radiological incident or accident to a controlling authority and would not jeopardized the future work of the persons involved in radiation areas. Simulation studies suggest that 4 metres of concrete would be needed to shield the accelerator ring to meet this requirement (3). Similar simulation studies have been made for a full beam loss occuring in the ALICE experiment; the results from one of these studies is given in Figure 3 which shows dose equivalent contours in the experimental cavern and in the occupied area in a deep pit above a shield plug. The numbers indicate the dose in mSv from a full loss on one side of the experiment and show that with the shield configuration under study at the time, the dose constraints were not achieved (4). Access to the top of the shield plug must therefore be restricted.

The layout of one of the two high-luminosity experiments, ATLAS, is indicated in Figure 4. In these experiments the main source of radiation to be shielded is given by the inelastic p-p collisions. These provide a source which slowly varies in time according to the luminosity. The experiments themselves provide some shielding around the interaction point. However inside the detector during operation, radiation levels can be high enough to damage the detectors and their associated equipment. Typical fluence levels inside the ATLAS detector are given in Figure 5 for neutrons having energies above 100 keV (5). The shielding provided by the detector is clearly visible, as is the fact that survival of silicon-based semiconductor devices is not at all assured inside the inner detector assemblies. It must be noted that radiation

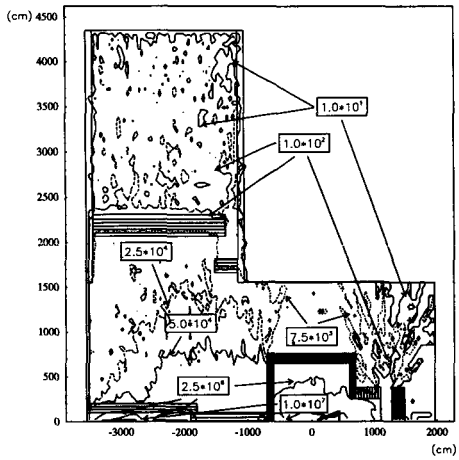

Figure 3: The radiation environment of the ALICE experimental region resulting from a full beam loss. Dose contours are in units of mSv (reproduced from reference (4)).

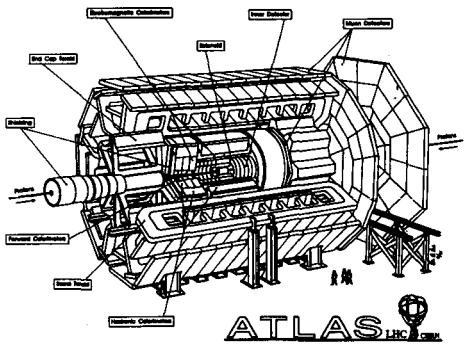

Figure 4: The ATLAS layout.

damage is always of concern to persons concerned with radiological protection at high-energy accelerators since damage goes hand-in-hand with high levels of induced radioactivity. Thus replacement of damaged components can lead to significant exposure of personnel.

In the very-forward regions of the experiment, special shielding-collimators must be placed in front of the first superconducting quandrupoles of the accelerator to avoid excessive energy deposition in the magnets. These collimators act as strong secondary sources of stray radiation and have to be specially shielded to avoid serious background problems in the muon chambers installed around the main detector. The result of the shielding provided around these collimators and by the detectors themselves is that the experimental regions give the false appearance of being only lightly shielded against beam losses in relation to the LHC main ring, with only 2 m as the thickness of the lateral shield around ATLAS and 3 m around CMS.

INDUCED RADIOACTIVITY IN THE ACCELERATOR STRUCTURE

As explained above, a proton accelerator equipped with superconducting bending magnets and quadrupoles such as the LHC has to be protected from beam losses to avoid quenches. The fact that beam losses in the superconducting magnets must be low means that high radiation levels from induced radioactivity cannot occur in the arcs of the LHC. It has been estimated however that the dose rate from induced radioactivity close to the cryostats will be $\lesssim 1\ \mu$Sv/h (6), arising from inelastic interactions of the circulating protons with the residual gases in the vacuum chamber.

It therefore appears to be prudent to expect the need to designate the LHC ring during shut-downs as a Controlled Radiation Area on this consideration alone. This level of dose rate can be considered as a minimum base level. Local areas of high radioactivity will be concentrated in three distinct areas of beam losses, *i.e.* the interaction points of the experiments, the dumps and the collimators.

Figure 6 indicates the dose rates from induced radioactivity that can be expected in the CMS exper-

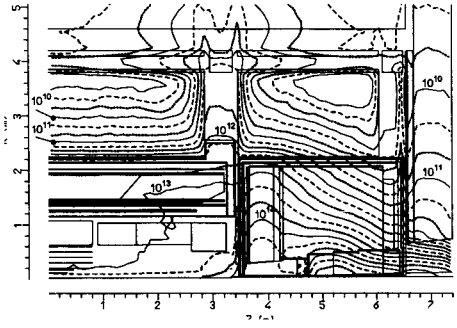

Figure 5: The annual fluence of neutrons (in cm^{-2}) having energies above 100 keV in the inner detector and calorimeter system of ATLAS (reproduced from reference (5)).

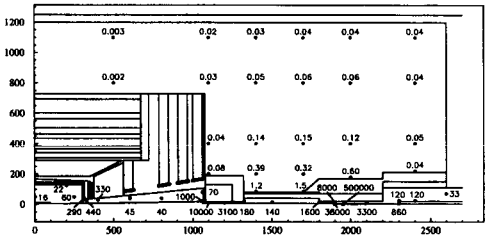

Figure 6: Estimated dose rates from induced radioactivity (in μSv/h) 1 day after the stop of a 60 day operating period for the CMS detector region (reproduced from reference (7)).

imental region (7). It will be seen that the levels outside the detector and the forward shielding are not significant. However maintenance work inside the inner detector region will have to be strictly controlled and special handling procedures will be necessary for work on equipment in the very-forward regions close to the beam-pipe.

The dumps are essential for absorbing the protons when emptying the rings of the circulating beams once these are no longer useful for physics experiments. These dumping operations will take place once or twice a day, and since the dumps will be heavily shielded with iron, the induced activity in the dump caverns will only lead to dose-rates of the order of several tens of μSv/h.

Scraper-collimators are essential for the successful operation of the LHC. They are needed to remove off-momentum protons which otherwise would be lost in sectors of the LHC containing superconducting magnets. A system of well adjusted beam apertures is required around the machine in specific places, limiting the beam size by scraping off the halo that builds up with time due to the proton-proton collisions in the experiments. The cleaning sections installed in octants 3 and 7 will concentrate beam losses and the whole region around these scraper elements will become highly radioactive. Iron shields of approximately 1 metre thickness will be required in some places to reduce dose rates from induced activity to tolerable levels in the passage-way alongside the machine elements.

All items inside the LHC tunnel during operation must be considered as and in most cases are slightly radioactive, and must be treated as such when removed from the machine *e.g.* properly marked. In the case of the LHC, the dispersion of machine and experimental components taken out for repair and maintenance, and moved into non-radioactive areas like normal workshops, must be careful controlled. By no means can such items be eliminated from the site as normal scrap material so long as it has not been shown that the activity contained therein is below the exemptions limits for the specific materials concerned. Due to the sheer number of items that are installed in the machine not all maintenance and repair operations can be closely followed by a radiation protection technician. Maintenance staff for the machine and experiments therefore need careful education in radiation matters.

CONCLUDING REMARKS

It is not possible in such a short review to cover in depth all aspects of the radiological situation of the LHC. These are treated in detail in many laboratory reports which are available to the public. The final design of the accelerator has yet to be determined, but sufficient studies have already been completed to show that it will be possible to operate the LHC well within the limits of radiation exposure for both the personnel of CERN and persons living in the region of the LHC.

ACKNOWLEDGEMENTS

The authors wish to thank G. Chabratova of ALICE, S. Rollet of ATLAS and M. Huhtinen of CMS for permission to reproduce Figures 3, 4, 5 and 6.

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THE ENVIRONMENT

As for other proton accelerators, the topics that are of concern for the environment of the LHC installations are the propagation of prompt radiation (muons and neutron skyshine) and the activation of air, cooling-water, soil/rock and ground-water.

The activation of soil is generally of concern for the environment if some of the radionuclides formed can dissolve in the ground water. Here the particularity of the LHC is that the machine is mostly situated more than 100 m underground in a geological formation of compressed sandstone called molasse that does not contain any mobile water and in particular has no exchange with the groundwater contained in the moraine layers near the surface from which the local drinking water supply is taken.

Thanks to the depth of the implementation of the LHC, radiation levels from prompt radiation will not be observed at the surface. Although strongly directional beams of high-energy muons *e.g.* behind the beam dumps propagate over several kilometers they always remain underground and thus present no radiation hazard for the environment. The underground machine tunnel, the experimental areas and the service galleries are connected to the surface by access shafts of 10–20 metres in diameter. Due to the length of these shafts and the presence of local shielding in the underground areas, or additional top shielding at the top of the shafts, the radiation levels from neutrons at the ground surface-level will always be low and the propagation of these neutrons via skyshine will not lead to radiation levels of any significance.

As far as the release of airborne and liquid radioactivity into the environment is concerned, CERN has fixed annual release limits that have been agreed upon by the Host State authorities. With the advent of new dose factors in the recently published Swiss Ordinance on radiation protection that relate the release of a specific radionuclide to the effective dose of an exposed member of the public the original limits are actually under review at CERN.

The activation of air in high-energy proton accelerators gives rise to short-lived radionuclides such as ^{13}N, ^{15}O and ^{11}C formed by spallation reactions and ^{41}Ar which is formed in air by a thermal neutron reaction, and hence is normally detected only in small quantities at accelerators. In the LHC however particular attention has to be paid to this radionuclide due to the presence of some 100 tonnes of liquid argon in the ATLAS detector where the direct formation of ^{41}Ar is possible. This could escape into the air of the experimental caverns in case of leakage from the argon calorimeters.

Air from the LHC underground areas will be released at four points equally spaced around the ring with distances of about 7 kilometres between them. One of the realistic approaches when calculating the new radionuclide specific release figures will be to assume that the four plumes from the ventilation are not superimposed and thus the hypothetical member of the general public could never receive the dose from the sum of the four releases.

Water that is collected in the drains that run all around under the floor of the LHC tunnel becomes activated due to proton beam losses. There is a particularly abundant water flow in the drains of the part of the machine tunnel which runs through the limestone at the foot of the Jura mountains and which will be activated particularly by the known beam losses in the collimator region of point 3. The water which is pumped to the surface will have an activity that will not allow its immediate release into the environment but requires a delay of several hours to ensure that any activity in this water meets drinking-water standards. The necessary delay is automatically provided by the passage of the water through decantation tanks to remove any suspended solid matter (mainly sand).

Beam-Limiting and Radiation-Limiting Interlocks

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ABSTRACT

This paper reviews several aspects of beam-limiting and radiation-limiting interlocks used for personnel protection at high-intensity accelerators. It is based heavily on the experience at the Los Alamos Neutron Science Center (LANSCE) where instrumentation-based protection is used extensively. Topics include the need for "active" protection systems, system requirements, design criteria, and means of achieving and assessing acceptable reliability. The experience with several specific devices (ion chamber-based beam loss interlock, beam current limiter interlock, and neutron radiation interlock) designed and/or deployed to these requirements and criteria is evaluated.

INTRODUCTION

The title of this paper refers to active, engineered radiation-safety systems consisting of electronic devices which sense errant beam conditions or excessive prompt radiation and then command other devices to automatically limit or shut off the beam current. Such systems are increasingly used at a number of US accelerator Laboratories to mitigate potential prompt radiation accident scenarios. A number of factors motivate this practice including, most importantly, the cost of additional shielding but also the need to protect accelerator components from beam-induced damage and even to prevent potential failure modes of the fixed shielding. This paper will only be concerned with systems intended for personnel protection.

Common situations where instrumentation-based radiation-safety systems become an important option include:1) changes in radiation safety standards, 2) upgrades that significantly increase beam power to an area, and 3) configurations where certain equipment or procedural failures can send high intensity beam to an area that normally receives low intensity beam.

At one time, it was considered adequate to design shielding to meet regulatory dose management limits for source terms corresponding to the highest beam losses expected to be encountered during normal operating conditions. There was less concern with what could happen under abnormal or accident conditions. Indeed, when the assumed source term was a local loss of the order 0.1 - 1% of the maximum normal operating beam power, then the worst case accident, a full-power spill, would not be life threatening even if the accident lasted as long as hour or so. In this case, the shielding would still limit the radiation fields on the outside of the shielding to less than 10 mSv/h (1 rem/h) for a full power spill (assuming that the shielding was designed to give 10 μSv/h for a 0.1% local spill). While radiation fields at this level are a serious matter, they will not lead to a life-threatening dose if the exposure time is an hour or so.

Modern high-powered accelerators and beam lines, such as those that serve the meson factories or spallation neutron sources, are designed and operated with maximum allowable losses of 10^{-6}/m or less over much of their extent. If shielded only for normal or expected losses, the accelerator has an accident potential that is now a much greater since a full power local spill could produce radiation fields of 10Sv/h (1000 rem/h) or greater outside the shielding. Life-threatening doses could be incurred if the spill lasts an hour or so. In these circumstances, instrumentation-based protection systems offer the promise of accident mitigation.

Even if the shielding for the initial accelerator capability was adequate for full-power spills, improvements and upgrades of accelerator beam power are the norm and create the potential for higher radiation fields under abnormal conditions. Shielding retrofits are frequently much more difficult and costly than the same shielding augmentation at the time of initial construction at a "green field" site. Here as well, active systems offer a possible solution.

From the beginning, LAMPF (now LANSCE) used automatic beam-loss limiting instrumentation to protect the accelerator from damage and keep activation low enough for hands-on-maintenance.

The construction of the proton storage ring (PSR) required a high-intensity H^- ion source which was also used to provide highly chopped beam for a fast-neutron facility, WNR. With these new facilities came the capability for new errant beam situations where high intensity beams could appear in beam lines which normally saw much lower beam intensity. The same instrumentation used at LAMPF to limit activation was also used in these new facilities and was thought to be adequate for limiting loss in accident situations as well. Critical examination of these issues lead to the development of a more reliable system of beam limiting devices suitable for personnel protection. Retrofits to the shielding were still implemented where possible in the areas of greatest concern (1).

Each of the situations enumerated earlier have been encountered at LANSCE. Extensive use has been made of improved beam-limiting and radiation-limiting interlocks to reduce the probability of large doses in severe accident scenarios. The first preference was for shielding or physical barriers rather than instrumentation. However, in some situations we found that shielding retrofits sufficient to deal with the accident potential by shielding alone were very expensive or practically impossible to implement without a major rebuild of the facility. For these situations, there was no acceptable alternative other than greater use of active protection systems.

SYSTEM REQUIREMENTS AND DESIGN CRITERIA

Acceptance of beam-limiting and radiation-limiting interlocks for personnel protection, especially as a substitute for additional shielding, depends upon making the case that the interlocks will reliably prevent **any** unacceptably large beam spills. The system requirements and design criteria that further this goal are discussed below.

A. Ultra-High Reliability

A key issue in the acceptance of active safety systems is the achievement and demonstration of ultra-high reliability or equivalently, ultra-low failure rates for these systems to perform their safety function. Consensus on a single, suitable criterion for the reliability required of "active" protection systems has not been achieved. A frequently sought goal is that failure of the overall protection system leading to death or serious injury should be "incredible", which is often taken to be a failure rate of less than once in a million years. Demonstration of such low failure rates by direct observation is obviously out of the question. Rigorous proof through analysis and computation is also elusive. However, probabilistic analysis and assessment does have value as will be discussed later.

In a graded approach, reliability requirements and potential consequences of a system's failure to mitigate worst-case accident scenarios would be correlated such that higher reliability is required for the more serious consequences. Quantification of the relationship is another elusive goal because it is fundamentally a judgment of acceptable risk. In practice, one is driven to provide the most reliable system possible within a budget.

In lieu of numerical reliability specifications which can not be demonstrated by direct measurement, deterministic features or processes can be specified which are known to make positive contributions to reliability and which can be evaluated without much ambiguity. Included would be such features as redundancy, testability, fail-safe characteristics, and rigorous quality standards.

B. Criterion for "Fail-Safe" Operation

Fail-safe operation of the protection systems is a feature often required or sought after for engineered safety systems. In practice, no system can be shown to be truly fail-safe with respect to all possible failure modes. A definition or criterion that appears feasible in practice is to use the term "fail-safe" for systems which fail safe with respect to any single-point failure. At Los Alamos, this concept was pioneered by Andrew Browman and employed in the design of the both the beam loss interlock and beam current limiter interlock which are now part of the radiation security system (RSS). To be more specific, the criterion used was that any single point hardware failure will have one of three outcomes: the device continues to perform its protection function, or the device shuts off the beam and holds it off until the fault/failure is corrected, or the gain changes by less than some specified tolerance (20% was chosen the beam loss interlock) and the device continues to perform its protection function at the changed sensitivity. This definition of the term fail-safe will be used through out the rest of this paper.

Some techniques used to achieve fail-safe design include:

- Use of redundancy within a module,
- Continuous self-checking,
- Continuous monitoring of power supply voltages or other critical parameters.

To follow the fail-safe concept to its logical conclusion, one should also consider single-point human failures in design, construction, testing, operation and maintenance. However, this is rather difficult to do with completeness. Redundancy in the form of independent verification of construction, modifications, maintenance and testing is often used to minimize "single-point" human/administrative failures.

C. Redundancy

Redundancy is used extensively to improve reliability and avoid single-point failures. It primarily improves reliability and availability with respect to uncorrelated failure modes or events. The main weakness is common cause failures which reduce simultaneous multiple failures to an equivalent single point failure mode or event. Some examples of correlated or common cause failures are aging and wear-out phenomena, common environmental changes (such as temperature, humidity, dust, vibration, radiation damage, etc.), lightening induced failures, mechanical crushing from a single event, and design flaws in redundant systems employing identical units. The use of different physical principles or different technologies in the redundant devices is helpful in reducing common cause failure modes.

D. Complete Coverage by Radiation Detector Interlocks

To be most effective, the interlock system should be capable of detecting any unacceptable beam losses no matter where it occurs. In this context, unacceptable beam losses are those that can lead to unacceptable radiation exposure in areas where occupancy is not prevented by physical barriers. Therefore, sufficient devices are deployed in such a way that unacceptable beam loss at any point in the beam acceleration, transport or beam handling systems is detected at the required or specified minimum sensitivity (or at greater sensitivity). Mere sampling is not sufficient.

E. Required Dynamic Range

To provide protection against the full range of possible accident scenarios it is necessary that the overall system have a dynamic range that covers all possibilities from the minimum required sensitivity in normal operations to the maximum possible dose/dose rate that can be delivered under full- power beam spill accident scenarios.

F. Testability/Verifiability

Frequent testing is a well-known technique for improving device availability and one that the system design should facilitate. It should be possible to easily verify or test for proper functioning all essential safety features of the device while it is in place. Where redundancy is used to achieve fail-safe operation with respect to single-point failure modes, it is crucial to find and correct any single-point failures in redundant components shortly after they occur since such units or sub-systems no longer satisfy the fail-safe criteria (at least in the case of two-fold redundancy).

G. Acceptable False-Alarm Rate

Frequent false alarms discredit a device with operators. At some frequency of false alarms the device will be ignored or taken out of service even if the false alarms are fail-safe. In my experience, false alarms at the rate of one/month of operation are tolerable; once per week is not.

H. Isolation and Tamper Resistance

Safety-critical systems should be made resistant to inadvertent or willful tampering with components that are essential to the safety function. One reason for isolation of safety systems from other interlock or instrumentation systems is to avoid inadvertent compromise of the safety features; another is that isolation makes it easier to lock off the safety system components, and junction boxes and wiring plant.

I. Beam Plugs and Reliable Beam Shut Off

The active protection system is only as reliable as its weakest link. Care must be taken to ensure that the detection of an errant beam or excessive radiation condition results in a command that is reliably transmitted to a sound and reliable beam shut off system. Electronic beam deflectors will

quickly shut off the beam, but, they should be backed up by fail-safe beam plugs. These would be beam stoppers that can take the full power of the beam indefinitely or else create a passive shutdown of the accelerator before the beam burns through the plug. The later feature is implemented for the fusible beam plugs used at LANSCE. The fusible beam plug is an idea picked up from SLAC. The face of the beam plug is a fairly thin metal window (the fuse) covering a cavity filled with air and connected by tubes to the atmosphere. If high power beam strikes the window it causes the window to melt and let air into the accelerator before the body of the plug can melt. The air is sufficient to ruin the vacuum and stop acceleration without destroying the accelerator.

J. Quality Control

Rigorous quality assurance and quality control standards for critical safety systems components, wiring plant, design, construction, installation, testing, maintenance, training, certification and documentation are needed to ensure that the design policies, principles and criteria are properly implemented. Quality standards approaching those used for critical safety systems in the nuclear industry are recommended.

SOME EXAMPLES

The concepts and principles discussed above have been applied to the very challenging situation at LANSCE. The aim was to have a redundant, three-layered, "defense in depth" where each layer employed a different technology so as to minimize common mode failures. The first layer was a system of fail-safe beam loss interlocks based on ion chamber detectors placed in the beam tunnels. The second layer employed fail-safe beam current limiters to prevent excess beam currents in critical areas. The third layer consisted of neutron radiation detector interlocks outside the shielding deployed to enforce limits on the maximum neutron radiation levels in occupied areas. The fail-safe ion chambers and the current limiter were developed specifically to meet these criteria for the Line D facilities at LANSCE and have since been used at other LANSCE facilities. The well known "Albatross" was used as the detector for the third layer. While it was modified to satisfy some of the criteria listed earlier, it proved to be not as robust and reliable as the other two.

A. Beam-Loss Interlocks

Ion-chamber based interlocks are used at several leading US accelerator laboratories. At LANSCE, the ion-chamber based, beam-loss interlock is referred to as the errant beam detector. Errant beams are detected by the radiation generated from beam losses somewhere in the vicinity of the detector. The unique features introduced to make it fail-safe and suitable as a safety interlock are discussed here. For details and other features refer to reference (2).

The ion chamber is filled with nitrogen gas at 1 std. atmosphere, so that if it leaks, the pressure will drop to about 0.75 std. atmosphere. (local atmospheric pressure), and remain mostly nitrogen. After a leak it will still function as a satisfactory ion chamber, albeit with a somewhat lower gain but still within the 20% tolerance.

The signal from the ion chamber is converted to a voltage and presented to two redundant processing channels. Fail-safe operation is achieved by self-checking the common portion and by redundancy in the rest of the unit. Continuous self-testing is implemented using a background current generated by a resistor assembly between the HV and signal electrodes. A fault signal is generated if the ion chamber current is not greater than 80% of the design value for the background current. This checks that the chamber HV has not dropped by more than 20% and checks continuity of the signal cable and HV cable. A fault signal due to excessive beam spill is generated when the ion chamber current (background + signal) is greater than the trip set point (background + threshold).

For those components that are not checked by the self-testing feature, redundancy is used to ensure that no single point failure is unsafe. The device has dual-redundant current sensing circuitry except for the analog input from the chamber (which is checked by the self-testing feature). Redundant fault outputs are supplied to the dual-line interlock system backbone to transmit the shut-off command to accelerator shut-off system. The design has been checked carefully to see that the following types of "single point" failures will be fail-safe: an open, short or significant change in value for resistors and capacitors; an open at any junction between two components; a short between

any two junction points; a short between any junction point and ground; a short between any junction point and the power supply rails; and shorts between any number of pins on a single IC. Additional circuitry monitors power supply voltages and generates a fault signal if these are out of tolerance. On a complete loss of power the system will fault both channels.

A full suite of test functions are available via front panel switches. These are designed to test the full functionality of each leg of the redundant processing circuitry and are performed periodically. An overall test with a source placed on the detector is also perform periodically. Construction, bench testing, installation, repair, maintenance, operation and field testing are covered by documented and approved procedures (3-4).

The final level of redundancy within this layer is achieved by the deployment of extra detectors in the beam tunnel. A spill at any point in the tunnel is viewed by at least two detectors at a specified minimum sensitivity.

The response of this detector has been tested with beam over the full range of dose rates and spill conditions that could be encountered in service at LANSCE (5-6). At the very highest levels (~ 3 rads/pulse) with the short PSR pulse (200 nanosecond width), the response is non-linear but still monotonic. These tests did show the need to keep the threshold setting below the maximum available from the front panel adjustment (administrative control) in order to ensure that the device will always trip for any large spill regardless of pulse width and repetition rate. Administrative control of the threshold settings is also needed because the trip levels required for personnel protection vary with location depending on the configuration of the shielding and distance of the detector from the beam line.

We are convinced that this system now meets the criteria for being fail-safe in the limited sense defined in this paper. It was not easy, as is evidenced by that fact that the present version is Model III. Previous versions were found to have subtle flaws; some found by actual failures in the field, others by additional analysis of the design. Some forty units of Model III have been in service since 1992 with no "unsafe" failures i.e. no failures that were not fail-safe.

B. Beam Current Limiter Interlocks

Some beam facilities are designed or operated to take only a small fraction of the beam current that is possible from the accelerator. For these beam areas, errant beam conditions can include an unintended increase in the beam current directed to the facility. Here, a single device that detects excessive current can protect a large area.

A fail-safe beam current limiter based on a beam current transformer has been developed and implemented at LANSCE. The fail-safe feature is achieved by a combination of self-checking and redundancy. Self-checking is implemented using a test winding on the beam transformer toroid. The signal from the toroid is split and the signals sent to dual-redundant processing channels which present the fault status to the corresponding channels of the dual-redundant RSS "backbone". Faults are generated by failure of the self-checking circuitry or by excess beam current. Loss of power will fault both channels of the unit. The power supply voltages are also checked by the self-check circuitry. A calibrate winding is also installed though the toroid. It permits test signals to be injected to verify proper functioning of all critical features including the proper value of the trip point.

A reliability analysis of this device was performed by an experienced team of safety analysts from LANL in collaboration with the designers of the beam current limiter. It was part of a limited-scope, probabilistic safety analysis of selected safety systems at LAMPF (8).. An analysis of the beam current limiter was performed initially using failure modes and effects analysis and criticality tools to examine the system components and potential failure modes. This was followed by a fault tree analysis. The analysis provide estimates of system unavailability (ratio of average downtime to uptime in the time interval between testing) of 3.7×10^{-3} with an estimated error factor of 2.2. Annual testing and operation for half a year per year were assumed. These imply an estimated failure rate of 1.5×10^{-2}/y for unsafe failures.

Experience with the current limiters has been good. Several units have been in service since 1989 and there have been no unsafe failures. Experience has demonstrated its vulnerability to radiation damage. The electronics are placed near the toroid to minimize electrical noise. In one case

a unit failed that was placed close to a beam stop, however, it failed safe and faulted until replaced. In another case, a unit faulted when exposed to a large radiation pulse produced by beam spill from an occasionally misfiring of the PSR extraction kickers. It always failed safe, but the rate of these false alarms was too high and the unit was withdrawn from service at this location.

There is set of circumstances or accident scenarios where the current limiter interlock will not provide the level of protection usually sought. The limiter operates by sensing the algebraic sum of currents passing through the toroid and does not sense neutral beams or equal mixtures of positive and negatively charged beam. This means it will not function as needed in situations where the beam is partially or totally neutralized. At LANSCE, where H^- beam is used, protection can be lost in accident scenarios where the beam is partially stripped either to H^0 or a neutralized mixture of H^-, H^0 and H^+ before passing through the toroid. Such stripping might be the result of poor vacuum or material that partially covers the beam.

C. Area Radiation Interlocks

An interlocked detector suitable for use outside of the shielding is also a challenging problem, particularly if it is to do double duty i.e. monitor routine levels and serve as an interlock to mitigate any and all prompt radiation accident scenarios. The albatross IV neutron detector was chosen at LANSCE for two reasons; it was designed for use at pulsed accelerators and it was already in routine use at this laboratory (8). It was modified to fail safely on power failures and a self-checking feature was added which required that the detector produce a minimum count rate from an internal gamma-ray source.

For a number of reasons, primarily limited resources and the over confidence that was a result of successful experience with it as a survey instrument, the design of this complex unit was not subjected to the same level of scrutiny as the two other instruments described above. In doing so an important limitation to the dynamic range was overlooked, but one that was pointed out in a subsequent safety review. The Geiger tubes in the unit were used in a counting mode and could "lock up" i.e. cease to count at all at high event rates just when the protection is needed most. The cure was to add current mode detection that is activated at high count rates so that the combination of detection modes covers the required dynamic range.

A radiation detector more suited to safety-interlock requirements is still needed. The Albatross was designed as a survey instrument not as an interlock. It is more complicated than desirable for the interlock function. Reliability has not been as high as with the other instruments. Failure rates in the LANSCE environment have been on the high side and there have been four failures that were not fail-safe. The unsafe failures were investigated and modifications made to prevent future occurrences. Reliability has improved since the last modifications. In the past two years, 40 units were in service and there have been no unsafe failures.

It should be noted that complete coverage outside the shielding is more difficult than in the beam tunnel primarily because the radiation distribution (from a local spill) coming through the shielding is more narrowly concentrated around the direction of the ray with the least number of attenuation lengths. Thus, the spacing of detectors needed for full coverage is reduced and the number required increases.

ASSESSMENT OF RELIABILITY

Demonstration of ultra-low failure rates is difficult. Direct measurement of such rates for a complete system is not practical. One has little choice but to rely on some type of analysis. The techniques of probabilistic risk assessment (PRA) used, for example, in the nuclear reactor industry constitute a well-developed methodology appropriate to this problem. The main thrust of this methodology has been applied to parts of the LANSCE radiation security interlock system (7,9). The reliability analysis of the beam current limiter discussed earlier is a good example. The analysis provides more than just an absolute estimate of failures rates. Once the model has been developed, it can be used to isolate the factors that make the greatest contribution to unreliability. One can also study the effect of changes to the design or to specific input parameters. Relative probabilities are often more accurate than absolute probabilities since common factors drop out of the ratio. Such

analyses help to identify the most cost-effective measures for improving reliability, assuming that the costs of various changes can be estimated readily.

The complexity of probabilistic methods used in PRA is another barrier to acceptance of the conclusions. Those who must act on the results or conclusions often do not have the expertise to adequately judge the results. Peer-review of the study by PRA experts who do not have a vested interest in the outcome can help decision makers.

The operating experience with the 3 specific devices discussed above is reaching the stage where it provides useful data for assessment, perhaps even quantitative estimates, of device reliability. For example, over 100 device-years of service have been logged for the loss monitor device with no failures implying an upper bound on the failure rate in the neighborhood of 10^{-2}/y. Combining the device reliability estimates to produce a reliability estimate for the entire protection system is possible if one assumes statistical independence. The present data is inadequate to test that assumption in a straightforward way.

CONCLUSIONS

The use of instrumentation-driven, beam-limiting and radiation-limiting interlocks for radiation protection is on the increase at US accelerator laboratories. Most would agree that well-conceived and well engineered active protection systems can reduce the risks from prompt radiation accidents. The main debate centers over the extent to which these systems can be used to define the safety envelope for accelerator operations or, expressed in trade jargon, the amount of credit to give to these systems towards meeting safety goals. Acceptance of instrumentation-based radiation safety systems ultimately depends on the confidence that is developed in both the systems' reliability and the systems' "completeness". The latter term refers to the system's ability to deal effectively with all possible prompt radiation accident scenarios.

The set of system requirements and design criteria discussed in this paper are aimed at producing a personnel protection system that is both highly reliable and highly complete in its coverage of accident scenarios. In addition, the requirements and design criteria were formulated to make it possible to objectively determine compliance. The experience, to date, with systems designed to these requirements is encouraging. The problem is challenging but solvable.

ACKNOWLEDGEMENTS

Many people have contributed to the development of the concepts and systems discussed here. I want to acknowledge my many colleagues at LANSCE who made important contributions to development of the concepts and did all of the implementation. They include the LANSCE Radiation Safety Committee, now chaired by Olin van Dyck; Andrew Browman, Floyd Gallegos, Rich Ryder, Mike Plum, David Brown, Martha Zumbro. Mohsen Sharirli played a leading role in our effort to apply the methods of PRA to accelerator safety systems.

An important function was served by a workshop on the subject at Los Alamos chaired by Ralph Thomas. Many of the fundamental concepts discussed here were aired at this workshop and were peer-reviewed by a panel of experts from the field who were in attendance. Proceedings are available as an Los Alamos report (10).

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ICRP CONCEPTS AND TASKS IN INTERNAL DOSIMETRY

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INTRODUCTION

The uptake of a radionuclide and its retention with time in an organ or tissue (source organ or tissue) following its ingestion, inhalation, or systemic administration either by injection or through wounds, is described by biokinetic models, which are sufficiently simple to facilitate estimates of organ equivalent doses for radiation protection purposes. For the assessment of *annual limits for intake of radionuclides by workers* (ALI) and of *dose coefficients for intakes of radionuclides by members of the public* and *patients*, the International Commission on Radiological Protection (ICRP) has developed biokinetic models for the uptake of radionuclides by the gastrointestinal tract and the respiratory system, the latter for different ages. For the *systemic behaviour of radionuclides*, i. e. translocation of radionuclides to and between body organs and tissues, specific biokinetic models were adopted by the ICRP. Considering urinary excretion the ICRP has developed a kidney-bladder model to be applied to all substances used for kidney function tests, and to other radionuclides if urinary excretion results in a significant absorbed dose to the bladder wall.

The fraction of energy absorbed in a target organ or tissue from the transformations of a radionuclide in a source organ or tissue is mathematically described by a dosimetric model.

In the Symposium „Internal Dosimetry" the concepts of the ICRP for the assessment of doses to workers, members of the public and patients from radionuclides are described in detail by K. F. Eckerman and M. Roy considering biokinetic and dosimetric models and dosimetry of the respiratory tract, respectively. In addition application of ICRP concepts of internal dosimetry is presented by J. W. Stather for the special case of estimating the dose to the embryo and fetus following acute and chronic intakes of radionuclides by the mother.

The introductory paper describes present and future activities of the ICRP on internal dose assessments.

PRESENT AND FUTURE TASKS OF THE ICRP IN INTERNAL DOSIMETRY

The 1990 recommendations of the ICRP (1) are based on the most recent information from the epidemilogical studies in the populations of Hiroshima and Nagasaki. These recommendations indicate that the risk of radiation-induced cancer exceeds those assumed in the 1977 recommendations (2) by a factor of 3 to 4. Consequently, the primary dose limits for workers and members of the public were correspondingly reduced, i.e. from 50 mSv per year to 100 mSv during 5 years corresponding to an average of 20 mSv per year, and from 5 to 1 mSv per year, respectively. Consequently the devired or secondary limits, i.e. the ALI and the Derived Air Concentration DAC have to be adapted.

Since Committee 2 of the ICRP is responsible for the development of values of secondary limits and thus for the assessment of those data underlying these limits the major future tasks of the Committee is to adjust internal dose calculations to the new recommandations of the ICRP (1) considering organ/tissue and radiation weighting factors, and taking into account recent knowledge on biokinetic behaviour of radionuclides.

DOSES TO MEMBERS OF THE PUBLIC

As an extension of its work for many years in the field of developing secondary limits for the control of intakes of radionuclides by workers, in March 1987 the ICRP established a Task Group of Committee 2 charged with the assessment of internal dose coefficients as a function of the individual's age. The work of the Task Group has been limited to a consideration of those radioisotopes of elements which are expected to be released into the environment due to human activities, and which may have significance for environmental radiation protection purposes.

Age-dependent ingestion dose coefficients for radioisotopes of 29 elements together with preliminary inhalation dose coefficients for radioisotopes of 11 elements, based on the ICRP 30 lung model and the ICRP 26 tissue weighting factors, were published in Publications 56, 67 and 69 (3, 4, 5). Age-dependent inhalation dose coefficients for radioisotopes of the above 29 elements together with those of curium and calcium were calculated by means of the Human Respiratory Tract Model for Radiological Protection (to be described by M. Roy in this symposium) in ICRP Publication 66 (6) for a median aerodynamic diameter of 1 μm and were published in ICRP Publication 71 (7). Compounds, previously assigned to lung clearance Classes D, W, and Y in Publication 30 (8, 9, 10), have been allocated to lung absorption types F, M and S, with f_1 values for material cleared to the GI tract taken from ICRP publications 56, 67 and 69 (3, 4, 5). If additional information on GI absorption was available, then this was applied and detailed in the datasheets.

In addition, Committee 2 is developing a report on doses to the embryo and fetus following acute or chronic intakes by the mother of radioisotopes of the above 29 elements plus curium and calcium (to be described by J. W. Stather in this symposium).

A further activity of Committee 2 is to estimate the reliability of ICRP's dose coefficients for intakes of radionuclides by ingestion and inhalation. The main aim is to develop an approach which should be quantitative, selective, addressing only health individuals of various ages and a limited number of elements. Reliability refers to

- the GI tract model,
- f_1 values for gastrointestinal absorption,
- the respiratory tract model,
- systemic biokinetic models.

In addition, uncertainty concerns the biokinetics of radioactive progeny produced in the body. Variability of doses will be considered quantitatively for specified examples of elements.

DOSES TO WORKERS

The terms of reference of the Task Group of Committee 2 on Internal Dosimetry was extended in 1993 to include a revision of ICRP Publications 30 (8, 9, 10) on Limits for Intakes of Radionuclides by Workers ALI. In order to permit immediate application of the Commission's 1990 Recommendations (1), revised values of the ALI based on the methodology and biokinetic information from ICRP Publication 30, but which incorporated the new dose limits and radiation and tissue weighting factors, were calculated by the Task Group on Dose Calulations and issued as ICRP Publication 61 in 1991 (11).

Since issuing this Publication, ICRP has published the revised kinetic and dosimetric model of the respiratory tract in Publication 66 (6). To give values of inhalation dose coefficients for workers using this new model, both Task Groups an Internal Dosimetry and Dose Calculations of Committee 2 have recalculated the dose coefficients for workers underlying the ALI in

Publication 61 (11). These revised dose coefficients were published as ICRP Publication 68 (12).

In due time, possibly in about 5 years, a complete revision of ICRP Publication 30 (8, 9, 10) will be issued, taking into account new anatomical and physiological data, and newer biokinetic models. It is intended to publish inhalation and ingestion dose coefficients for radioisotopes of elements being relevant for work places, together with bioassay data for monitoring purposes.
The comprehensive revision of Publication 30 will include:

- a review of the choice of lung absorption parameters and default absorption types for inhalation of different chemical forms of the elements found in the workplace;
- a consideration of data on f_1 values for ingested or swallowed chemical forms after inhalation;
- an update of the biokinetic data and models for systemic activity for the elements not covered in Publications 56, 67, 69, and 71 (3, 4, 5, 7);
- provision of models that can be used both for internal dosimetry and for interpretation of bioassay data.

An interim revision of ICRP Publication 54 (13) on „Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation" is inteded to be published early in 1997 to provide data for monitoring that are consistent with ICRP Publication 68 (12). The report will cover selected radioisotopes of the following elements: ^{3}H, Fe, Sr, Ru, I, Cs, Ra, Th, U, Pu, Am, Cu, and Cf. Tabular data will be provided for the excretion over the first 7 days post intake. Equilibrium levels will be given only for long-lived radionuclides.

The Task Group on Dose Calculations is giving consideration to changes in dosimetric models before the revision of Publication 30 (7, 8, 9) can be issued. These include:

- modifications to absorbed fractions of α and β/γ emitters deposited in the skeleton;
- need to allow for bremsstrahlung in the calculation of the Specific Effective Energy;
- use of phantoms based on medical imaging data for dosimetric modelling;
- taking account of the distribution of blood in different organs and tissues;
- consideration of new Reference Man data for the distribution of red bone marrow;
- consideration of females;
- depth doses in walled organs, e.g. the urinary bladder;
- the possibility of a revised model for the GI tract.

For dosimetric purposes parts of Publication 23 (14) on Reference Man will be revised. The report on skeleton was published recently as Publication (15).

The report on anatomical data and gross body composition is under preparation. Further reports will be on

- elemental composition,
- physiological data, and
- gastrointestinal tract.

DOSES TO PATIENTS

In 1988 a Task Group of ICRP Commitee 2 has published a report as Publication 53 (16) on Absorbed Doses to Patients from Radiopharmaceuticals to give absorbed and effective

doses for 113 radiopharmaceuticals labelled with 74 radioisotopes of 36 elements, including the range of variation to be expected in pathological states, for adults, children, infants and the fetus. This data is continuously updated by a Task Group of Committees 2 and 3 and published as addenda to the Annals of the ICRP (17), and can be used also as a measure for the dose from occupational exposure of personal engaged in the manufacture or use of radiopharmaceuticals.

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DOSIMETRY OF THE RESPIRATORY TRACT

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INTRODUCTION

A new dosimetric model of the human respiratory tract has been recently recommended by the International Commission on Radiological Protection, in ICRP Publication 66 (1). This model was intended to update the previous lung model of the Task Group on Lung Dynamics (2) that was adopted by ICRP in Publication 30 (3). With this aim, extensive reviews of the available knowledge were made for anatomy and physiology of the respiratory tract and for deposition, clearance and biological effects of inhaled radionuclides. Finally, expanded dosimetry requirements resulted in a widely different approach from the former model. The main features of the new model are the followings: instead of calculating the average dose to the total mass of blood filled lung, the model takes account of differences in radiosensitivity of the various respiratory tract tissues. It applies not only to adult workers but also to all members of the population, and provides reference values for children aged 3 months, 1, 5, 10, and 15 years, and adults. Deposition modelling of airborne gases and aerosols associates age dependent breathing rates, airway dimensions and physical activity, to particle size, density and chemical form of inhaled material. Clearance results of competition between mechanical transport clearance and absorption to blood. At each step of the calculation, adjustment guidance is provided to account for use of exact values of particle sizes and specific dissolution rates of inhaled material in order to calculate their own parameter of retention in the airways, and to assess accurately doses to the respiratory tract. Possible influence of smoking, of respiratory tract diseases and of eventual exposure to airborne toxicants is also addressed.

MORPHOMETRY

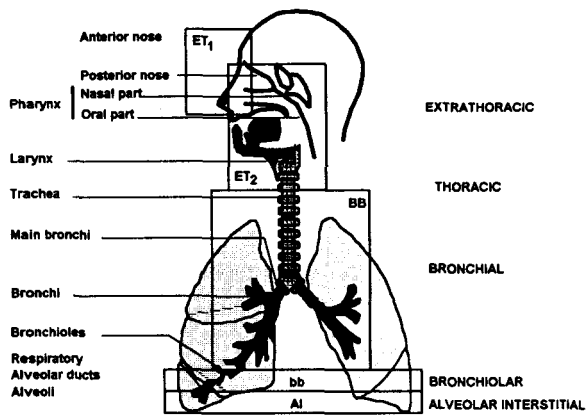

Figure 1: Respiratory tract regions of the model.

The dosimetric model considers the respiratory tract as four anatomical regions (**Figure 1**):

- **ET**, Extrathoracic region comprising the anterior nose, $\mathbf{ET_1}$ and the posterior nasal passages, larynx, pharynx, and mouth, $\mathbf{ET_2}$;
- **BB**, the bronchial region, consisting of the trachea and bronchi from which deposited material is cleared by ciliary action;
- **bb**, the bronchiolar region, consisting of the bronchioles and the terminal bronchioles;
- **AI**, the alveolar-interstitial region, consisting of the respiratory bronchioles, the alveoli and the interstitial connective tissue.

All four regions contain lymphatic tissue and lymph nodes, LN_{ET} that drain the extrathoracic region, and LN_{Th} that drain the thoracic region.

The cytology, histology and morphometric structure are modelled for each region. Reference values for dimensions important to dose calculation are specified, such as airway dimensions, lengths and diameters, and masses of target organs.

TISSUES AND CELLS AT RISK

Information about effects of radiation on the respiratory tract were reviewed from animal studies and data from irradiated human populations. The most sensitive tissues and cells to radiation-induced cancer were identified. The target tissues in each region are located at different depth from the airway surface:

- in ET, the average depth of the target basal cell nuclei is 40-50 micrometers (μm).
- in BB, the average depth of the target secretory and basal cell nuclei are 10-40 and 35-50 μm, respectively.
- in bb, the average depth of the target secretory cell nuclei are 4-12 μm.
- in AI, where the target Clara cells and type II alveolar cells are covering the alveolar surface, the average dose is calculated to the whole region.

To take account of differences in sensitivity between tissues, each region is given a risk apportionment factor that represents the contribution from each region towards the total detriment associated with lung.
Although it is generally accepted that the regions of the respiratory tract differ greatly in radiosensitivity, and in their contribution to the total radiation detriment, precise data on regional sensitivity is lacking. Therefore, the weighting factor of 0.12, given by ICRP 60 (4) is divided equally among the bronchial, bronchiolar and alveolar-interstitial regions. In the extra-thoracic region, clinical observations suggest that the risk is to the posterior nose, larynx and pharynx. **(Table I)**

Table I: Detriment apportionment factors in the airways.

Tissue	Lung W_T ICPR 60	Apportionment factors ICPR 66
Extra-thoracic region	**0,025**	
ET_1 Anterior nose		0,001
ET_2 Posterior nasal passages pharynx, larynx mouth		0,998
LN_{ET} Lymphatics		0,001
Thoracic region	**0,12**	
BB Bronchial		0,333
bb Bronchiolar, G9-G16		0,333
AI Alveolar, interstitial		0,333
LN_{TH} lymphatics		0,001

PHYSIOLOGY

Aerosol inhalability, partitioning of airflow between nose and mouth as well as breathing rates determine the amount of radionuclides inhaled. Therefore comprehensive reference values for volumes and frequencies are recommended for adults of both genders and children aged 3 months, 1, 5, 10, and 15 years, for various levels of activity: resting, sitting, light and heavy exercise. Information is also provided to derive appropriate values for various ethnic groups. Combined with habit data of physical activity, they allow to give reference volumes inhaled per day, for reference age groups of a population.

DEPOSITION

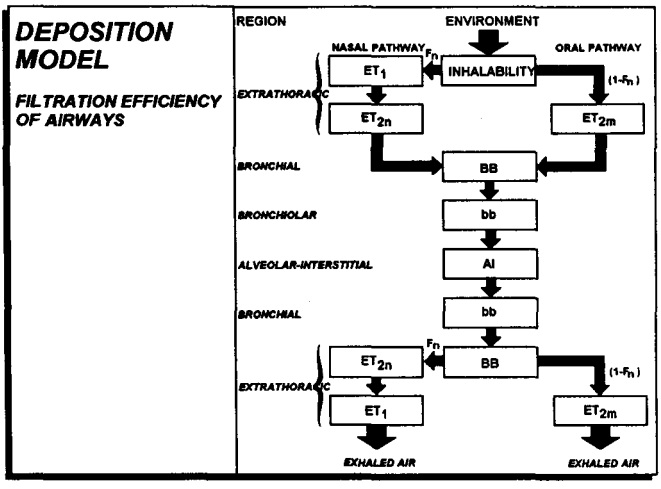

Figure 2: Deposition model: Fn = fractional airflow through nose; 1-Fn = fractional airflow through mouth.

The deposition model describes the initial process that determine the fractions of the inhaled material remaining in the four regions of the airways after expiration. (**Figure 2**). Deposition efficiencies are calculated from human measurements in relationship with particle sizes and airflows. In order to predict deposition in women and children these values are scaled by airway dimensions and breathing rates. The regions are treated as a series of filters. Their efficiencies are assessed on aerodynamic processes such as gravitational settling and inertial impaction, and the thermodynamic process of diffusion, acting during both inspiration and expiration. Deposition fractions are calculated for all possible aerosols (0.0006-100μm) assuming log/normal particle size distributions. Corresponding curves for each population age group are also given as functions of the activity median thermodynamic diameter, **AMTD**, and of the activity median aerodynamic diameter, **AMAD.** Guidance is given to take account of the growth of hygroscopic particles in the airways during the respiratory cycle.

In the absence of information on the particle sizes, deposited fractions are calculated for default values. AMAD of 5μm for workers and 1μm for the public are recommended, with reference to published experimental data in the relevant environments. Whenever particle sizes are readily measured, input of this data is recommended.

A material specific model is derived for gases and vapours. Inhaled gas molecules reaching the airways return to the expired air unless they dissolve in or react with the surface lining. Information on these solubility and reactivity properties are obtained from experimental data.
As default approach the model assigns gases and vapours to three classes:

- **SR-0**. Insoluble and non-reactive: negligible deposition in the respiratory tract. In many cases dose is from the filling air irradiation to the airways.
- **SR-1**. Soluble or reactive: deposition may occur throughout the respiratory tract; in the absence of other information 100% deposition is assumed with the following distribution: 10% ET1, 20% ET2, 10% bb, 20% BB, and 40% AI.
- **SR-2**. Highly soluble or reactive: total deposition and instantaneous translocation to blood in the extrathoracic region.

CLEARANCE

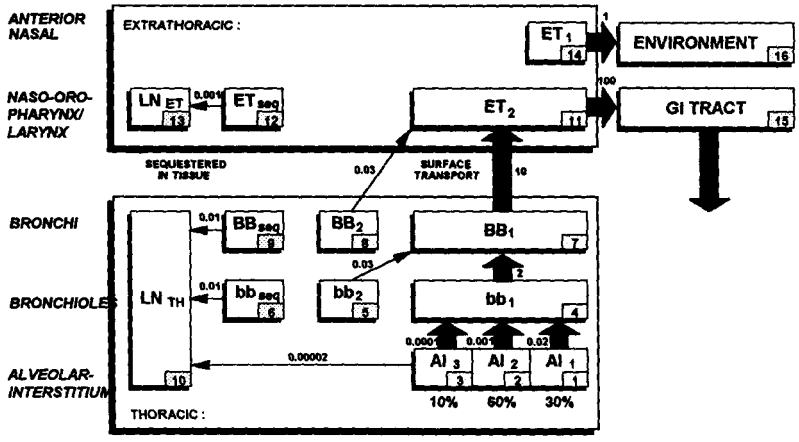

Figure 3: Particle transport model. Rate constants are in d^{-1}.

In each region, after deposition, clearance rates will change with time, and are represented in the model by a combination of compartments. Each one clears at a specific constant rate, independent of age and sex.
Material deposited in ET1 is removed by extrinsic means such as nose blowing but in other regions clearance is competitive between two complementary processes: **particle transport** towards the gastro-intestinal tract and lymph nodes, and **absorption** into body fluids.

Particle transport rates are assumed to be the same for all deposited aerosols and the values of rate constants were derived from experimental studies, so far as possible from human data. An ideally insoluble material deposited in the airways would be cleared according to the model described in **Figure 3.**

Absorption into body fluids depends on the physical and chemical form of the deposited material. It is assumed to occur at the same rate in all regions, except ET1, where none occurs. Absorption into blood is generally a two stage process. (**Figure 4**)

- dissociation of the particles into material that can be absorbed into blood,
- dissolution, then absorption into body fluids of soluble material and of material dissolved from particles.

Rates of absorption observed for compounds for which reliable human or animal data exist are preferably used in the model. Otherwise, default values of absorption rates are specified according to whether the absorption rate is considered to be fast (Type F), moderate (M) or slow (S).

These absorption rates as approximate half-times are:

- 10 minutes for 100% of F material,
- 10 minutes for 10% and 140 days for the remaining 90% of M material, and
- 10 minutes for 0.1% and 7000 days for the remaining 99.9% of S material. (It has been observed that even the most insoluble material show an early absorption of a small fraction of the amount deposited). **(Table II)**

Table II: Default absorption parameters for Type **F, M** and S material

Type of absorption	F (fast)	M (moderate)	S (slow)
Fraction dissolved rapidly, f_r	1	0.1	0.001
Rapid dissolution rate, s_r, d^{-1}	100	100	100
Slow dissolution rate, s_s, d^{-1}		0.005	0.0001

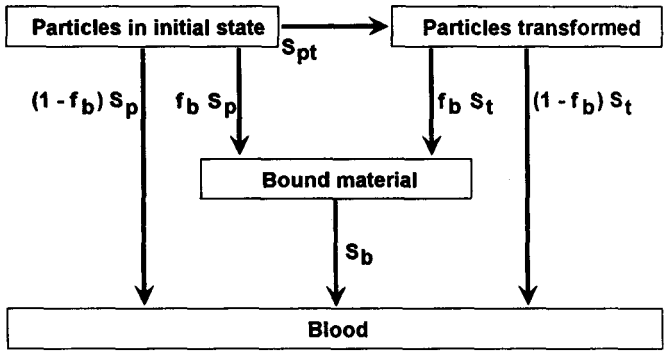

Figure 4: Airway to blood absorption model:

Sp	= rate of dissolution in initial state;
Spt	= rate of transformation;
St	= rate of dissolution in transformed state;
fb	= fraction retained in bound state;
Sb	= rate of dissolution in bound state.

CALCULATION OF TISSUE AVERAGE DOSE

Doses are calculated in accordance with the ICRP 30 approach, in which energy absorbed is averaged over the mass of the target tissue in each organ. The results are then summed up for each region. The committed equivalent dose is calculated over an average working life of 50 years, for workers, and life span of 70 years for persons of the public. These regional equivalent doses are weighted for relative radiation sensitivity using the above defined apportionment factors. The adjusted values are summed separately for the whole extrathoracic and thoracic regions. To these two equivalent tissue doses the ICRP 60 weighting factors are then applied to calculate their contribution to effective dose:

- 0.12 to the thoracic region formerly specified for lungs, and
- 0.025 to the extra-thoracic region, half of the remainders', when it exceeds the highest dose in any of the twelve specified organs.

For inhalation, the age-dependency of dosimetry relates mainly to the smaller sizes of airways and to specific values of deposition. As clearance is not precisely known in children, adult parameters are used as default for particle transport and absorption to blood through airways. Finally, breathing rates being smaller in children, smaller intakes balance higher doses per unit of intake, and the resulting effective doses are close to the adults' for the same exposition.

Effective Doses per Unit of Intake, (Sv. Bq^{-1}) have been calculated with this model for inhaled radionuclides that are most important to Radioprotection. They are now given in Publication 68, (5) for workers and in Publication 71, (6) for members of the public. Compared with doses previously given by ICRP Publication 30, for workers, the data obtained with the new model tend to be:

- lower for soluble alpha and beta emitters,
- similar for moderately soluble material, except for long-lived transuranic isotopes where the recent calculations give twofold lower doses,
- lower for the insoluble forms of most of the radionuclides.

The fact that Publication 68 gives lower doses than those given by Publication 30 is mainly related to deposition that was overestimated by ICRP 30 in the thoracic regions, and in the case of former class D, to slower absorption to blood than in the recently reviewed Type F.

Examples of comparison of Annual Limits of Intakes, calculated for workers according to both publications, are presented in **Table III**. In the case of Publication 30, the primary limit of annual effective dose is 50 mSv,

whereas in Publication 68 it is 20 mSv. Particle size defaults are: 1 μm AMAD, in Publication 30 and 5 μm AMAD in Publication 68, (ALI for 1 μm AMAD are also given to allow for exact comparison).

Table III: Annual Limits for Intakes of some radionuclides inhaled by workers

	CIPR 30		CIPR 68 (1 μm)		CIPR 68 (5 μm)
^{60}Co	W	6×10^6	M	2.1×10^6	2.8×10^8
	Y	1×10^6	S	0.7×10^6	1.2×10^6
^{131}I	D	2×10^6	F	2.6×10^6	1.8×10^6
^{137}Cs	D	6×10^6	F	4.2×10^6	3.0×10^6
^{235}U	D	5×10^4	F	3.9×10^4	3.3×10^4
	W	3×10^4	M	7.1×10^3	1.1×10^4
	Y	2×10^3	S	2.6×10^3	3.3×10^3
^{237}Np	W	200	M	950	1.3×10^3
^{239}Pu	W	200	M	425	625
	Y	500	S	1330	2410
^{241}Am	W	200	M	510	740
E*	**50 mSv**		**20 mSv**		

* = Annual limit of Effective dose

CONCLUSION

The new Human Respiratory Tract Model for Radiological Protection provides flexibility to calculate doses for a wide range of exposure conditions and for specific individuals. This flexibility also allows for revision of reference parameters as new information becomes available without changing the general model characteristics. The wide availability of personal computers allows the user to go into detailed dose calculations and to modify standards values. Recent softwares have been developed for this purpose.

Acknowledgments:
Members of the ICRP Committee 2 Task Group were: W.J.Bair (Chairman), M.R.Bailey, F.T.Cross, R.G.Cuddihy, P.Gehr, A.C.James, R.Masse, M.Roy and W.Stahlhofen.

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RADIATION DOSES TO THE EMBRYO AND FETUS FOLLOWING INTAKES OF RADIONUCLIDES BY THE MOTHER

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ABSTRACT

In 1987 the International Commission on Radiological Protection set up a Task Group of Committee 2 charged with the responsibility for calculating radiation doses from incorporated radionuclides for all age groups in the population. This includes the development of models for calculating doses to the embryo and fetus following intakes of radionuclides by the mother.

The development of models for calculating doses to the embryo and fetus is complex. Particular problems that have had to be addressed are the limited amount of human data available and the consequent need to use both the results of animal studies and chemical analogies; the varying rate of tissue and organ development in different species; the lack of detailed information on the distribution and retention of radionuclides in tissues of the embryo and fetus following intakes by the mother, either before or during gestation; and the radiation sensitivity of tissues of the embryo and fetus.

In the development of dosimetric models for specific elements, human data have been used as far as is possible. Where this has not been available a generic modelling approach has been adopted. The models are being used to calculate doses to both mother and offspring for acute and chronic intakes, both before conception and at various times during gestation. Committed doses are being calculated as well as doses to birth. The results of preliminary dose calculations are considered.

RADIATION DOSES TO THE EMBRYO AND FETUS FOLLOWING INTAKES OF RADIONUCLIDES BY THE MOTHER

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INTRODUCTION

The need for the assessment of radiation doses to all age groups in the population from both naturally occurring and artificial radionuclides present in the environment is increasingly being recognised by national and international organisations. Whilst dosimetric models for infants and children can generally be based on those developed for adults, with appropriate modifications to biokinetic parameters[1], in the case of exposures *in utero* fundamentally new models need to be developed. Such models are also important for assessing doses to the embryo and fetus for women who may be either occupationally exposed or given radio-labelled drugs for clinical reasons. In the most recent recommendations of the International Commission on Radiological Protection, ICRP[2] particular emphasis is placed on measures to control exposures *in utero* for women who are occupationally exposed. This arises because of the recognition of the greater sensitivity of developing tissues to ionising radiation[3,4] and the need to treat the fetus broadly as a member of the public.

There are at present no generally accepted methods for calculating radiation doses to the developing embryo and fetus following intakes of radionuclides by the mother. A number of laboratories have developed models for a few radionuclides[5-7] and recently the United States Nuclear Regulatory Commission issued a report describing approaches to calculating doses to the embryo and fetus from internally incorporated radionuclides[8]. Ideally, dosimetric models should be based on human data but information is available for radioisotopes of only a few elements and for most the results of animal experiments must be used, although even here data are frequently very limited. Models may also be based on chemical analogies.

An ICRP Task Group on Internal Dosimetry (INDOS) is developing models that can be used for calculating doses to the embryo and fetus following intakes of radionuclides by the mother. Specific information will be used, as far as is possible. When appropriate data are not available, a generic modelling approach has been adopted.

STAGES OF DEVELOPMENT

When calculating doses from incorporated radionuclides the different stages in development of the conceptus need to be taken into account.

Pre-implantation

In most mammalian species, the ovum passes from the oviduct into the uterine lumen a few days after fertilization and implants in the mucosa about 2 to 3 days later, ie at about 6 to 7 days after fertilization. At implantation the developing conceptus (blastocyst) weighs less than 10 μg and consists of an outer layer of trophectoderm cells, which will give rise to the bulk of the placenta and the inner cell mass from which the embryo and the fetal membranes will form. It is thought to be nourished by the transfer of low molecular weight substances in uterine secretions through the zona pellucida[9]. The timing of this initial period is very similar in most mammalian species (Table 1).

The Embryo

After embedding in the epithelial lining of the uterus the implanted cell mass becomes closely surrounded by maternal tissue, the progressive erosion of this tissue constituting a source of nourishment. The period of organ formation may be considered to last up to about the end of the second month, at which time the developing embryo

still weighs less than 10 g[10]. Because of the close apposition between the embryo and the uterus wall, it has been commonly assumed, in the absence of more specific information, that the dose to the embryo up to the end of the second month of gestation can be approximated by the dose to the uterus. While this may be a reasonable approximation for many ß/γ emitters it is not necessarily the case for alpha emitters which have a short path length (a few tens of μm) in tissue.

The Fetus

From the end of organogenesis (taken to be 60 days post conception in man, Table 1) progressive formation of new tissue (histogenesis) and growth are the characteristic processes that take place during the fetal period. There are, however, significant differences between species in this latter stage of pregnancy. The human fetus, as with most other mammalian species, is contained in the amniotic fluid where it is surrounded by fetal membranes and nourished via the placenta through the umbilical cord. The newborn child at birth weighs about 3.5 kg. The mass of the conceptus has thus increased about five-hundred fold from the end of the embryonic period to birth and by more than a factor of 10^7 since conception.

The Newborn Child

The amount of radionuclide present in the newborn child also needs to be estimated. The calculated body content at birth may then be used to calculate the committed dose to age 70 years using biokinetic models for infants and children given in ICRP Publications 56, 67 and 69[1,11,12].

DEVELOPMENT OF BIOKINETIC AND DOSIMETRIC MODELS

Transfer of Radionuclides to the Fetus

Radioactive materials have to cross a number of membrane barriers to reach the tissues of the embryo and fetus from the maternal circulation. The processes involved in this transfer include simple diffusion, facilitated transport, active transport, movement through pores and channels, and pinocytosis[13-16]. Radionuclides which are isotopes of stable elements that are required by the developing embryo and fetus will follow the normal pathways for that element (eg, Na, K, Ca). Isotopes of elements that have similar chemical properties to essential elements are likely to follow similar pathways, although the rates of transfer may differ (eg, Cs behaves similarly to K, Sr like Ca). For other radionuclides the rates of transfer to the embryo and fetus will depend upon their chemical affinity for different transport systems in the body and, as a result, the extent of uptake is unpredictable.

Human data: Most available human data on placental transfer of radionuclides are available from studies with labelled metabolites or pharmaceuticals although some data are available for radionuclides in weapons fallout (eg, Sr, I, Cs). Analyses of autopsy samples have given information on both naturally occurring and artificially produced radionuclides and some comparative information has been obtained through *in vitro* perfusion studies using human placentas.

The rather limited amount of human data makes it essential to use the results of animal studies in the development of dosimetric models for man, although even here information is frequently very limited.

Animal data: In the extrapolation of animal data to man, care is needed; particular problems in the development of dosimetric models for the embryo and fetus include: the varying progress of organ development in different species; the presence of several types of placenta in different species, all of which provide a selective but potentially different barrier between maternal and fetal blood; the rapidity of growth; the complex pattern of growth and differentiation with the potential for quite different distributions of radionuclides in the embryo and fetus from that in the mother; and uncertainty about the location of sensitive cells at various stages of development. The pattern of results obtained experimentally for most radionuclides is similar, although overall transfer to the newborn is very variable. In general, concentrations of radionuclides at birth are highest following intakes towards the end of pregnancy although the concentrations are less than in normal tissues. For some bone-seeking radionuclides (eg, Sr) administered at the end of gestation, however, the concentration may be higher in the newborn than the mother, reflecting the demand of the rapidly developing skeleton for calcium.

Dosimetric models: In the development of dosimetric models information is ideally needed on the initial uptake of the radionuclide by the fetal and maternal tissues following its entry into maternal blood and on its subsequent distribution and retention. Data are required on the extent to which activity deposited in maternal tissues is subsequently translocated to the fetus[7] and doses to the fetus from maternal deposits of radionuclides also need to be assessed.

Where sufficient information is available on the uptake and distribution of radionuclides in the embryo and fetus then elemental specific models are being developed. This applies for example to tritiated water and to isotopes of Fe, Sr, I and Cs. For most other elements animal studies provide the basis for the models. In practice, for many elements studies with experimental animals may only give data on relative concentration ratios in the fetal and maternal tissues (C_F/C_M) and details of the distribution of radionuclide in fetal tissues are often not given. In these circumstances, a general approach has been adopted for calculating the dose to the developing embryo and fetus.

For the embryo (to 60 days after conception), the dose is taken to be the same as that to the uterus; on this basis all tissues of the embryo will receive the same dose. For calculating the dose to the tissues of the fetus for non-penetrating radiation (alpha and beta emissions and photons with E <10 keV), experimentally determined C_F/C_M ratios obtained shortly after administration of radionuclides to the mother are used. Multiplying this ratio by the dose to maternal tissues gives an estimated dose to the fetal tissue, on the assumption that the distribution of activity in the fetus is the same as that in the mother. This approach is expected to be conservative in most instances as rapid growth of the fetus will reduce the concentration of radionuclides in fetal tissue, although this dilution effect may be to some extent offset by transfer to the fetus from radionuclide deposits in the maternal tissues. A number of radionuclides would be expected to equilibrate rapidly between maternal and fetal tissues. These include isotopes of the alkali metals, which are predominantly ionic in body fluids and have a rapid turnover in tissues, as well as many tritium and ^{14}C labelled compounds.

For the majority of radionuclides, penetrating radiation from activity deposited in maternal tissues will also contribute to the dose to the fetus. This dose is presently taken to be the same as the dose to the uterus. It is expected that in final calculations doses will be based on models for a pregnant woman.

As indicated above, for many animal studies the distribution of radionuclide in fetal tissues is not reported and the assumption has therefore to be made for calculating doses that the distribution in the fetus is the same as that in the mother. Recent autoradiographic studies have, however, shown that following administration to the mother early in gestation there can be a significant uptake of some radionuclides in the yolk sac membranes (eg, ^{95}Nb, ^{106}Ru, ^{210}Po and ^{239}Pu). This may be important as haemopoietic development involves the migration of stem cells and it appears likely that the primitive stem cell population may arise in the yolk sac, or even at the earlier egg cylinder stage. These stem cells may subsequently migrate to the liver once the extra embryonic and embryonic circulations become connected. Ultimately haemopoietic stem cells pass to the bone marrow. It has therefore been proposed that for calculating doses to the haemopoietic stem cells in the developing fetus, the calculated dose should reflect this migration and the potential of these cells to receive a radiation dose at different sites in the body (yolk sac, liver and bone marrow), taking into account the appropriate period of gestation in the human.

DOSE COEFFICIENTS FOR THE EMBRYO AND FETUS

For the calculation of dose coefficients a range of intake regimens by the mother have been adopted that should allow doses to the embryo and fetus to be determined for any pattern of exposure of the mother. For acute exposures, intakes of radionuclides by ingestion or inhalation (for aerosols with activity median aerodynamic diameters of 1 μm and 5 μm) are taken to occur at the start of weeks 1, 5, 10, 15, 25 and 35 of pregnancy and at 6 months and 2½ years before conception. For chronic exposures, intakes are taken to occur during the year of pregnancy starting from the time of conception, and for 1 year and 5 years before conception. It is proposed that in the final report equivalent doses to the date of birth will be given for the embryo and fetus for all tissues with a specific tissue weighting factor in adults as well as the average dose to remainder tissues (calculated as the mass weighted average). The effective dose will also be given using the w_Ts recommended by ICRP in Publication 60[2]. Whilst these values are not strictly appropriate for exposures *in utero*, they have been used as no alternative tissue weighting factors are presently available. The effective dose provides a useful quantity for assessing the consequences of different patterns of intake of radionuclides and for comparison with maternal doses. Effective doses (to age 70 years) received after birth will also be calculated, together with the total effective dose (before and after birth) received by the offspring and its ratio to the total committed effective dose received by the mother. It

is intended that the dose coefficients will be given for all elements and radionuclides in Publications 56, 67, 69 and 71[1,11,12,17]. The results of some preliminary calculations are given in Tables 2 and 3.

In general, the results of preliminary calculations indicate that for many radionuclides doses to the offspring will be less than those to the mother. For isotopes of caesium and other radionuclides that equilibrate rapidly in the body, the dose to the newborn child is expected to be similar to that of the mother, although the total committed dose may be much less (Table 1). For acute intakes of ^{131}I late in pregnancy the dose to the newborn child may approach twice that of the mother. Similarly, intakes of alkaline earth radionuclides towards the end of pregnancy can result in higher doses to fetal tissues at birth than to maternal tissues, although the total committed dose to the offspring will be less. There is little suggestion that doses to the child will be appreciably higher than maternal doses, particularly for chronic intakes by the mother either before or during pregnancy.

It is expected that the report will be ready for publication in about 18 months time.

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TABLE 1

Approximate time in days of the beginning and end of the major developmental periods in some mammalian species

Species	Pre-implantation	Major organogenesis	Post Conception Fetal period
Hamster	0-5	6-12	13-16.5
Mouse	0-5	6-13	14-19.5
Rat	0-7	8-15	16-21.5
Rabbit	0-5	6-15	16-31.5
Guinea-pig	0-8	9-25	26-63
Dog	0-17	18-30	31-63
Man	0-8	9-60	60-270

UNSCEAR (1977)[18]

TABLE 2

Comparison of effective doses (Sv) to offspring and mothers for 1 Bq acute intake by ingestion at start of week 10 of gestation

Nuclide	Offspring			CED to mother	Offspring/mother
	in utero	Birth to 70 y	Total		
HTO	$2.5\ 10^{-11}$	$6.9\ 10^{-15}$	$2.5\ 10^{-11}$	$1.8\ 10^{-11}$	1.4
^{106}Ru	$2.8\ 10^{-10}$	$2.5\ 10^{-11}$	$3.0\ 10^{-10}$	$1.0\ 10^{-8}$	0.03
^{131}I	$4.0\ 10^{-8}$	$3.2\ 10^{-17}$	$4.0\ 10^{-8}$	$2.2\ 10^{-8}$	1.8
^{137}Cs	$1.0\ 10^{-8}$	$5.0\ 10^{-10}$	$1.1\ 10^{-8}$	$1.3\ 10^{-8}$	0.80
^{239}Pu	$6.6\ 10^{-10}$	$4.2\ 10^{-9}$	$4.9\ 10^{-9}$	$2.5\ 10^{-7}$	0.02

TABLE 3

Comparison of effective doses (Sv) to offspring and mothers to birth for 1 Bq acute intake by inhalation[a] at various times before and after conception

Nuclide	Time of intake (wks)				
	-26	1	5	25	35
	Fetus/Mother				
HTO (V)[b]	0.1	1.1	1.1	1.4	1.4
^{106}Ru (M)	0.01	0.03	0.02	0.01	<0.01
^{131}I (F)	<0.01	0.01	0.15	1.9	1.9
^{137}Cs (M)	0.28	0.14	0.14	0.09	0.06
^{239}Pu (M)	<0.01	0.02	0.02	<0.01	<0.01

Notes:

[a] Activity Median Aerodynamic Diameter = 1 μm.
[b] Default lung Type from ICRP Publication 71[(17)].
V - Vapour, F - Fast clearance rate, M - Moderate clearance rate.

RADIATION DOSES TO THE EMBRYO AND FETUS FOLLOWING INTAKES OF RADIONUCLIDES BY THE MOTHER

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INTRODUCTION

The need for the assessment of radiation doses to all age groups in the population from both naturally occurring and artificial radionuclides present in the environment is increasingly being recognised by national and international organisations. Whilst dosimetric models for infants and children can generally be based on those developed for adults, with appropriate modifications to biokinetic parameters[1], in the case of exposures *in utero* fundamentally new models need to be developed. Such models are also important for assessing doses to the embryo and fetus for women who may be either occupationally exposed or given radio-labelled drugs for clinical reasons. In the most recent recommendations of the International Commission on Radiological Protection, ICRP[2] particular emphasis is placed on measures to control exposures *in utero* for women who are occupationally exposed. This arises because of the recognition of the greater sensitivity of developing tissues to ionising radiation[3,4] and the need to treat the fetus broadly as a member of the public.

There are at present no generally accepted methods for calculating radiation doses to the developing embryo and fetus following intakes of radionuclides by the mother. A number of laboratories have developed models for a few radionuclides[5-7] and recently the United States Nuclear Regulatory Commission issued a report describing approaches to calculating doses to the embryo and fetus from internally incorporated radionuclides[8]. Ideally, dosimetric models should be based on human data but information is available for radioisotopes of only a few elements and for most the results of animal experiments must be used, although even here data are frequently very limited. Models may also be based on chemical analogies.

An ICRP Task Group on Internal Dosimetry (INDOS) is developing models that can be used for calculating doses to the embryo and fetus following intakes of radionuclides by the mother. Specific information will be used, as far as is possible. When appropriate data are not available, a generic modelling approach has been adopted.

STAGES OF DEVELOPMENT

When calculating doses from incorporated radionuclides the different stages in development of the conceptus need to be taken into account.

Pre-implantation

In most mammalian species, the ovum passes from the oviduct into the uterine lumen a few days after fertilization and implants in the mucosa about 2 to 3 days later, ie at about 6 to 7 days after fertilization. At implantation the developing conceptus (blastocyst) weighs less than 10 μg and consists of an outer layer of trophectoderm cells, which will give rise to the bulk of the placenta and the inner cell mass from which the embryo and the fetal membranes will form. It is thought to be nourished by the transfer of low molecular weight substances in uterine secretions through the zona pellucida[9]. The timing of this initial period is very similar in most mammalian species (Table 1).

The Embryo

After embedding in the epithelial lining of the uterus the implanted cell mass becomes closely surrounded by maternal tissue, the progressive erosion of this tissue constituting a source of nourishment. The period of organ formation may be considered to last up to about the end of the second month, at which time the developing embryo

still weighs less than 10 g[10]. Because of the close apposition between the embryo and the uterus wall, it has been commonly assumed, in the absence of more specific information, that the dose to the embryo up to the end of the second month of gestation can be approximated by the dose to the uterus. While this may be a reasonable approximation for many ß/γ emitters it is not necessarily the case for alpha emitters which have a short path length (a few tens of μm) in tissue.

The Fetus

From the end of organogenesis (taken to be 60 days post conception in man, Table 1) progressive formation of new tissue (histogenesis) and growth are the characteristic processes that take place during the fetal period. There are, however, significant differences between species in this latter stage of pregnancy. The human fetus, as with most other mammalian species, is contained in the amniotic fluid where it is surrounded by fetal membranes and nourished via the placenta through the umbilical cord. The newborn child at birth weighs about 3.5 kg. The mass of the conceptus has thus increased about five-hundred fold from the end of the embryonic period to birth and by more than a factor of 10^7 since conception.

The Newborn Child

The amount of radionuclide present in the newborn child also needs to be estimated. The calculated body content at birth may then be used to calculate the committed dose to age 70 years using biokinetic models for infants and children given in ICRP Publications 56, 67 and 69[1,11,12].

DEVELOPMENT OF BIOKINETIC AND DOSIMETRIC MODELS

Transfer of Radionuclides to the Fetus

Radioactive materials have to cross a number of membrane barriers to reach the tissues of the embryo and fetus from the maternal circulation. The processes involved in this transfer include simple diffusion, facilitated transport, active transport, movement through pores and channels, and pinocytosis[13-16]. Radionuclides which are isotopes of stable elements that are required by the developing embryo and fetus will follow the normal pathways for that element (eg, Na, K, Ca). Isotopes of elements that have similar chemical properties to essential elements are likely to follow similar pathways, although the rates of transfer may differ (eg, Cs behaves similarly to K, Sr like Ca). For other radionuclides the rates of transfer to the embryo and fetus will depend upon their chemical affinity for different transport systems in the body and, as a result, the extent of uptake is unpredictable.

Human data: Most available human data on placental transfer of radionuclides are available from studies with labelled metabolites or pharmaceuticals although some data are available for radionuclides in weapons fallout (eg, Sr, I, Cs). Analyses of autopsy samples have given information on both naturally occurring and artificially produced radionuclides and some comparative information has been obtained through *in vitro* perfusion studies using human placentas.

The rather limited amount of human data makes it essential to use the results of animal studies in the development of dosimetric models for man, although even here information is frequently very limited.

Animal data: In the extrapolation of animal data to man, care is needed; particular problems in the development of dosimetric models for the embryo and fetus include: the varying progress of organ development in different species; the presence of several types of placenta in different species, all of which provide a selective but potentially different barrier between maternal and fetal blood; the rapidity of growth; the complex pattern of growth and differentiation with the potential for quite different distributions of radionuclides in the embryo and fetus from that in the mother; and uncertainty about the location of sensitive cells at various stages of development. The pattern of results obtained experimentally for most radionuclides is similar, although overall transfer to the newborn is very variable. In general, concentrations of radionuclides at birth are highest following intakes towards the end of pregnancy although the concentrations are less than in normal tissues. For some bone-seeking radionuclides (eg, Sr) administered at the end of gestation, however, the concentration may be higher in the newborn than the mother, reflecting the demand of the rapidly developing skeleton for calcium.

Dosimetric models: In the development of dosimetric models information is ideally needed on the initial uptake of the radionuclide by the fetal and maternal tissues following its entry into maternal blood and on its subsequent distribution and retention. Data are required on the extent to which activity deposited in maternal tissues is subsequently translocated to the fetus[7] and doses to the fetus from maternal deposits of radionuclides also need to be assessed.

Where sufficient information is available on the uptake and distribution of radionuclides in the embryo and fetus then elemental specific models are being developed. This applies for example to tritiated water and to isotopes of Fe, Sr, I and Cs. For most other elements animal studies provide the basis for the models. In practice, for many elements studies with experimental animals may only give data on relative concentration ratios in the fetal and maternal tissues (C_F/C_M) and details of the distribution of radionuclide in fetal tissues are often not given. In these circumstances, a general approach has been adopted for calculating the dose to the developing embryo and fetus.

For the embryo (to 60 days after conception), the dose is taken to be the same as that to the uterus; on this basis all tissues of the embryo will receive the same dose. For calculating the dose to the tissues of the fetus for non-penetrating radiation (alpha and beta emissions and photons with E <10 keV), experimentally determined C_F/C_M ratios obtained shortly after administration of radionuclides to the mother are used. Multiplying this ratio by the dose to maternal tissues gives an estimated dose to the fetal tissue, on the assumption that the distribution of activity in the fetus is the same as that in the mother. This approach is expected to be conservative in most instances as rapid growth of the fetus will reduce the concentration of radionuclides in fetal tissue, although this dilution effect may be to some extent offset by transfer to the fetus from radionuclide deposits in the maternal tissues. A number of radionuclides would be expected to equilibrate rapidly between maternal and fetal tissues. These include isotopes of the alkali metals, which are predominantly ionic in body fluids and have a rapid turnover in tissues, as well as many tritium and ^{14}C labelled compounds.

For the majority of radionuclides, penetrating radiation from activity deposited in maternal tissues will also contribute to the dose to the fetus. This dose is presently taken to be the same as the dose to the uterus. It is expected that in final calculations doses will be based on models for a pregnant woman.

As indicated above, for many animal studies the distribution of radionuclide in fetal tissues is not reported and the assumption has therefore to be made for calculating doses that the distribution in the fetus is the same as that in the mother. Recent autoradiographic studies have, however, shown that following administration to the mother early in gestation there can be a significant uptake of some radionuclides in the yolk sac membranes (eg, ^{95}Nb, ^{106}Ru, ^{210}Po and ^{239}Pu). This may be important as haemopoietic development involves the migration of stem cells and it appears likely that the primitive stem cell population may arise in the yolk sac, or even at the earlier egg cylinder stage. These stem cells may subsequently migrate to the liver once the extra embryonic and embryonic circulations become connected. Ultimately haemopoietic stem cells pass to the bone marrow. It has therefore been proposed that for calculating doses to the haemopoietic stem cells in the developing fetus, the calculated dose should reflect this migration and the potential of these cells to receive a radiation dose at different sites in the body (yolk sac, liver and bone marrow), taking into account the appropriate period of gestation in the human.

DOSE COEFFICIENTS FOR THE EMBRYO AND FETUS

For the calculation of dose coefficients a range of intake regimens by the mother have been adopted that should allow doses to the embryo and fetus to be determined for any pattern of exposure of the mother. For acute exposures, intakes of radionuclides by ingestion or inhalation (for aerosols with activity median aerodynamic diameters of 1 μm and 5 μm) are taken to occur at the start of weeks 1, 5, 10, 15, 25 and 35 of pregnancy and at 6 months and 2½ years before conception. For chronic exposures, intakes are taken to occur during the year of pregnancy starting from the time of conception, and for 1 year and 5 years before conception. It is proposed that in the final report equivalent doses to the date of birth will be given for the embryo and fetus for all tissues with a specific tissue weighting factor in adults as well as the average dose to remainder tissues (calculated as the mass weighted average). The effective dose will also be given using the w_Ts recommended by ICRP in Publication 60[2]. Whilst these values are not strictly appropriate for exposures *in utero*, they have been used as no alternative tissue weighting factors are presently available. The effective dose provides a useful quantity for assessing the consequences of different patterns of intake of radionuclides and for comparison with maternal doses. Effective doses (to age 70 years) received after birth will also be calculated, together with the total effective dose (before and after birth) received by the offspring and its ratio to the total committed effective dose received by the mother. It

is intended that the dose coefficients will be given for all elements and radionuclides in Publications 56, 67, 69 and 71[1,11,12,17]. The results of some preliminary calculations are given in Tables 2 and 3.

In general, the results of preliminary calculations indicate that for many radionuclides doses to the offspring will be less than those to the mother. For isotopes of caesium and other radionuclides that equilibrate rapidly in the body, the dose to the newborn child is expected to be similar to that of the mother, although the total committed dose may be much less (Table 1). For acute intakes of ^{131}I late in pregnancy the dose to the newborn child may approach twice that of the mother. Similarly, intakes of alkaline earth radionuclides towards the end of pregnancy can result in higher doses to fetal tissues at birth than to maternal tissues, although the total committed dose to the offspring will be less. There is little suggestion that doses to the child will be appreciably higher than maternal doses, particularly for chronic intakes by the mother either before or during pregnancy.

It is expected that the report will be ready for publication in about 18 months time.

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TABLE 1

Approximate time in days of the beginning and end of the major developmental periods in some mammalian species

Species	Pre-implantation	Major organogenesis	Post Conception Fetal period
Hamster	0-5	6-12	13-16.5
Mouse	0-5	6-13	14-19.5
Rat	0-7	8-15	16-21.5
Rabbit	0-5	6-15	16-31.5
Guinea-pig	0-8	9-25	26-63
Dog	0-17	18-30	31-63
Man	0-8	9-60	60-270

UNSCEAR (1977)[18]

TABLE 2

Comparison of effective doses (Sv) to offspring and mothers for 1 Bq acute intake by ingestion at start of week 10 of gestation

Nuclide	Offspring			CED to mother	Offspring/mother
	in utero	Birth to 70 y	Total		
HTO	$2.5\ 10^{-11}$	$6.9\ 10^{-15}$	$2.5\ 10^{-11}$	$1.8\ 10^{-11}$	1.4
^{106}Ru	$2.8\ 10^{-10}$	$2.5\ 10^{-11}$	$3.0\ 10^{-10}$	$1.0\ 10^{-8}$	0.03
^{131}I	$4.0\ 10^{-8}$	$3.2\ 10^{-17}$	$4.0\ 10^{-8}$	$2.2\ 10^{-8}$	1.8
^{137}Cs	$1.0\ 10^{-8}$	$5.0\ 10^{-10}$	$1.1\ 10^{-8}$	$1.3\ 10^{-8}$	0.80
^{239}Pu	$6.6\ 10^{-10}$	$4.2\ 10^{-9}$	$4.9\ 10^{-9}$	$2.5\ 10^{-7}$	0.02

TABLE 3

Comparison of effective doses (Sv) to offspring and mothers to birth for 1 Bq acute intake by inhalation[a] at various times before and after conception

Nuclide	Time of intake (wks)				
	-26	1	5	25	35
	Fetus/Mother				
HTO (V)[b]	0.1	1.1	1.1	1.4	1.4
^{106}Ru (M)	0.01	0.03	0.02	0.01	<0.01
^{131}I (F)	<0.01	0.01	0.15	1.9	1.9
^{137}Cs (M)	0.28	0.14	0.14	0.09	0.06
^{239}Pu (M)	<0.01	0.02	0.02	<0.01	<0.01

Notes:

[a] Activity Median Aerodynamic Diameter = 1 μm.
[b] Default lung Type from ICRP Publication 71[(17)].
V - Vapour, F - Fast clearance rate, M - Moderate clearance rate.

CANCER RISK AMONG ATOMIC BOMB SURVIVORS

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The Radiation Effects Research Foundation (RERF) and its predecessor, the Atomic Bomb Casualty Commission (ABCC), has been conducting a long-term follow-up of a cohort of the atomic bomb survivors in Hiroshima and Nagasaki. The continuing follow-up of this population, known as the Life Span Study (LSS) cohort, has been a major source of epidemiological data for radiation risk assessment (1, 2). Periodic analyses of the LSS mortality data have resulted in a series of reports that describe and quantify radiation effects on cancer mortality (3). More recently, a series of comprehensive reports of cancer incidence for this cohort has also been published (4-7). The latest report on the LSS cancer mortality data through 1990 will soon be published. The purpose of this presentation is to provide an updated overview of the LSS cancer and leukemia data.

LIFE SPAN STUDY

Population

The LSS cohort includes most survivors within 2.5 km of the bombings who lived in Hiroshima or Nagasaki in 1950 and who met certain conditions that were considered favorable for follow-up. The cohort also includes a sample of survivors who were within 2.5 - 10 km of the hypocenter; this group was chosen to be of the same size and to have the same age and sex distribution as the original group of more proximally exposed survivors. The portion of the LSS cohort used in the latest analysis includes about 86,500 survivors for whom DS86 radiation dose estimates are currently available (8, 9). This represents 93% of all the survivors in the cohort, and consists of 34,300 survivors with estimated doses less than 0.005 Gy kerma and 52,200 with estimated doses of 0.005 Gy or higher (a mean dose of 0.26 Gy). The LSS data are characterized primarily by acute exposure to low LET gamma radiation, with a non-negligible neutron component in Hiroshima. However, the 37,000 subjects in the dose range of .005 - .20 Gy can also provide useful information on low-dose effects.

Follow-up

Mortality follow-up for the period from 1950 onward has been undertaken using regular checks on the vital status of all cohort members based on the Japanese family registration system known; causes of death are obtained from information recorded on death certificates. This method of follow-up provides virtually complete ascertainment of vital status and mortality data. The LSS cancer incidence data that have recently become available from the Hiroshima and Nagasaki tumor registries provide better diagnostic information and improve ascertainment of relatively non-fatal cancers; these are an important complement to the mortality data. A subset of the LSS cohort has been followed by biennial health examinations under the Adult Health Study program, providing longitudinal data on disease morbidity and health status. These mortality and morbidity follow-up data enable comprehensive analyses of a wide spectrum of health outcome associated with radiation exposure from the atomic bombings.

RESULTS TO DATE

At the end of 1990, 56% of the LSS cohort members were alive while most of those exposed at middle ages or later in life had died (Table 1). However, it is particularly important to note that some 85% of those exposed at ages less than 30 years are still alive

Table 1
Cohort survival by age at exposure

Age at exposure	People in 1950	Percent alive in 1990
0-9	17,824	94
10-19	17,557	86
20-29	10,882	77
30-39	12,270	51
40-49	13,489	16
50 +	14,550	1
Total	86,572	56

Excess cancer deaths

Excess leukemia deaths among the atomic bomb survivors became apparent a few years after the bombings, reaching a peak 5-10 years after the exposure, followed by a gradual decline. Since 1950, a total of 249 leukemia deaths occurred in 86,500 survivors in the LSS cohort. Of these, 86 leukemia deaths are considered as excess deaths due to radiation, and these account for 49 % of 176 leukemia deaths in the 52,200 subjects with significant exposures (Table 2).

Table 2
Cancer deaths among 52,200 A-bomb survivors with significant exposures
1950-1990

	Total number of deaths	Estimated number of deaths due to radiation exposure	% attributable to radiation exposure
Leukemia	176	86	49%
Other types of cancer	4,687	341	7%
Total	4,863	427	9%

Excess deaths from cancers other than leukemia emerged 5-10 years after the exposure. To date, a total of 4,687 deaths from cancers other than leukemia have occurred in 52,200 survivors with significant exposures in the LSS cohort (Table 2). Of these, 341 or 7 %, are considered to be associated with radiation exposure. Thus, for cancers other than leukemia, or solid cancers, the proportion of deaths due to radiation is much smaller than for leukemia but the absolute number of excess deaths is much larger.

Temporal patterns of cancer risk

The long-term follow-up of the LSS provides a unique opportunity to examine the temporal patterns of cancer and leukemia risk as they are affected by age at exposure and sex. As evident from the discussion below, understanding of such temporal patterns is essential in risk assessment. The age-dependent temporal patterns are strikingly different for solid cancers and leukemia. Figure 1 depicts the age and sex dependent temporal patterns of solid cancer risk. One of the most important findings here is that, for those exposed as adults, the excess relative risk per unit radiation dose (ERR) of solid cancers has been remarkably constant over the follow-up period. This means that the excess absolute risk has increased over the entire follow-up period roughly in proportion to the rapid age-specific increase of the background risk. For those exposed as children, the ERR of solid cancers, which was especially high in the early years of follow-up, has been found to be decreasing over time. Since the very high ERR for this group early in the follow-up is based on a very small background risk at young ages, the absolute risks for the young survivors are still relatively small. Because of the large number of the young subjects who are yet to be followed, how the solid cancer risk will behave for the young survivors in the future is one of the main remaining questions.

The ERR for solid cancer is also higher for women than men. However, in interpreting the sex difference in ERR, one must keep in mind that the background rates for men are at least twice as high as for women. The absolute risk for cancers other than leukemia is quite similar for men and women.

In sharp contrast to the solid cancer deaths, most of the excess leukemia deaths occurred early in the follow-up. Patterns of risk for leukemia are shown in Figure 2. Generally the excess relative risk decreases with time (or with aging), but the rate of decrease in the risk is lower (or the decrease is less rapid) for those exposed at older ages. Particularly among those exposed as children, essentially all of the excess deaths occurred during the several years after the exposures. On the other hand, among those exposed as adults, the excess risk persists at low levels throughout the entire follow-up period.

Dose response curves

Another important finding from the LSS follow-up data is a striking difference in the shape of a dose response curve for solid cancers and leukemia. The dose response curve for solid cancers is unequivocally linear whereas there is a statistically significant upward curvature in the leukemia dose response curve up to about 3 Gy. The difference in dose response curve for solid cancers and leukemia may reflect different underlying mechanisms for carcinogenesis and leukemogenesis, as the former is known to involve multi-step processes and multiple factors whereas for the latter chromosomal rearrangements are of primary importance.

The low dose extrapolation factor (LDEF) is the adjustment factor to be applied to linear risk estimates in extrapolating the excess risk to low doses. The shape of the dose response curve for solid cancers and leukemia has been analyzed using the LSS data (10,11). Based on the LSS incidence data, LDEF values for solid cancers were in a narrow interval around 1, with a value of 1.4 or greater being inconsistent with the data. For leukemia, the best fitting LDEF value was 2.5 with but a value as high as 10-15 could not be ruled out.

Lifetime risk projection

The LSS cancer data are used by various national and international organizations concerned with radiological protection. In setting up safety standards, considerations are given to projected life time risk estimates and other measures of health detriment attributable to radiation exposure. At the present time, one of the most important sources of uncertainty in lifetime risk estimates is the projection that must be made from the incomplete follow-up. This is especially true for those exposed at young ages.

To examine the effect of different assumptions about the future course of the solid cancer risk, the UNSCEAR 1994 Report used three different models in projecting the lifetime risk that an individual would die from cancer (or leukemia) as a result of exposure to radiation (Table 3). In one model, the ERR for solid cancer is assumed constant throughout the lifetime of the survivors. This is a commonly used method but seems somewhat unreasonable since it is now clear that the solid cancer ERR among young survivors has been decreasing over the follow-up. In two alternative models, the solid cancer ERR for the survivors who were less than 45 years old at exposure is assumed to decrease linearly starting 45 years after exposure but at different rates of decrease. The excess leukemia risk in the LSS occurred mostly in the early years of follow-up and appears to be decreasing for the youngest survivors. Therefore, there is no need to consider different projection models.

Table 3
Comparison of estimates of lifetime risk of mortality form solid cancers and leukemia following acute whole-body exposure to 1 Sv
(from UNSCEAR 1994 Report)

Projection method	Lifetime risk in percent	Years lost per case
Solid cancers		
Constant relative risk [a]	10.9	11.6
Decline to risk for age at exposure 50 [b]	9.2	12.3
Declines to zero risk at age 90 years [c]	7.5	13.3
Constant relative risk (UNSCEAR 1988)	9.7	11.4
Leukemia		
Linear-quadratic dose response model	1.1	31
Constant relative risk (UNSCEAR 1988)	1.0	26

[a] ERR constant from 10 years after exposure

[b] ERR constant for first 45 years after exposure; risk then decreases linearly with age. At attained age 90 years, the risk is equal to that for a person aged 50 years at exposure

[c] ERR constant for first 45 years after exposure; risk then decreases linearly with age. At attained age 90, the risk is equal to that for a person aged 50 years at exposure.

COMMENT

The LSS cohort data have provided unique opportunities for elucidating the nature and magnitude of risk associated with radiation exposure as it is affected by sex, age and time. With only about 50% of the subjects for whom the follow-up has been completed, there clearly is still much to be learned from the future follow-up. The most important uncertainty at the present time relates to the temporal pattern of solid cancer risk of those survivors who were exposed early in life. How the risk for this group of young survivors will behave in the future is one of the most important remaining questions. Although the patterns of leukemia risk are much more complex than those for solid cancers, most of the excess deaths occurred in the early years of follow-up. There appears much less uncertainty with respect to risk estimates for leukemia, although continued work is also important to obtain a complete picture on the temporal distribution of leukemia risk.

Figure with legend

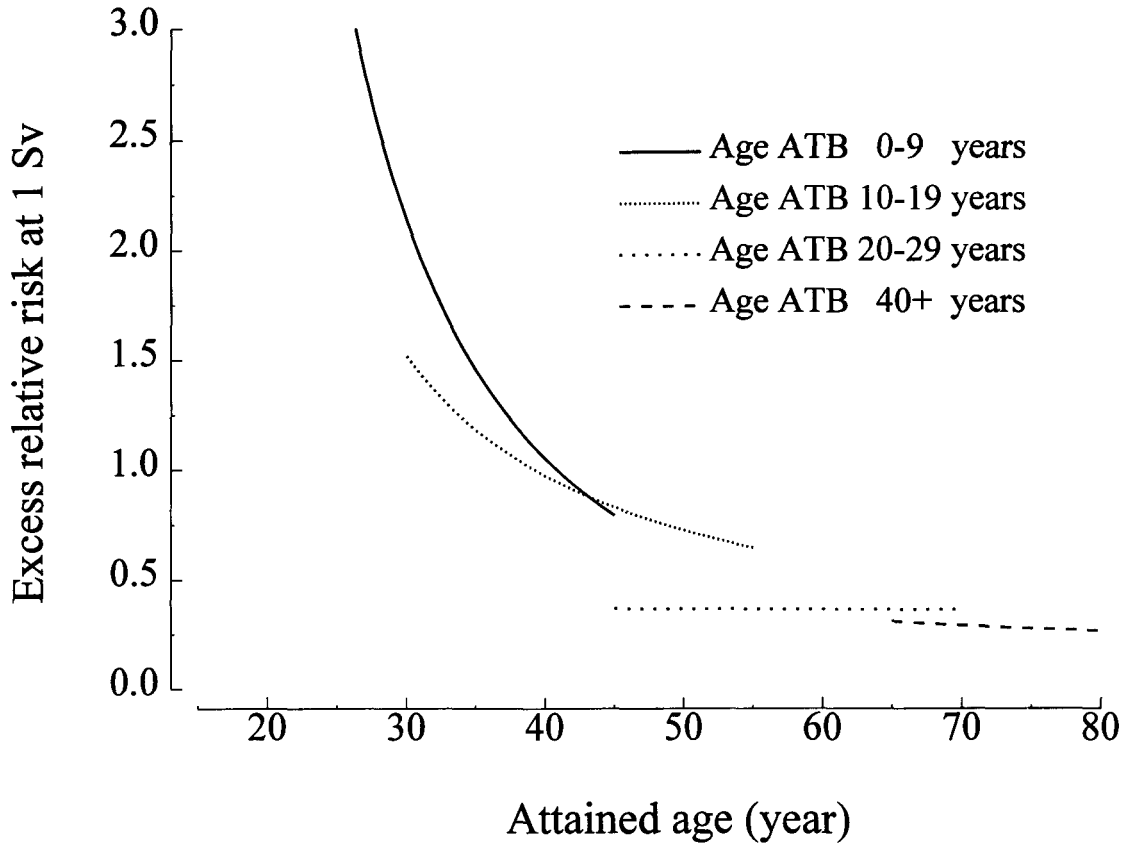

Figure 1. Temporal patterns of solid cancer risk

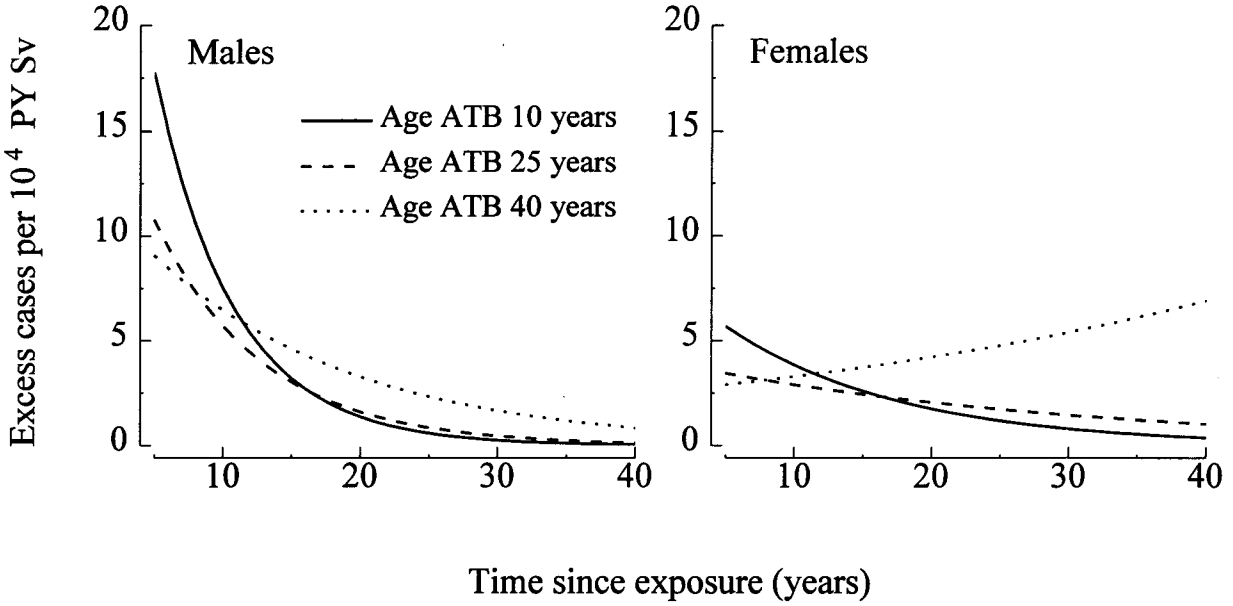

Figure 2. Temporal patterns of leukemia risk

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METHODOLOGY OF ENVIRONMENTAL DOSE ASSESSMENT

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INTRODUCTION

It has often been said that risk assessment is more an art than a science. In the early days of risk assessment, this may have been a true statement. Indeed, the components that comprise a risk assessment evolved as individual sciences, considering transport of radionuclides in the environment, dosimetry, and risk, once exposure has occurred. These individual sciences gradually and more frequently were merged together into comprehensive dose and risk assessment models. Risk assessment is a process that is rapidly becoming very sophisticated in its techniques, and through extensive efforts in validation of models, it is gaining credibility as a reliable scientific approach to understanding risk and making decisions about risk. Today, risk assessment is indeed a science that will become more important in the future of radiological health.

Risk assessment can be either prospective or retrospective. Risks can be estimated for possible future releases of materials (prospective), or they can be estimated for releases that occurred in the past (retrospective). Prospective risk assessments are carried out in conjunction with new facilities that are being constructed or perhaps to demonstrate that compliance will be met for an operating facility. An excellent example of retrospective risk assessments are the dose reconstruction studies being conducted on the weapons complex facilities in the US and studies of populations exposed following the Chernobyl reactor accident. Although the two types of risk assessment may be undertaken somewhat differently in their methods, there are many similarities in the techniques applied to both.

In explaining the process of risk assessment to my colleagues and to the public, I have often used this illustrative equation to express the interdisciplinary nature of this research and how these disciplines are inherently linked together.

$$\text{Risk} = (S \bullet T \bullet U \bullet D \bullet R)_{uvcp}$$

where,

S = source term (characterization of the quantity and type of material released),
T = environmental transport and fate of the material released,
U = usage factors (characteristics of individuals exposed),
D = conversion to dose,
R = conversion to risk,
u = uncertainty analysis,
v = validation,
c = communication of results, and
p = public participation.

This presentation will briefly review the state-of-the-art of each of these basic elements of risk assessment and describe what, in my judgment, the future holds and where the focus of our efforts should be for improving this science. Since our research has recently focused on dose reconstruction in the US, I will use examples from these studies throughout this paper.

SOURCE TERM (S)

The source term is the characterization and quantification of the material released to the environment. It is the heart of a risk assessment. We frequently give too little attention to the derivation of the source term, and yet this step of risk assessment is where the greatest potential lies for losing scientific and public credibility. Therefore, it is important that development of the source term be given highest priority and that the source term be carefully checked before making risk calculations.

Two key points are offered about how source terms should be derived for risk assessments. First, uncertainties must be included with the estimates of releases. This aspect of the source term has frequently been overlooked in the past, with release estimates being reported as single values, when in reality we know there is a range of possible values that exist. The uncertainty estimates should account for all possible sources of uncertainty in the calculation.

Second, the source term should be derived from as many different independent approaches as possible. The Fernald Feed Materials Production Center (FMPC), near Cincinnati, Ohio in the US is a part of the US nuclear weapons complex that processed uranium ore. We estimated the release of uranium from the FMPC using two

methods. The first method considered the amounts of material being processed at the site and estimated the fractional release of uranium to atmosphere through the various offgas treatment systems (primarily scrubbers and dust collectors). Using this approach, it was determined that the "best estimate" of uranium released to atmosphere was 170,000 kg with the 5th and 95th percentiles ranging between 270,000 kg and 360,000 kg (Table 1). An alternative calculation was performed (1), looking at the amount of uranium deposited on soil within 7.5 km of the site based on soil samples that had been collected over time. Taking into account environmental removal of some of the uranium, and the amount of uranium that would have been deposited from the atmosphere, it was estimated that the source term for uranium released from the site to the atmosphere would lie between 78,000—90,000 kg with a median value of 212,000 kg. This alternative calculation, although the uncertainties are large, gives us additional confidence that our estimate of the source term for uranium is reasonable.

The Fernald site included several concrete silos in which ore containing high concentrations of radium was stored. Between 1952 and 1979 these silos (known as K-65 silos because of the radium bearing material stored inside) were vented to the atmosphere, thus creating a strong source of radon and radon daughter products, that exposed nearby residents (Figure 1). After 1979 several remedial measures were taken to reduce the emissions of radon, and in more recent years, a number of measurements were made to estimate radon releases. Unfortunately during the period prior to 1979, few measurements were made to quantify the release of radon, at a time when the releases would have been the greatest.

Table 1. Uranium and Radon Source Terms for the FMPC for 1951-1988[a] (1)

Source	Median release estimate	5th percentile	95th percentile
U to Atmosphere			
Primary estimate	310,000	270,000	360,000
(Alternative calculation)	212,000	78,000	390,000
U to Surface Water	99,000	85,000	120,000
Radon to Atmosphere			
Primary estimate	6,300 TBq	4,100 TBq	8,500 TBq
(Alternative calculation)	3,200 TBq	540 TBq	11,000 TBq
Radon-222 daughters[b]	4,800 TBq	3,200 TBq	7,000 TBq

[a] Values are in kg of uranium, except releases from the K-65 silos which are reported in TBq.

[b] The release quantities for decay products are quantities of each of the short-lived decay products, polonium-218, lead-214, bismuth-214, and polonium-214.

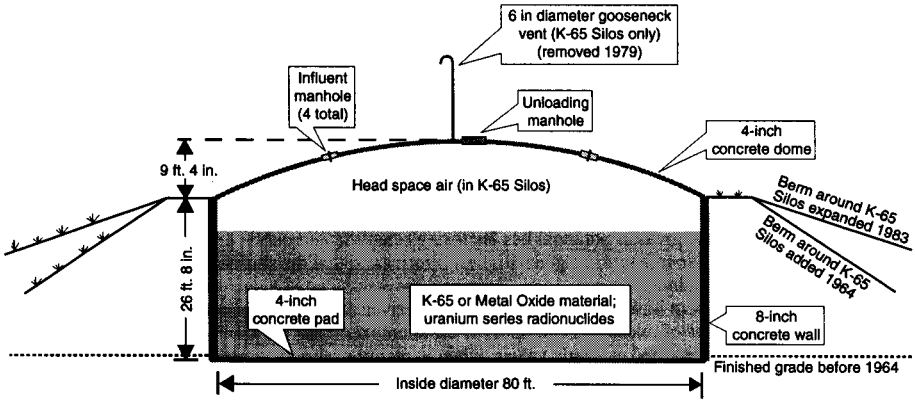

Figure 1. Schematic diagram of the K-65 silos located on the Fernald site.

The radon source term for Fernald was particularly challenging to derive. As with uranium, we estimated radon emissions and uncertainties for the silos using two different techniques. One method used measurements that were made prior to and after the sealing of the silos in 1979, along with information we had about the history of the silos and characteristics of the material stored. The second technique was a strictly theoretical approach that calculated the amount of radium in the silos, estimated the radon production and diffusion into the head space and out of the silo vents. The theoretical technique resulted in a greater uncertainty because more parameters with wide distributions were involved in the calculation. However, as can be seen in Table 1, there is overlap in the two estimated source terms, giving us confidence that our estimates for radon are plausible.

Environmental Transport (T)

With regard to environmental transport, one important recommendation stands out — always apply site specific data when they are available. The more data that exist characterizing the environment around a site, the more defensible the risk assessment. In fact, where measurements of environmental concentrations can be used in place of models, this is always preferable.

One example of a case where we have applied environmental measurements to assist in risk assessment is for the dispersion of releases to the atmosphere at the Fernald site. As with many dose reconstruction studies, detailed information on atmospheric conditions (wind speed and direction, stability categories, precipitation) at the site were not available until the late 1980s. In our search of the historical records, we located two sets of measurements of radon at different distances downwind from the silos, collected during the 1980s. A diffusion coefficient was developed based on a Gaussian area source model for a circular area of radius 50 m (representing the area of the two silos from which the radon emerged). The curve was fit to the radon measurement data. The uncertainties account for factors such as parameters used in converting concentrations to diffusion values and uncertainties in release rates of radon from the silos as well as measurement error.

We applied this empirical dispersion model to the releases of both uranium and radon released to the atmosphere at Fernald (Figure 2). Obviously, many other factors had to be accounted for in the overall model such multiple release points at the site, different chemical forms of uranium, particle size, deposition and resuspension among others. However, the air concentration as a function of distance downwind was based on measurements taken near the site. This approach resulted in less overall uncertainty in the environmental transport calculations. Therefore, it is always a smart investment in resources to carefully characterize environmental properties at the site for which the risk assessment will be performed. In the end, a more defensible calculation can be made and in addition, less uncertainty is likely to be introduced into the calculation.

USAGE FACTORS (U)

The importance of usage factors is often overlooked in risk assessment. The term, usage factor, refers to diet, lifestyle, and residence history data for individuals who live near the site, and other information that characterize the region around the facility being studied such as land use and agricultural practices. Typically we put little effort into developing usage factors in risk assessment and opt instead to insert values taken from the literature which often have little relevance to the particular region being studied. Although this tactic may not introduce large uncertainty into the overall estimation of risk, it can result in a loss of credibility with the public who are the object of the study.

In almost every risk assessment there are special population groups who do not fit the usage factors for the general public. One example of this occurred in the dose reconstruction project for the Hanford Site, another facility within the US nuclear weapons complex, located in south-central Washington state in the US. The Hanford facility released large amounts of radionuclides, and I-131 in particular to the atmosphere (3), however significant quantities of materials were also released directly into the Columbia River which was used for cooling the production reactors at the site (4). Pathways of exposure from the river were investigated thoroughly. Members of the general public who lived near the river received relatively small doses (estimated to be 15 mSv over about 40 years) from consumption of river water, consumption of fish from the river (150 kg of fish per year), and activities in and around the river. However, special attention had to be given to Native Americans who relied on the river for a major component of their food, fish. Therefore, the types of fish consumed, the quantities of fish eaten, and other uses of the river had to be determined specifically for Native American tribes near the site. Such information is not available in the literature and had to be collected by surveying elders of the tribes to develop a typical diet and to determine where the principal fishing locations existed. These data are currently being collected and analyzed, however, it is evident that the Native Americans who lived near and downstream of the site are likely to have received significantly higher doses from this pathway than the general public. Such an analysis could not be made without the cooperation of the Native American tribes in the Hanford study.

Surveys of individuals being studied to obtain person-specific usage factors were also carried out in the Utah Thyroid Cohort Study (5). In that research, surveys were conducted on the parents (or another living relative) of approximately 3,500 individuals who were children at the time of exposure to iodine from fallout resulting from the

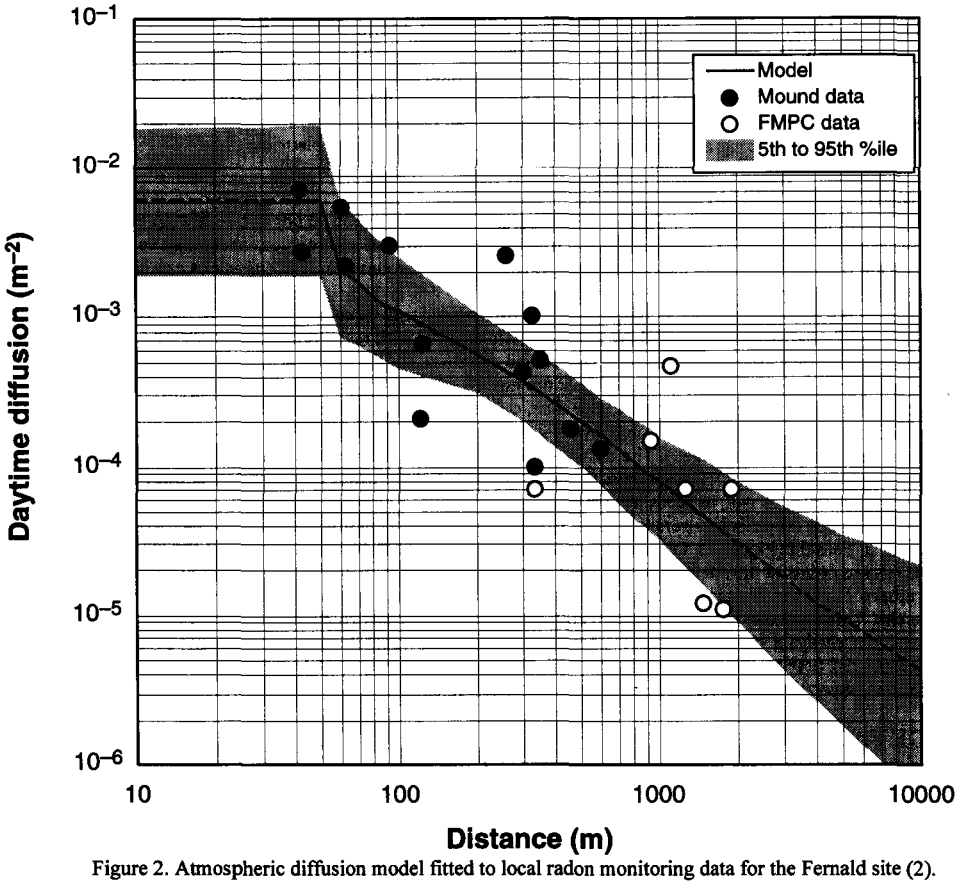

Figure 2. Atmospheric diffusion model fitted to local radon monitoring data for the Fernald site (2).

atmospheric nuclear weapons tests in the US. These data allowed individual doses and uncertainties to be estimated for each member of the cohort. The doses and uncertainties were used to determine if a higher incidence of thyroid disease existed in the group and could it be attributed to the fallout.

These two examples illustrate the importance of usage factors in risk assessment and the fact that the collection of such site-specific data cannot be underestimated.

DOSE CONVERSION FACTORS (D)

The conversion of intake of radionuclides or external exposure to dose has become a simple process because of effort put into deriving and publishing dose conversion factors using metabolic models. However, these factors typically do not include uncertainties and therefore a key component in the overall risk calculation is often not addressed.

In the Hanford Environmental Dose Reconstruction project (3-4), it was determined that one of the two most important contributors to overall uncertainty was the dose conversion factor for I-131 (the other key component of uncertainty was the feed-to-milk transfer coefficient.) In this analysis, it was pointed out that the uncertainty in the iodine dose conversion factor was due primarily to variability in the mass of the thyroid, uptake of iodine in the gastrointestinal tract, transfer of iodine to the thyroid, and the biological half-time of iodine.

Estimating uncertainties for dose conversion factors is a fertile and important area for research in the future. Although work in this area is underway, it will be a considerable time before dose conversion factors and their

uncertainties can be established for all radionuclides. Indeed, this is an area where priorities must be set, in order to establish the methods to be used and to be sure we are focusing first on radionuclides of greatest importance in risk assessment.

CONVERSION TO RISK (R)

Two key issues are important to note with regard to risk conversion. The first issue deals with inclusion of uncertainties in estimates of dose in epidemiological studies to analyze incidence of disease and resulting risk. Kerber et al. (5) incorporated individual doses and uncertainties (6) into their analysis of thyroid disease in a cohort exposed to fallout from the Nevada test site. They concluded that there was a statistically significant excess of thyroid neoplasms, with an increase in excess relative risk of 0.7% per mGy. This approach to incorporating uncertainties associated with doses into the epidemiological analysis will be used more commonly in the future and is a relatively new technique for deriving estimates of risk in studies of large populations where individual doses have been assigned.

The second issue related to conversion to risk is that of quantifying uncertainty in the risk factors being applied. As with conversion of intake or external exposure to dose, conversion of dose to risk is a straight forward process with the publication of risk factors by a number of different groups (7-9). Little is known about uncertainties associated with the risk factors in use today. The current risk estimates of cancer following exposure to ionizing radiation are based primarily upon analyses of Japanese survivors of the atomic bombings at Hiroshima and Nagasaki. These risk estimates essentially relate to uniform whole-body low LET radiation exposures to doses ranging from 0.01 Gy to 4 Gy delivered at high dose rate.

Uncertainty in the risk factors for radiation was described by Sinclair (10) as having five primary components: (1) epidemiological uncertainties; (2) dosimetrical uncertainties (3) projection to lifetime; (4) transfer between populations; and, (5) extrapolation to low dose and dose rate. Epidemiological uncertainties include statistical uncertainties associated with quantifying the relatively small number of excess cancers attributable to ionizing radiation from the background cancers resulting from all causes. Also included in epidemiological uncertainties are uncertainties from underreporting of cancers per unit population and non-representativeness of populations used to determine risk. Dosimetrical uncertainties include those from random errors in individual dose estimates arising from errors in the input parameters used to compute doses and systematic errors due to the presence of more thermal neutrons at Hiroshima than originally estimated. Risk projection includes uncertainties associated with extrapolating beyond the time period covered by the observed population. Transfer of estimates of risk from one population (Japanese) to another introduces an additional source of uncertainty that must be considered. Finally, since the exposures for the A-bomb population was at relatively high dose rate, uncertainty is introduced when we extrapolate estimates of risk to low dose, low dose rate situations, common in most risk assessments. This area of risk assessment research is very important for the future and the ideas introduced by Sinclair (10) must be pursued. Indeed, it is very possible we may find that the risk factors themselves introduce more uncertainty into the overall estimate of risk than any other single component.

UNCERTAINTY ANALYSIS (u)

Uncertainty analysis is an essential element of risk assessment. Methods for quantification of uncertainty have been well established. Today, it is expected that when one carries out a risk assessment, the best estimate of risk is reported along with associated uncertainties. Although we commonly see uncertainties associated with doses, it is not likely to see uncertainties calculated in overall risk, primarily because as mentioned above, uncertainties in the risk factors are still being developed.

The most common method for uncertainty analysis uses Monte Carlo statistical techniques incorporating a random sampling of distributions of the various models and parameters involved (Figure 3). In this simplified illustration, A is an input parameter to the model, and Y is the result, or output, corresponding to A. For each specific value of A, the model produces a unique output Y. Such an application of the model is deterministic, because A determines Y. But A may not be known with certainty. If uncertainty about A is represented by a distribution, such as the triangular one in the figure, repeatedly sampling the distribution at random and applying the model to each of the sample input values A_1, A_2... gives a set of outputs Y_1, Y_2,..., which can be arranged into a distribution for Y. The distribution of Y is then the estimate of the uncertainty in Y that is attributable to uncertainty in A. This is a stochastic, or Monte Carlo application of the model.

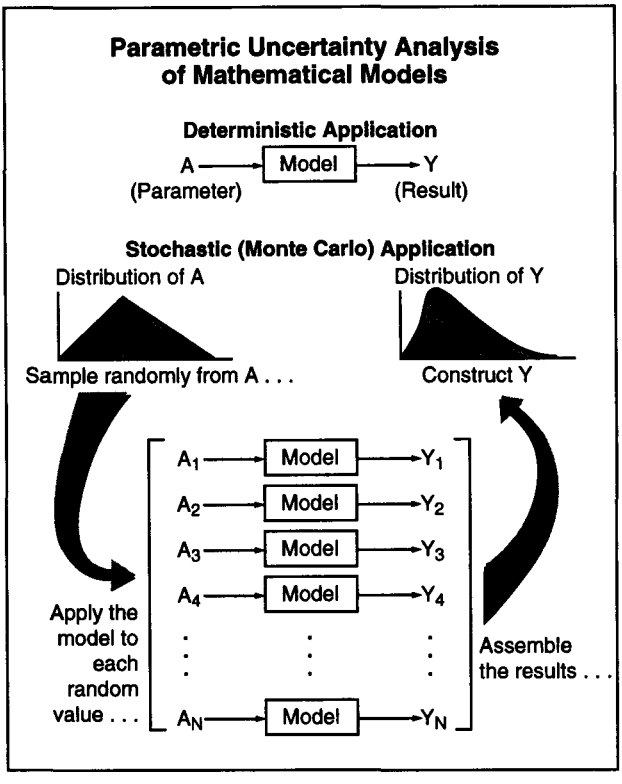

Figure 3. Schematic presentation of Monte Carlo methods for propagating a parametric uncertainty distribution through a model to its results.

The goal to estimate uncertainties in risk assessment demands that powerful computers be available for such a calculation. Until recently, limitations in computer hardware were the primary reason why uncertainties were not propagated throughout the entire calculation. Today, however, computer memory and speed of calculations have advanced to the point where relative inexpensive computers are required to undertake such a complex calculation.

VALIDATION (v)

Retrospective risk assessments in dose reconstruction are providing a unique opportunity to validate environmental transport models. The historical records contain a vast resource of measurement data, much of it collected many years ago, that can be used as the basis for comparisons between estimated environmental concentrations and values that were measured. These comparisons are providing us with some of the best indicators of reliability to date for the mathematical models being developed and applied by scientists in risk assessment.

In the Fernald Dosimetry Reconstruction Project, we have examined all available monitoring data from several different sources, and calculated predicted to observed ratios for a number of different pathways. As expected, it is rare that perfect agreement exists; however, overall confidence in our methodology is boosted by the strong agreement between historical data and estimated concentrations, suggesting that neither the source term nor the environmental transport calculations are seriously off the mark.

For example, Killough et al. (11) have made extensive use of historical monitoring data to confirm that estimates of releases of uranium in air and surface water are reasonable. Figure 4 shows one example of comparisons of predicted uranium concentrations in the Great Miami River near Fernald to those measured for the same period. Good agreement exists between predicted to observed values (mean P/O value of 1.3), enhancing

confidence that the source term and environmental concentrations predicted reasonably reflect conditions that actually existed at the time.

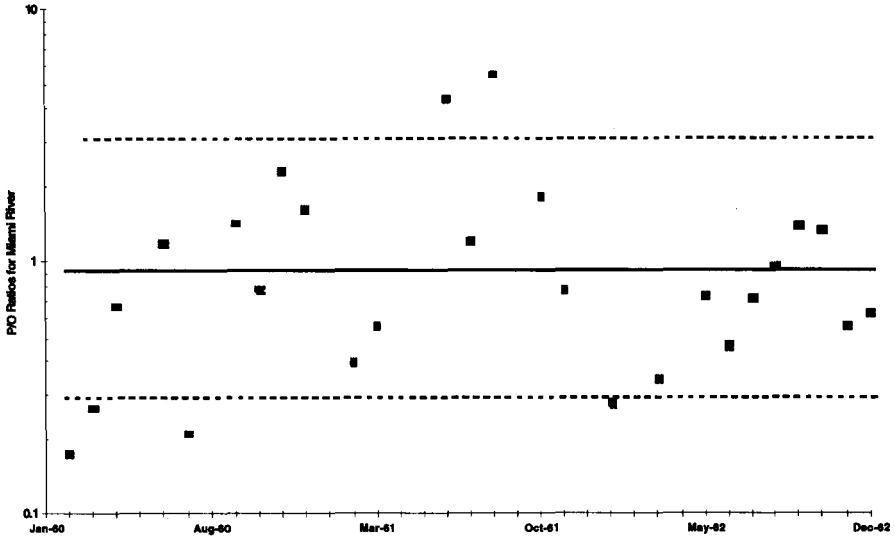

Figure 4. Comparison of predicted to observed concentrations of uranium in the Great Miami River downstream of the Fernald site (11).

Validation of our methods must continue to receive strong attention in the future. We must be certain that every opportunity is taken to make comparisons between predicted and observed estimates of concentrations in the environment.

COMMUNICATION OF RESULTS AND PUBLIC PARTICIPATION (*c,p*)

I include the remaining two elements of risk assessment into this final section because they are inherently linked together. Risk assessment is a public process. In almost all situations, we are required to estimate risk to members of the public who have been or who may be exposed to releases of contaminants to the environment. In the end, we must achieve both public and scientific credibility (12). As scientists we have come a long way in establishing the technical methods described in the sections above. On the whole, however, we are quite inept in communicating our results with the public and working with the public in conducting risk assessments. This is an area where the most progress could be made with the least investment of resources.

Risk communication has become a branch of risk assessment in itself. Many techniques have been devised to explain to the public the meaning risks resulting estimated doses. Many comparisons have been proposed between risks from exposure to radiation and risks from other sources common in life. We may be able to help people understand the potential importance of risks by referring to the overall risks posed by natural background, fallout, and other involuntary exposures and comparing these to our own estimates of risk for the particular situation. Although these comparisons help, there is no replacement for scientists personally conveying the results of their calculations to the public who are at risk. We would all like to think that society is ready to agree on a common set of risk levels for decision-making. Indeed it would make life simpler, but we are not at that point with either the technology to compute risks or the public's understanding and acceptance of risk. Trying to impose a level of risk as being "significant" or "insignificant" leads to a serious loss of credibility. During my past decade of work on dose reconstruction projects, I have learned that the best way to respond in communicating the significance of a level of risk is to respond personally. What I mean by this is that we as scientists can tell people how we feel about the significance of a level of risk, but we cannot tell the public what they should feel or think.

Public involvement in risk assessment is not common to us. I have found, however, that including the public in as many ways as possible in the risk assessment process builds credibility. Further, including the public

who are being addressed in the risk assessment can lead to the inclusion of valuable site-specific information as described in the discussions above about usage factors.

CONCLUSIONS

Risk assessment is a science that will gain momentum and recognition in the future. It began as many individual sciences addressing the release of radionuclides to the environment, their transport in environmental media, their ingestion by humans, the resulting dose to organs of the body, and the correlation between these doses and the incidence of disease in the population exposed. Slowly and methodically, the individual sciences were melded into the discipline we call risk assessment. Today, this science is rapidly advancing on both the radiological and chemical fronts. There is no question that in the future, risk assessment will be the major tool for making decisions about building new facilities that are yet to release contaminants, and for understanding the importance of contaminants that have been released in the past.

To say we have come a long way in the advance of this science is an understatement. To say we have much work yet to do is a fact.

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GAMMA-RAY SPECTROMETRY IN THE ENVIRONMENT AND MODELLING OF EXTERNAL EXPOSURES

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INTRODUCTION

Atmospheric releases of gamma emitting radionuclides lead to external radiation exposures of the population. As a rule, the external radiation contributes a significant part to the total exposure of the population that is caused by such a release (1). The importance of the external radiation pathway was demonstrated, *e.g.*, after the Chernobyl accident, especially for longer periods after the accident (2-5). In the aftermath of the Chernobyl accident, the data base for estimating external population exposures has been considerably extended and this paper is devoted to describe a few of the recent developments.

A powerful tool for determining external exposures is gamma-ray spectrometry in the environment (6), since it is fast and allows to determine radionuclide activities per unit area and radionuclide specific absorbed dose rates in air, which in turn is the basis for any solid prediction of the doses to be expected. Due to the pioneering work of Lowder *et al.* (7) and of Beck *et al.* (8,9), a widely accepted procedure of so called *in-situ* gamma-ray spectrometry has been developed (10,11). In this approach, an assumption on the distribution of the radionuclides in the soil has to be made, and this is the main source of uncertainty (6,12). A number of attempts were undertaken to reduce this source of uncertainty by using additional information contained in the measured spectrum. The approaches have been summarized in (6), and more recent approaches are (13,14). Due to large statistical uncertainties, most of these papers were not able to demonstrate the range of applicability of the proposed method. However, Rybaçek *et al.* (15) showed the suitability of a simultaneous evaluation of the 662 keV gamma peak and of the 32 keV Roentgen peak of ^{137}Cs for an improved activity determination during the first years after the deposition of the radionuclide. In this paper, a combined approach based on the peak-to-valley method (16) and on the multi-line approach (17) is demonstrated to improve *in-situ* gamma-ray spectrometry for longer times after a ^{137}Cs deposition.

Since the reactor accident of Chernobyl, the knowledge on three basic parameters in the modelling of external population exposures after radionuclide depositions has been considerably improved. The first parameter is the gamma dose rate absorbed in air per ^{137}Cs activity per unit area deposited on an open undisturbed site. In this paper, a phenomenological approach parameterizing the different values of this parameters for fuel particles, condensation particles and ^{137}Cs attached to the background aerosol by the distance from the release point is described. Further, recent results on the other two parameters, location factors (18,19) and group factors, here defined as the ratio of the effective dose of a population group relative to the effective dose of an adult anthropomorphic phantom standing on an open undisturbed site with a representative ^{137}Cs activity per unit area, are shortly summarized.

GAMMA-RAY SPECTROMETRY IN THE ENVIRONMENT

In this section a new approach is described for determining several years after a ^{137}Cs deposition the activity per unit area and the gamma dose rate absorbed in air by gamma-ray spectrometry. This new method is based on four ideas:

i) the distribution of cesium in the soil is approximated by the Lorentz function as proposed by Hillmann *et al.* (20);
ii) the peak-to-valley method proposed by Zombori *et al.* (16) is applied to the 661.6 keV peak of ^{137}Cs;
iii) valley corrections are obtained by photon transport simulation calculations as proposed by Hillmann (21); and
ii) the Q-value method proposed by Karlberg (17) is applied to four lines of the uranium decay chain and three lines of the thorium decay chain.

In the peak-to-valley method the difference of the counts in an spectral area below the peak and of background peaks is attributed to the photons which were emitted with the same energy as those in the peak and which were scattered under a small angle. Thus, together with the peak counts, two parameters derived from the spectrum are used to obtain information on the distribution of the radionuclide under consideration in the ground.

The *in situ* measurements were performed with a n-type high purity germanium detector with an efficiency of 25%. The detector was posed at a height of 1 m above ground. Spectra were recorded for exposure times of four hours at six open lawns in Southern Bavaria during measurement campaigns in 1993 (21) and in 1995 (22). ^{137}Cs and ^{134}Cs distributions in the ground of the six sites were measured in 1993 (20). Figure 1 shows an example of a recorded spectrum with an identification of the dominant peaks. The peaks used in the analysis are marked by an arrow, the areas used as valley and background are indicated by shaded regions.

The area around the peak is shown in more detail in Fig.2 with an indication of the valley (620-650 keV), the peak (657-666 keV), and the background area (670-700 keV). In the valley area, A represents the

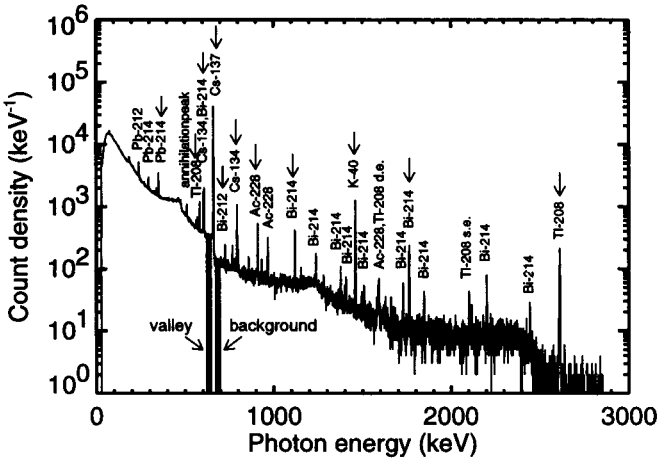

Figure 1. Gamma spectrum recorded in 1995 during four hours at a height of 1 m above ground at a open flat meadow close to Graben in Southern Bavaria. The arrows indicate the lines that were used in the spectrum analysis, the regions the valley and the background for the application of the peak-to-valley method to ^{137}Cs.

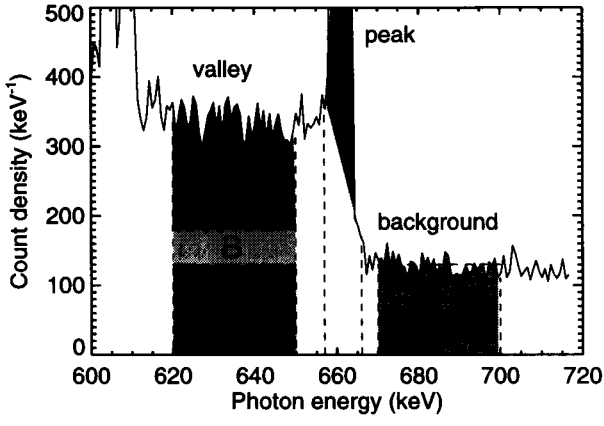

Figure 2. Valley, peak and background counts in the spectrum of Fig. 1 are indicated here by shaded areas. In the valley area, A represents the background component as above the peak, B the additional background due to the inhomogeneous spectrum of the scattered radiation of photons with higher energies, and C scattered photons with an initial energy of 662 keV.

background above the peak, B the addional background counts due to the nonconstant spectrum of scattered radiation from photons with originally higher energies than 700 keV, and C the scattered radiation of photons with an initial energy of 661.6 keV. In the first step of the spectrum evaluation, contributions to the valley, peak and background area by unscattered photons emitted from ^{214}Bi (626.9 keV, 630.8 keV, 633.1 keV, 639.6 keV, 649.2 keV, 661.4 keV, 665.5 keV) and from ^{228}Ac (620.3 keV, 629.2 keV, 640.2 keV, 648.8 keV, 660.4 keV, 666.3 keV, 674.6 keV, 677.3 keV). The magnitude of these contributions is derived from the more expressed peaks of ^{214}Bi and of ^{228}Ac in the spectrum.

The second step in the analysis is the determination of the additional background counts B (22). The distribution of radionuclides in the uranium and thorium decay chains in the ground are approximated by a two layer model. The radionuclides are assumed to be in radioactive equilibrium. Photon transport simulations have been used to calculate the fluences of unscattered photons, and contributions to the background area(670-700 keV) and the valley area (620-650 keV), at a height of 1 m above ground for various of such two layer distributions of each of the two decay chains. As indicated in Fig. 1, four measured lines for the thorium decay chain and three measured lines for the uranium decay chain are than used to determine the effective two layer

distributions of the radionuclides by the Q-value method (17). The corresponding contributions to background and valley area are obtained from the photon transport calculations. For ^{40}K the assumption of a homogeneous distribution was tested with the peak-to-valley method for the 1460.8 keV peak and no significant inconsistency could be found.. Again corresponding contributions to background and valley were obtained from photon transport simulations. A similar procedur was adapted for ^{134}Cs, the distribution of which was approximated by a representative Lorentz distribution. Finally, to take into account cosmic radiation and smaller lines not simulated, the additional background counts B were multiplied by the ratio of measured and simulated background counts. Results are summarized in Tables 1 and 2.

Year and site	Counts in spectral areas			Other lines in			Counts after subtraction in		Peak-to-valley ratio
	Peak	Valley	Back-ground A	Peak	Valley	Back-ground A	Peak	Valley*	
1993:									
GSF	50335	10409	6024	251	64	22	50084	3609	13.9
EGA	31607	7117	4210	219	33	5	31388	2384	13.2
Grafing	94477	14419	6244	182	44	14	94295	7135	13.2
Pullach	62736	12818	7055	182	42	13	62554	4813	13.0
Giglbg.	27363	11658	8617	225	49	15	27138	2146	12.6
Graben	110838	15506	6246	202	42	12	110636	8145	13.6
1995:									
GSF	40100	8738	5097	224	43	11	39876	3120	12.8
EGA	21000	6343	4053	212	32	4	20788	1906	10.9
Grafing	72800	11307	5147	145	30	8	72655	5545	13.1
Pullach	51000	11828	6829	199	51	17	50801	4303	11.8
Giglbg.	25800	11924	9193	248	60	20	25552	1884	13.6
Graben	111000	14142	5413	155	45	17	110845	7963	13.9

* Besides other lines and background A, background B as given in Table 2 was subtracted.

Table 1. Results of the evaluation of *in situ* spectra to obtain the peak-to-valley ratio for ^{137}Cs.

Year and site	^{134}Cs	^{40}K	Uranium decay chain	Thorium decay chain	Cosmic radiat. and others	Total
1993:						
GSF	255	117	162	78	122	734
EGA	146	62	152	49	86	495
Grafing	481	148	123	104	154	1010
Pullach	301	163	140	190	127	921
Giglbg.	100	229	205	202	125	861
Graben	527	130	155	115	158	1085
1995:						
GSF	76	103	156	90	64	489
EGA	38	56	149	33	80	356
Grafing	164	121	102	107	99	593
Pullach	109	151	155	137	110	662
Giglbg.	46	252	200	186	123	807
Graben	276	120	137	86	118	737

Table 2. Contributions to additional background B in the spectra evaluation shown in Table 1.

Site	*In situ* spectrometry 1993	*In situ* spectrometry 1995	Soil samples 1993
GSF	19.9±2.1	20.6±2.4	16.7±0.8
EGA	14.4±1.9	16.9	13.2±0.7
Grafing	42.7±2.8	35.1±2.6	42.6±2.1
Pullach	29.7±2.7	32.8±4.0	27.8±1.4
Giglberg	13.7±3.1	11.8±2.8	11.1±0.6
Graben	46.5±2.7	45.6±2.4	52.2±2.6

Table 3. ^{137}Cs activity per unit area in 1993, as derived in two measurement campaigns by the proposed method of *in situ* gamma-ray spectrometry and by laboratory measurements of soil samples.

From the measured peak-to-valley ratios given in Table 1, effective Lorentz distributions are derived for ^{137}Cs. For this purpose, peak-to-valley ratios for various Lorentz distributions have been derived by photon transport calculations (21) and a correlation between the depth and the width of the maximum of the Lorentz distribution has been assumed (20). From the effective Lorentz distributions, the ^{137}Cs activity per unit area was derived. A thorough analysis of uncertainties was performed (22). Results are summarized in Table 3 and compared with results obtained by soil sampling (20). The dose absorbed in air due to the radiation from ^{137}Cs is determined with a higher accuracy than the activitiy per unit area.

MODELLING OF EXTERNAL EXPOSURES

In the framework of the project 'JSP 5-Pathway analysis and dose distributions*' a model was developed to describe external dose distributions among the Russian population in areas contaminated after the Chernobyl accident (23). The developed model is assumed to be applicable from the year 1990 to the end of the lifetime of people being born before the reactor accident of Chernobyl.

At the time t after a deposition of a cesium isotope, the effective dose rate $E_i(t)$ of a member of a population group i may be calculated by

$$\dot{E}_i(t) = A \cdot \dot{g} \cdot \exp(-\lambda \cdot t) \cdot r(t) \cdot \sum_j f_j(t) \cdot p_{ij}(t) \cdot k_{ij}(t), \qquad (1)$$

where A is the activity deposited per unit area on a reference site, $\dot{g}$ the gamma dose rate in air per activity per unit area with a reference distribution of the cesium in the ground, λ the decay constant, and r(t) the gamma dose rate in air at the reference site divided by the gamma dose rate in air for the reference distribution. The summation index j indicates types of locations, f_j the gamma dose rate in air at a location j relative to the gamma dose in air at the reference site, $p_{ij}(t)$ the relative frequency of stay for members of population group i at location j and $k_{ij}(t)$ the conversion factor from the gamma dose in air to effective dose.

In this report, reference sites are considered to be open fields with undisturbed soil, normally lawns or meadows. A plane source below a soil slab with a mass per unit area of 0.5 g cm^{-2} has been chosen as a reference distribution to approximate the energy and angular distributions of the radiation field in air over an undisturbed field during the first years after the deposition. For this geometry, a value for $\dot{g}$ of 1.7 nGy h^{-1}per kBq•m^{-2} has been obtained for ^{137}Cs and of 4.7 nGy h^{-1}per kBq•m^{-2} for ^{134}Cs (24).

Gamma dose rates in air over open fields

A data base on the attenuation of the gamma dose rate in air due to the migration of cesium into the soil has been established. Main sources of information were (25,26) and new results obtained in the framework of the projects JSP 5* and ECP 5**. The data base has about 450 data sets, each containing besides other information the time of measurement, the value of r(t), and the distance from the Chernobyl nuclear power reactor plant.

The data set indicates the factor r(t) to decrease with increasing distance from the release point. Two factors may be responsible for this observation. First, cesium bound to fuel and condensation particles is known to be less available for migration than cesium that has been evaporated and then attached to the background aerosol. Second, in the data base for short distances, observation points with dry deposition dominate, whereas for long distances observation points with wet deposition dominate. Data grouped in three distance groups are given in Table 4.

Time after deposition (years)	Distance category		
	D ≤ 100 km	100 km < D ≤ 1000 km	D>1000 km
	a_1=0.48; a_2=0.81	a_1=0.60; a_2=0.63	a_1=0.53; a_2=0.51
0.75	1.17 (1.15)	0.98 (1.05)	0.87 (0.88)
2.75	0.88 (0.93)	0.84 (0.78)	0.67 (0.64)
4.75	0.87 (0.82)	0.68 (0.66)	0.55 (0.53)
6.75	0.79 (0.77)	0.60 (0.60)	0.46 (0.48)
24	— (0.59)	— (0.45)	0.39 (0.36)
30	— (0.54)	— (0.42)	0.33 (0.33)

Table 4. Average values for r(t) of entries in data base for the first seven years after Chernobyl and derived from measurements of the atomic weapons tests fallout. The values in parentheses give the results of Eq. (2) with the parameters indicated in the heading of the table.

* Supported by the EC under contract COSU-CT94-0091
** Supported by the EC under contract COSU-CT94-0081

Prediction of the future behaviour of r(t) may be based on observations of the fallout from nuclear weapons tests. An analytical approximation of the data for measurement sites at large distances from the release point and for the nuclear weapons tests in the form

$$r(t) = a_1 \cdot \exp(-\ln 2 \cdot t/T_1) + a_2 \cdot \exp(-\ln 2 \cdot t/T_2), \quad (2)$$

yielded half lifes to be T_1 = 1.5 years and T_2 = 50 years. Based on the observation that fuel particles tend to dissolve over a period of several years in the environment, the same half lifes were assumed for the other distance categories and obtained values for a_1 and a_2 are indicated in Table 4.

The weapons fallout was deposited over long periods and column experiments have shown that under these conditions the initial penetration of the cesium into the soil is negligeable (27). However, the sites in the far field from Chernobyl studied here had different meteorological conditions and an initial migration of the cesium during the wet deposition. It may be assumed that this initial difference retains and that the Chernobyl cesium will have migrated more deeply than corresponding profiles for the weapons fallout at the same time after deposition. Nevertheless both data sets are pooled here, since no better information is available for long times after deposition.

Location factors

In settlements of urban and rural type, the characteristics of the radiation field differ from those over an open plot of virgin land due to shielding and to varying source distributions on the surfaces as a result of different deposition velocities, run-off and weathering. Gamma-ray spectrometry has been used in urban and rural environments in Denmark, Germany and Sweden to obtain extensive results on location factors for ^{131}I, ^{103}Ru, ^{137}Cs, ^{134}Cs, ^{140}La, (28,29). In Russia, location factors have been measured with thermoluminescence detectors in the higher contaminated areas of Ukraine ($A^{Cs137} > 185$ kBq • m^{-2}) in the period 1987-1989 (19). In Russia, first measurements were performed in summer 1989 (30) in three large villages of the Bryansk region with Cs-137 activities per unit area above 1000 kBq/m^2 (Nikolayevka, Yalovka and Svyatsk). During this campaign, several thousand gamma dose rate measurements were performed in different points in the settlements and in their vicinity. The analysis included only data obtained in villages before decontamination actions were taken. Similar measurements were performed in the period 1992 to 1994 in ten villages in the contaminated area of Russia. The results obtained in the two countries were found to be consistent. Location factors measured after 1987 in rural environments were found to be independent of time. Values are given in Table 5.

The presence of a snow cover during the time period November-March reduces according to experimental measurements performed in the Brjansk region in the average the value of the dose rate in air over virgin plots by a factor of 0.76. Since the average reduction over streets may be assumed to be less, location factors during winter time might be a little higher than during summer time. This is confirmed by TLD-measurements of individual doses performed in the same settlements in winter and in summer time. According to these

Location	f_j	p_{ij} (anual)				
		Indoor workers	Outdoor workers	Pensioners	School children	Preschool children
Living areas						
house (wooden)	0.13					
house (brick)	0.07	0.50	0.48	0.68	0.60	0.52
multi storey house	0.02					
outside of houses	0.55	0.21	0.15	0.30	0.24	0.14
Work areas						
buildings	0.07	0.25	0.07	0.0	0.0	0.0
multi storey house	0.02	0.0	0.0	0.0	0.15	0.25
work yard	0.30	0.03	0.07	0.0	0.0	0.09
ploughed field	0.50	0.0	0.15	0.0	0.0	0.0
virgin land	1	0.0	0.04	0.0	0.0	0.0
Rest areas						
forest, meadow	1	0.01	0.01	0.02	0.01	0.0
Group factor	—	0.22/0.19/ 0.17	0.30/0.27/ 0.24	0.27/0.23/ 0.20	0.22/0.19/ 0.16	0.18/0.14/ 0.12

* The first number is for residents in wooden houses, the second in brick houses and the third in multistorey houses

Table 5. Location factors f_j, relative frequencies of stay p_{ij} and group factors $\sum p_{ij} \cdot f_j$ for rural population groups.

measurements, the mean reduction of the annual external dose to snow cover of 0.94 was derived. This small effect is not considered in the following.

Relative frequencies of stay

Information about relative frequencies of stay of the adult rural population at various locations were obtained in Ukraine from 8984 responses of evacuees from the 30 km zone to a questionnaire (19). In Russia, in 1989, 1992, and 1993 corresponding responses of the rural population to questionnaires were obtained and results are summarized in Table 5.

Effective dose per gamma dose in air

The migration of the radionuclides into the soil changes the spectral-angular distribution of the photon fluence exposing the human body. In principle, the value of k_{ij} will change as well. To estimate experimentally conversion factors for a real vertical radionuclide distribution in soil, in summer 1991 and 1992 a series of phantom experiments was carried out. In the experiments three antropomorphical phantoms were used, the Alderson Rando phantom presenting adults and two phantoms of children of 5 and 1 year of age (produced by ATOM Ltd, Riga, Latvian Republic). As experimental sites, two open sites of virgin land, one open ploughed site, and a location inside a wooden house in contaminated areas were chosen.

The experimental results for the three outdoor locations agreed within the limits of only 10% for each of the phantoms. Results of Monte Carlo calculations (31) for the reference distribution (plane source at 0.5 g/cm^2) of radionuclides were found to be intermediate between the experimental results for outdoor and indoor locations. In the present model a value of 0.9 Sv•Gy^{-1} for preschool children (0-7 years), of 0.8 Sv•Gy^{-1} for school children (8-17 years) and of 0.75 Sv•Gy^{-1} for adults is assumed for all locations.

Model results

Annual effective doses of rural indoor workers living in woodframe houses are shown in Fig. 3. For comparison results of the UNSCEAR model (32) are given approximating the attenuation due to the cesium migration into the soil by a constant factor. Dose estimates for Russian settlements in 1991 (2) agree within 10 % with the current model. In the present model for the period 1990-2056, annual effective doses of different population groups differ by a constant factor. The factors for rural population groups are given in Table 6. Urban population groups have lower external exposures due to the Chernobyl accident (33). According to Table 6 rural outdoor workers living in woodframe houses are the critical group for external exposures. Among them, forestry workers and herdsmen receive the highest dose (33).

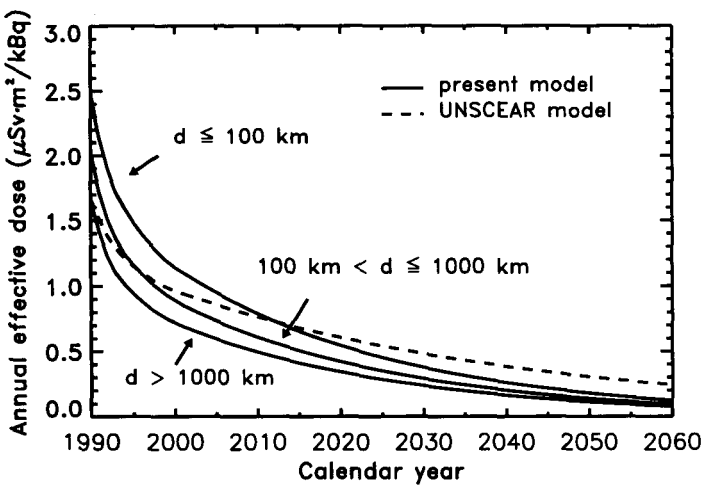

Figure 3. Annual effective doses due to external exposures of rural indoor workers living in woodframe houses. Results are normalized to the ^{137}Cs activity initially deposited per unit area, a ratio of the ^{137}Cs activity to the ^{134}Cs activity of 1.8 has been assumed. The dotted line represents the UNSCEAR model (32).

Population group	Wooden houses	Brick houses	Multi storey houses
Indoor	1.00	0.86	0.77
Outdoor	1.36	1.23	1.09
Pensioners	1.23	1.05	0.91
Schoolchildren	1.07	0.92	0.78
Preschoolchildren	0.98	0.76	0.65

Table 6. Annual effective doses due to external exposures of rural population groups, normalized to the annual effective dose due to external exposures of rural indoor workers living in woodframe houses.

Comparison with measurements of individual external doses

Monthly external doses were measured with TLDs during the spring-summer periods of 1990-1993 in 21 settlements of the Bryansk region with average ^{137}Cs activities above 0.4 MBq/m^2 (34). The conversion factor from readings of an individual dosimeter to the value of the effective dose was determined on the basis of the results of phantom experiments, and it was 0.9 Sv/Gy for adults, 0.95 Sv/Gy for schoolchildren and 1 Sv/Gy for preschoolchildren. Table 7 shows a good agreement between the average doses in the population groups obtained by the model and by measurements. A comparison with measured individual doses showed some of those to be larger than the largest of the average doses in the population groups. Doses to individuals are not the scope of the present model.

Year	Monthly effective dose (μSv)					
	Wood houses			Brick houses		
	Model	Measurement	Ratio	Model	Measurement	Ratio
1990	290	285(174)	1.02	271	230(100)	1.18
1991	246	235(414)	1.05	230	205(103)	1.12
1992	216	200 (95)	1.08	202	200 (49)	1.01
1993	195	150(195)	1.30	182	115 (84)	1.58
Average			1.11			1.22

Table 7. Monthly effective doses (μSv) of outdoor workers in the Bryansk region in the periods (April-November) of the years 1990 to 1993. Results of the present model and of TLD measurements (34) are given, the figures in parentheses give the number of measurements.

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EVALUATION OF THYROID DOSES FROM RADIOIODINE RELEASES

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INTRODUCTION

Iodine is a volatile element which is very mobile in the environment. Iodine enters the metabolism of living organisms and is selectively taken up and concentrated in the thyroid gland; it plays a major role in the synthesis of the thyroid hormone and is partly secreted in milk.

There are at least 25 iodine isotopes with mass numbers ranging from 117 to 141. All except ^{127}I are radioactive. Omitting the very short-lived ^{140}I and ^{141}I, thirteen isotopes are produced by fission: ^{127}I (stable), ^{128}I (25 min), ^{129}I (1.57×10^7 a), ^{130}I (12.4 h), ^{131}I (8.06 d), ^{132}I (2.3 h), ^{133}I (21 h), ^{134}I (52.8 min), ^{135}I (6.7 h), ^{136}I (83 s), ^{137}I (23 s), ^{138}I (5.9 s), and ^{139}I (2 s). From the point of view of environmental contamination and resulting doses to man, the most important isotope of iodine is ^{131}I, with minor roles occasionally played by the short-lived ^{132}I and ^{133}I, and by the long-lived ^{129}I. Because of its half-life of approximately 8 d, ^{131}I is a short-term problem: within two months after its environmental release, ^{131}I will have decayed to insignificant levels.

Iodine-131 is mainly found in the environment as a result of nuclear explosions and of airborne releases from nuclear reactors and fuel reprocessing plants. Thyroid doses are, in most cases, mainly due to the consumption of ^{131}I-contaminated fresh cows' milk, with minor contributions resulting essentially from the consumption of leafy vegetables and other foodstuffs with short shelf life, and from inhalation. Because of the smaller mass of their thyroid gland, children receive higher doses than adults for a given intake of ^{131}I. Atmospheric releases of, and thyroid doses from, ^{131}I from the nuclear industry are relatively well documented (1-2):

- Testing of nuclear weapons in the atmosphere, which took place essentially in the early 1960s, caused the release of about 650 000 PBq of ^{131}I (2). The major part of this activity was produced in high-yield tests, and was carried upwards into the stratosphere, from which it descended to ground level in a matter of months or years. Consequently, little ^{131}I was available to deliver doses to man. Collective thyroid doses from global fallout of ^{131}I are estimated to have been about 3.3×10^6 man Gy, corresponding to a per caput thyroid dose to the world population of about 1 mGy. These figures, however, do not take into account the local and regional collective thyroid doses from the early, low-yield, tests that were conducted in the 1950s. Some of these low-yield tests were carried out at the Nevada Test Site in the United States, mainly from 1951 to 1957; they caused a release of about 5 500 PBq of ^{131}I in the atmosphere. Because the explosive energy of those tests was relatively small, most of the ^{131}I produced remained in the troposphere and was rapidly available to contaminate human foodstuffs. A preliminary estimate of the collective thyroid dose to the population of the contiguous United States from the tests conducted at the Nevada Test Site is about 4×10^6 man Gy, corresponding to an average thyroid dose of about 20 mGy for that population (3). The thyroid doses received by the populations living in the vicinity of the Nevada Test Site have been studied in detail (4-6). Individual thyroid doses to children were estimated to range up to 4 600 mGy (5). It should also be noted that the test Bravo, which consisted in the detonation in March 1954 of a high-yield device mounted on a barge over the reef of Bikini Atoll in the mid-Pacific Ocean, resulted in intense local and regional fallout and in the evacuation of about 300 people within 3 days after the detonation as well as the exposure of Japanese fishermen on board the Lucky Dragon (7-8). The estimated thyroid doses which, in this case, were mostly due to short-lived radioiodines, ranged up to 20 000 mGy (9).

- The operation of nuclear reactors leads to relatively small releases of ^{131}I under normal conditions; it is estimated that the total release of ^{131}I from reactors used worldwide for the production of electrical energy up to 1989 was 0.045 PBq and that the corresponding collective thyroid dose to the local and regional populations residing around the reactors is 460 man Gy (2). However, much greater atmospheric releases and doses resulted from the Windscale accident of 1957 and from the Chernobyl accident of 1986. About 0.7 PBq of ^{131}I was released during the Windscale accident (10); the collective thyroid dose was approximately 20 000 man Gy (11), with individual thyroid doses ranging up to 100 mGy (12). During the Chernobyl accident, about 330 PBq of ^{131}I were released into the atmosphere (13); a preliminary estimate of the total collective thyroid dose received by the populations

residing in the contaminated areas of the former Soviet Union is about 1.2×10^6 man Gy (14); individual thyroid doses up to 10 000 mGy and more are estimated for children in the most affected areas (15). In comparison to those reactor accidents, the reactor accident of Three Mile Island in 1980 was minor with an estimated atmospheric release of ^{131}I of 0.00055 PBq (16-17).

- The operation of plants that reprocess nuclear fuel for civilian purposes results in small releases of ^{131}I because the nuclear fuel originating from the reactor is stored before reprocessing for a time long enough to ensure that ^{131}I has decayed to low levels. It is estimated that the total release of ^{131}I from worldwide civilian fuel reprocessing plants was 0.004 PBq up to 1989 and that the corresponding collective thyroid dose to the local and regional populations residing around the fuel reprocessing plants is 30 man Gy (2). However, much greater atmospheric releases and doses resulted from the operation of plants reprocessing nuclear fuel for military purposes during the early days of the nuclear age when only a short storage time was allowed before reprocessing. Releases of ^{131}I at Hanford from 1947 to 1949 amounted to approximately 25 PBq (18) and a similar figure has been advanced for the releases at Chelyabinsk (19). Assuming a collective thyroid dose of 10^{-11} man Gy per Bq of ^{131}I activity released (2), the collective thyroid doses resulting from the releases from each plant would be about 2×10^5 man Gy.

The evaluation of the thyroid doses from radioiodine releases depends on the type of monitoring data that is available. The preferred types of data are, in order of decreasing usefulness: (a) direct thyroid measurements, (b) concentrations in air, milk, and leafy vegetables, (c) depositions per unit area of ground, and (d) atmospheric releases. The use of each type of data will be illustrated in turn.

EVALUATION OF THYROID DOSES BASED ON DIRECT THYROID MEASUREMENTS.

In the case of the nuclear reactor accident that occurred at Chernobyl in April 1986, the content of ^{131}I in the thyroid of people living in the highly contaminated areas of the former Soviet Union was large enough that the gamma radiation emitted by the thyroid as a result of the radioactive decay of ^{131}I could be readily measured with a radiation detector held in the vicinity of the neck. Several hundred thousands people were monitored in this way in Ukraine, Belarus, and Russia within a few weeks after the accident (15; 20-21). The measured exposure rates were then converted into thyroid doses using appropriate assumptions on the dynamics of intake of ^{131}I both before and after the direct thyroid measurement. Since few measurements of ^{131}I in air and foodstuffs were made after the accident in former Republics of the Soviet Union, the dynamics of intake of ^{131}I was largely based on environmental transfer models. Because the ingestion of ^{131}I-contaminated milk was for the majority of individuals exposed the main contributor to the ^{131}I intake (22), the intake function was in most cases represented by the time dependence of the milk contamination following a single deposition.

In the three Republics of the former Soviet Union with highly contaminated areas (Ukraine, Belarus, and Russia), the databases on thyroid doses derived from direct thyroid measurements were used to infer the thyroid doses in areas with few, or no, direct thyroid measurements. In each of the three countries, relationships were sought between the thyroid dose and the ^{137}Cs deposition, which has been extensively measured throughout the contaminated areas:

- in the Chernigov region of Ukraine, the following empirical relationships were obtained (15):

  $D(T) = K \times a^{\exp(-b \times T)}$ [1]

  where: D(T), in cGy, is the thyroid dose for age T (y),
  a (unitless) is a parameter representing the thyroid dose at age 0
  b (y^{-1}) is a parameter describing the age dependence of the thyroid dose
  K (cGy) is a scaling parameter characterizing the radioiodine intake, calculated according to:

  $\log(K) = 640 \times (\log(\sigma(^{137}Cs)) / \rho^2) + 2.7 \times \cos(\phi) + 0.013 \times \rho \times \sin(\phi) - 1.6$ [2]

  where: $\sigma(^{137}Cs)$ is the ^{137}Cs deposition at the location considered,
  ρ (km) and ϕ are the polar coordinates of that location with respect to the Chernobyl site.

- in Belarus, a relationship has been established between the average thyroid dose received by people in rural settlements and the ground-deposition density of radionuclides (^{137}Cs or ^{131}I) in this settlement and in the area around the settlement (20). For the settlements in Hoiniki in Gomel oblast, and Kostukovichi and Krasnopolye in Mogilev oblast:

D_{jx} (adult) = $3.5 \times 10^{-8} \times \sigma_x(^{131}I) + 1.4 \times 10^{-8} \times \sigma_{jx}(^{131}I)$ [3]

$= 3.5 \times 10^{-8} \times R_x \times \sigma_x(^{137}Cs) + 1.4 \times 10^{-8} \times R_{jx} \times \sigma_{jx}(^{137}Cs)$ [4]

where: D_{jx} (adult) is the arithmetic mean thyroid dose, in Gy, for adult population in settlement, j, in area, x, in the absence of any countermeasures in the settlement and for typical lifestyle and dietary habits,

$\sigma_x(^{131}I)$, $\sigma_x(^{137}Cs)$ is the average ground-deposition density, in Bq m^{-2}, of ^{131}I (^{137}Cs), in area, x,

$\sigma_{jx}(^{131}I)$, $\sigma_{jx}(^{137}Cs)$ is the average ground-deposition density, in Bq m^{-2}, of ^{131}I (^{137}Cs), in the settlement, j, in area, x,

R_x, R_{jx} is the average ratio of the ^{131}I to ^{137}Cs ground-deposition densities in area, x, or in the settlement, j, in area, x.

It can be expected that, on average in an area, $\sigma_{jx}(^{131}I) = \sigma_x(^{131}I)$, so that the average relationship between ^{131}I deposition density and the average adult thyroid dose is:

$D_x(adult) = 4.9 \times 10^{-8} \times \sigma_x(^{131}I)$ [5]

Ongoing research may clarify whether equations 3 and 4 are generally applicable to all contaminated areas of Belarus. It is likely that correction coefficients taking into account the role of fuel particles, and parameters such as the standing crop biomass, the fraction of cow's intake from pasture grass, and the date of the beginning of cow's pasture will need to be factored in (20).

- in Russia, Zvonova and Balonov (21) analyzed the thyroid measurements and radiation survey data for the Bryansk and Tula oblasts. The thyroid doses derived from direct thyroid measurements for 3 to 6 years old children were found to be correlated with the ^{137}Cs ground-deposition density, the kerma rate in air on May 10-12, 1986 (i.e, 14 to 16 days after the beginning of the accident), the mean ^{131}I concentration in milk in the period May 5-12, 1986, and the average radiocesium content in adults in July-August, 1986. With respect to the ^{137}Cs ground-deposition density, the following relationship was obtained:

  $D(3\text{-}6) = (76 \times 10^{-8}) \times \sigma(^{137}Cs)$ [6]

  where: D(3-6) is the average thyroid dose for 3 to 6 years old children (Gy), and
  $\sigma(^{137}Cs)$ is the ^{137}Cs ground-deposition density (Bq m^{-2}).

  According to Zvonova and Balonov (21), the average ratio of the deposition densities of ^{131}I and ^{137}Cs was approximately 8 immediately after the period of intense deposition, so that:

  $D(3\text{-}6) = (9.5 \times 10^{-8}) \times \sigma(^{131}I)$ [7]

It should be noted that the measurement of the gamma radiation emitted by the thyroid with a radiation detector held in the vicinity of the neck is not the only method available to derive the thyroid dose from a direct measurement on man. Measurements of ^{131}I in urine also have been used for that purpose, notably to estimate the thyroid doses received by the Marshallese as a result of the test Bravo, in 1954 (9).

EVALUATION OF THYROID DOSES BASED ON MEASUREMENTS IN FOODSTUFFS AND IN AIR.

For regions other than the most contaminated areas of the former Soviet Union, the evaluation of the thyroid doses resulting from the Chernobyl accident was based on measurements in foodstuffs and in air. In Russia, Kryshev (22) made use of the detailed data of radioecological monitoring that were available for Sosnovy Bor near Leningrad to infer that the thyroid doses from inhalation were much lower than those due to ingestion and that the main contributors to the thyroid dose received by children were the consumption of milk, followed by the consumption of water, vegetables, bread, and home-baked goods. The following empirical relationship was derived (22):

$D(T) = K(T) \times R(T) \times \sigma(^{131}I)$ [8]

where: D(T) is the thyroid dose, in Gy, for age, T, in years,
R(T) is the thyroid dose coefficient, in Gy Bq^{-1},
$\sigma(^{131}I)$ is the ^{131}I deposition density, in Bq m^{-2}, derived from the ^{137}Cs ground-deposition density, and calculated for 15 May 1986 (about 17 days after the likely time of highest deposition),
K(T) is the ^{131}I transfer coefficient from ground deposition to intake by people of age, T, in Bq per Bq m^{-2}.

The following values of K(T) were obtained by Kryshev (22):

T (y)	1	5	10	15	adult
K (Bq per Bq m^{-2})	0.28	0.32	0.36	0.36	0.47

When recalculated to reflect the transfer coefficient related to the total ground deposition of ^{131}I, assumed to have occurred on 28 April 1986, the values of K(T) become:

T (y)	1	5	10	15	adult
K (Bq per Bq m^{-2})	0.065	0.074	0.083	0.083	0.11

On the basis of the variation of K with age shown above, using the thyroid dose coefficients recommended by ICRP (23), and population fractions of 0.02, 0.16, 0.20, and 0.62 for the age groups around 1 y, 5 y, 10 y, 15 y, and for adults, respectively, a transfer coefficient from ^{131}I deposition density to per caput thyroid dose of 7×10^{-8} Gy per Bq m^{-2} is obtained.

The same parameter values can also be used to derive the per caput transfer coefficients from the adult values for Belarus (equation 5) and from the 3 to 6 years old children for Russia (equation 7; data from Zvonova and Balonov). The per caput transfer coefficients obtained in that way are 7×10^{-8} Gy per Bq m^{-2} for Belarus and 5×10^{-8} Gy per Bq m^{-2} for Russia (data from Zvonova and Balonov).

The ^{131}I concentrations in milk and in leafy vegetables that were measured in most countries of the Northern hemisphere after the Chernobyl accident were used by the UNSCEAR Committee to estimate in a consistent manner the thyroid doses received by the populations of those countries (13). The ^{131}I concentrations in leafy vegetables were found to be, on average, 1.5 times greater than those in milk. However, when the relative consumption rates of milk and leafy vegetables are taken into account, the ^{131}I intakes from milk were, on average, much greater than those from leafy vegetables, especially in the countries of Northern, Central, and Eastern Europe that were most affected by the Chernobyl accident.

Measurements of ^{131}I concentration in milk were also used by the UNSCEAR Committee to estimate the thyroid doses due to global fallout from the nuclear weapons tests conducted in the atmosphere, mainly in the 1960s (2; 17). On the basis of measurements in Argentina (24), the relationship between ^{131}I ground-deposition density, $\sigma(^{131}I)$ in Bq m^{-2}, and time-integrated milk concentration, C_m in Bq a L^{-1}, was estimated to be:

$$C_m = 6.3 \times 10^{-4} \ \sigma(^{131}I) \qquad [9]$$

A transfer coefficient from ^{131}I deposition density to per caput thyroid dose of 8×10^{-8} Gy per Bq m^{-2} is derived from equation 9 using the values adopted by UNSCEAR (2) of 0.3 L d^{-1} for the average milk consumption rate and 1.2×10^{-6} Gy Bq^{-1} for the age and milk consumption weighted thyroid dose coefficient.

EVALUATION OF THYROID DOSES BASED ON MEASUREMENTS OF GROUND DEPOSITION.

The importance of the deposition - pasture grass - cow's milk pathway to man as a major dose contributor in the case of an atmospheric release of ^{131}I was not fully recognized until 1957. Therefore, the dose reconstruction efforts related to radioiodine releases prior to 1957 are largely based on measurements of radionuclides other than ^{131}I (or of total beta measurements or exposure rates) as well as on environmental transfer models.

Atmospheric testing of nuclear-weapons-related devices at the Nevada Test Site (NTS) began in 1951; most of the atmospheric releases of radioactive materials, including ^{131}I, took place in test series conducted in 1951, 1953, 1955, and 1957. In the Off-Site Radiation Exposure Review Project of the U.S. Department of Energy, doses were estimated for the "local" populations (less than 800 km from the NTS); the key data used to reconstruct thyroid doses from ^{131}I releases were estimates of ground-deposition densities of ^{131}I from each location of interest and each important test, which were derived either from the available exposure rates or from contemporary measurements of ^{137}Cs and $^{239,240}Pu$ in soil, combined with data on the relative abundance of radionuclides produced in each test (25-27). The transfer coefficients representing the ^{131}I intakes per unit deposition density were estimated to vary according to the fallout date and to the age and sex of the person considered (28). Examples of results, averaged over both sexes, for a date with low transfer coefficient (17 March) and for

a date with high transfer coefficient (24 July) are (28):

Age (y)	<1	1-11	12-18	Adult
Transfer coefficient (Bq per Bq m^{-2}): fallout on 17 March	0.0044	0.0092	0.013	0.016
Transfer coefficient (Bq per Bq m^{-2}): fallout on 24 July	0.17	0.14	0.13	0.08

The per caput thyroid doses per unit deposition density, calculated using the thyroid dose coefficients recommended by ICRP (23), are found to vary from 9×10^{-9} to 9×10^{-8} Gy per Bq m^{-2} according to the date of fallout.

The U.S. National Cancer Institute also is carrying out a study related to ^{131}I releases from the NTS; that study includes the estimation of thyroid doses received by populations across the continental United States (29). According to preliminary estimates, the amount of ^{131}I that was deposited throughout the continental United States as a result of the weapons tests conducted at the NTS is 1.5 EBq (over an area of 7.7×10^6 km^2) and the per caput thyroid dose was about 20 mGy, corresponding to an average thyroid dose per unit deposition density of about 10^{-7} Gy per Bq m^{-2} (3).

EVALUATION OF THYROID DOSES BASED ON MEASUREMENTS OF AIRBORNE RELEASES.

There are cases where human or environmental radiation monitoring data are not available, either because the radioiodine releases, although important, occurred during the early days of the nuclear age, when environmental monitoring was not a priority, or because the radioiodine releases are so low that they cannot be readily detected in the environment. In the absence of human or environmental monitoring data, the evaluation of the thyroid doses is based on estimates of radioiodine releases.

For example, the ^{131}I releases from the Hanford Site in the United States for the years 1945 to 1951 were estimated on the basis of the operating histories of facilities on the site and on the knowledge of the effluent control technologies in use at the time of release (18). Detailed thyroid dose estimates were then prepared for 1102 locations around the site, 12 different representative individuals, distinguished by age and gender, and a series of food source scenarios, by means of environmental transfer models and data on milk production, distribution and consumption, among other factors (30).

Routine reactor releases have been usually so low in recent years that the environmental concentrations are not detectable and the resulting thyroid doses can only be calculated by means of environmental transfer models. The UNSCEAR Committee, which compiles every few years the data available on routine releases from nuclear power plants and other nuclear facilities (2; 13; 17), uses a model site and model calculations to evaluate the collective doses resulting from routine releases of radionuclides; the collective thyroid dose per unit of ^{131}I activity released is estimated by the UNSCEAR Committee to be about 10^{-11} man Gy per Bq (2; 13; 17).

DISCUSSION.

General information on the major sources of radioiodine releases into the atmosphere from the nuclear industry is summarized on Table 1. Most of those releases occurred more than 30 years ago. The nuclear weapons tests that were conducted in the atmosphere account for most of the ^{131}I that was released into the environment and for most of the resulting collective thyroid dose.

The thyroid doses were evaluated using different methods according to the types of radiation data that were available. Table 2 presents values of collective thyroid doses normalized per unit of activity released and of average thyroid dose normalized per unit deposition density. As expected, the normalized collective doses are much greater for ground-level releases than for the nuclear weapons tests, which carried the radioactive materials that they produced to high altitudes. The average thyroid doses per unit deposition density are remarkably similar, given the variety of methods and models that were used to estimate that quantity. It should be kept in mind, however, that for a given deposition density, the individual thyroid doses vary by orders of magnitude according to the date of fallout, agricultural practices, the age of the person considered and dietary habits.

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Table 1. Major sources of radioiodine releases into the environment.

	Activity released (PBq)	Year(s) of release	Collective thyroid dose (man Gy)	Main basis for thyroid dose evaluation
MILITARY NUCLEAR FUEL CYCLE				
Atmospheric nuclear weapons tests:				
• Global fallout	650 000	1962-1963 (mainly)	3.3×10^6	^{131}I deposition densities and concentrations in milk
• Local and regional fallout (Nevada Test Site)	5 500	1951-1957 (mainly)	4×10^6	^{131}I deposition densities
Reactors: • Windscale accident	0.7	1957	2×10^4	^{131}I release and concentrations in milk
Fuel reprocessing plants (routine releases):				
• Hanford	25	1945-1948 (mainly)	2×10^5	^{131}I release
• Chelyabinsk	about 20	1949-1951 (mainly)	2×10^5	^{131}I release
CIVILIAN NUCLEAR FUEL CYCLE				
Reactors:				
• Routine releases	0.045	up to 1989	460	^{131}I release
• TMI accident	0.00055	1980	small	^{131}I release
• Chernobyl accident	330	1986	1.2×10^6	Direct thyroid measurements and ^{137}Cs deposition densities

Table 2 - Transfer coefficients

COLLECTIVE THYROID DOSE PER UNIT ACTIVITY RELEASED (man Gy per Bq)	
Global fallout from weapons tests	5×10^{-15} (world population)
Local and regional fallout from weapons tests (Nevada Test Site)	7×10^{-13} (U.S. population)
Reactor accident at Windscale	3×10^{-11}
Reactor accident at Chernobyl	4×10^{-12}
Routine releases at ground level	1×10^{-11}
AVERAGE THYROID DOSE PER UNIT DEPOSITION DENSITY (Gy per Bq m^{-2})	
Global fallout from weapons tests	8×10^{-8}
Local and regional fallout from weapons tests (Nevada Test Site) (3)	1×10^{-7} (U.S. population)
Local fallout from weapons tests (Nevada Test Site) (28)	from 9×10^{-9} to 9×10^{-8} (local population)
Reactor accident at Chernobyl: Belarus (20)	7×10^{-8}
Reactor accident at Chernobyl: Russia (21)	5×10^{-8}
Reactor accident at Chernobyl: Russia (22)	7×10^{-8}

THE INFLUENCE OF ICRP ON EUROPEAN LEGISLATION CONCERNING MEDICAL RADIATION EXPOSURE

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INTRODUCTION

There are two fundamental Directives in force in Europe concerning radiation protection. Staff and public protection is covered by a Basic Safety Standards Directive, the current version of which was enacted in 1980 (1,2), following the recommendations of the International Commission on Radiological Protection (ICRP) published as ICRP 26 (3). In 1984, a European Directive specifically covering patient protection was enacted for the first time (4) and is still in force.

In 1991, the ICRP published general recommendations as reported in ICRP 60 (5) updating their previous general recommendations (3) of 1977. Since its publication, ICRP 60 has been considered within the European frame and new Directives have been negotiated to include the latest concepts. These two Directives, which will replace the present ones, are at different stages of adoption but the process should be complete for both within eighteen months.

The aim of this paper is to describe the input of the Article 31 Group of the European Commission to the process of amending European radiation protection legislation and to detail the changes which have been incorporated, which should improve patient protection in the medical diagnostic and therapeutic environment in accordance with ICRP recommendations.

THE LEGISLATION AMENDMENT PROCEDURE

The group of scientific experts in public health referred to in Article 31 of the Euratom Treaty (Article 31 Group) and appointed by the Member States of the European Union (EU), advises the European Commission on matters concerning radiation protection within the European Union. In 1993, the Article 31 Group proposed revisions to the Basic Safety Standards Directive to take into account the recommendations of ICRP 60. Their proposal has been amended on its journey through the European legal process and is currently in the final stages of approval (6).

In 1994, the Article 31 Group set up a working party on Medical Exposures which had as its members, radiation physicists and medical practitioners from radiotherapy, nuclear medicine and radiology. They were asked to review the present Directive on patient protection and the experience gained in its implementation in the Member States. The working party considered the developments in the medical uses of ionizing radiation, the progress in radiation protection epitomized by the recommendations of ICRP and relevant aspects of the proposed Basic Safety Standards Directive. The outcome of this work was a revised patient protection Directive which was then discussed and revised by the Article 31 Group.

The final opinion of the Article 31 Group supporting this revision of the current patient protection Directive was reached at a meeting on 31 May 1995. This revised Directive on "health protection of individuals against the dangers of ionizing radiation in relation to medical exposures" (hereinafter referred to as the Patient Protection Directive or PPD) was submitted last Autumn to the European Commission for adoption. The schedule now is that the PPD goes to the Economic and Social Committee of the EU (which is the procedure laid down by Article 31) in April 1996. In the Summer it will then be introduced to the European Council who will seek the opinion of the European Parliament this Winter. If all goes well, the new Directive will be adopted by the European Council next Summer, that is July 1997.

Each Member State in the European Union will then have a period, probably up to 1999, to produce enacting legislation to implement the Directive. In the UK for example, this will take the form of revision to the Regulations on patient protection [7,8].

The protection of workers, including medical and paramedical staff, and of members of the public such as visitors to medical centres, is not affected by the Directive on medical exposures but is ensured by the Basic Safety Standards Directive (6).

THE INFLUENCE OF ICRP RECOMMENDATIONS

It is perhaps not surprising that the recommendations of the ICRP are incorporated into European legislation comparatively quickly. On the Article 31 Group, there are three members of ICRP Committee 3 (all of whom are members of the Medical Exposures working party), members of the three other ICRP Committees and, indeed, the Chairman of the Main Commission. All the members of the Article 31 Group either work directly with radiation (necessarily not just in the medical field), or have a role in radiation protection in their Member State.

Although the general recommendations of ICRP 60, where they relate to medical exposures, have been taken into account during the discussions on the Article 31 Group, other ICRP publications relating to medical exposures have obviously influenced the outcome of the proposals for revision which produced the new PPD. One consequence of which is that the proposed Patient Protection Directive has been enlarged in scope, to include both volunteers in research and, individuals, such as close family or friends, helping voluntarily in the support and comfort of patients (often referred to as comforters and carers).

The 1984 Directive had already incorporated the principles of justification and optimisation as they relate to medical exposures. The new PPD has several specific requirements which reinforce these concepts such as referral criteria and the use of reference levels for medical diagnostic exposures. There is published evidence that by applying the principles of justification and optimization, exposures can be reduced in a cost-effective manner without losing the efficacy of the related practice [9,10].

The concept of potential exposures to patients, together with the requirement of keeping their probability and magnitude as low as reasonably achievable, is specifically included in the PPD.

An important requirement of the current Directive is that any use of ionizing radiation in medical procedures is effected under the responsibility of a practitioner (this term being clearly defined within the Directive as relating to medical, dental of other health professional entitled to take clinical responsibility within their Member State). The new PPD requires that the overall responsibility remains with the practitioner but to provide support, practical aspects of the medical procedure could be delegated to other individuals as authorized by the competent national authorities. The requirements for the training of practitioners and these other individuals are now explicit, including continuing education and training, which should lead to improvements in practice.

The PPD expands the existing requirements on quality control of radiological installations and supplements them by requiring the establishment of quality assurance programmes which also include assessment of doses received by patients. Such programmes (10) have been demonstrated to reduce individual exposures (11). Health and safety requirements including radiation protection aspects regarding the design, manufacture, placing on the market of the medical devices used are actually covered by another Council Directive 93/42/EEC concerning Medical Devices (12).

Special requirements are included for sub-sets of individuals who are deemed to be at particular risk from radiation exposure. These include paediatric patients, asymptomatic individuals entering health screening programs, patients undergoing interventional radiology or high exposure studies such as computed tomography, pregnant patients, breast feeding patients and comforters and carers.

In the last part of the PPD, to monitor progress, Member States are requested to ensure the establishment of processes for auditing the implementation of the requirements of the Directive, and to make estimates of individual and collective doses from the various medical practices.

ICRP Committee 3 (Medical radiation) is currently producing reports on the practical aspects of implementing the ICRP recommendations. These are being designed to help, not just the legislators

and practitioners within European Member States but those in every country, to apply the principles of radiation protection every day as a natural extension of their medical radiation practices.

CONCLUSION

The objective of the 1984 Euratom Directive on patient protection was to reduce medical radiation exposures to the population without jeopardizing the benefits, whether these were early recognition of disease, diagnosis or therapy. With the continuing expansion of medical practices using ionizing radiation, measures need to be taken to reduce individual exposures so that collective effective dose is limited.

The revised Patient Protection Directive maintains the previous objective and provides further detail, founded on practical experience and ICRP reports, as a base for European Member States to develop their own legislation. This should result in improved use of radiological practices so that the population can continue to receive the benefits of medical exposures while the collective effective dose is better controlled.

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THE ROLE AND DETERMINATION OF PATIENT DOSE IN X-RAY DIAGNOSIS

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ABSTRACT

In radiation protection of the patient in x-ray diagnosis all three principles of radiation protection should be applied. So-called dose constraints which limit entrance surface doses ensure implicitly that patient doses should not exceed certain levels. With respect to justification, it is believed that there is a large potential for patient dose reduction by avoiding both clinically unjustified examinations and unnecessary repetition of diagnostic procedures. While this appears to be quite straightforward, the strategy for optimisation is more complicated. Here, a reasonable compromise between high image quality and low patient dose has to be found, as often measures aimed at improved image quality lead to an increase of patient dose, and, vice versa, measures aimed at a reduction of patient dose result also in reduced image quality.

Whereas the problem to quantitatively assess the quality of a given image is still not solved satisfactorily, the determination of patient doses has become increasingly feasible in recent years. For this purpose, computer codes, often based on Monte Carlo techniques, simulating the radiation transport in material are commonly used together with computational models of the human body. Most of the computational body models in use are so-called mathematical models, that means, mathematical expressions representing simple geometrical bodies are used to describe idealised arrangements of body organs. Additionally, tomographic models were developed in recent years which use computed tomographic data of real persons to provide three-dimensional representations of the body.

Using these computational models of the human body, numerous studies concerning organ and tissue doses from diagnostic radiology were performed. Although it is not recommended to apply the calculated doses to assess individual patient doses, the influence of single exposure conditions as, e.g., tube voltage, filtration, field size and location, focus-to-skin distance, on organ and tissue doses can be studied readily, thus resulting in information prerequisite for optimisation in x-ray diagnosis. Additionally, the tomographic models enable to assess the influence of moderate variations of the patient size on organ doses and, therefore, improve to a certain extent the applicability of literature data on patient doses to individuals.

INTRODUCTION

Because most procedures causing medical radiation exposures are clearly justified and because the procedures are usually for the direct benefit of the exposed individuals, less attention has been given to the optimisation of protection in medical exposures than in most other applications of radiation sources (1). On the other hand, it is a well-known fact that the radiation doses from diagnostic radiology are the largest contribution to the collective dose from all man-made sources of radiation (2). From this, it is obvious that diagnostic radiology should be of major concern for radiation protection and that, consequently, the guidelines established by

the ICRP for occupational radiation protection should be applied also to diagnostic radiology as far as possible.

(1) Justification:
In relation to the justification of examinations in diagnostic radiology it is accepted that most examinations result in information beneficial to the patient and that, consequently, the benefit of these examinations will, in general, by far outweigh the radiation detriment. It is, nevertheless, believed that there is a large potential for patient dose reduction by avoiding both clinically unjustified examinations and unnecessary repetition of diagnostic procedures. Furthermore, it should also be considered whether imaging techniques not involving ionising radiation could be applied if they result in the same diagnostic benefit.

(2) Optimisation:
Optimisation in diagnostic radiology commonly involves two aspects: the first is to establish quality assurance and quality control programmes to ensure a proper performance of the x-ray equipment; the second is the necessity to find a reasonable compromise between high image quality and low patient dose, as often measures aimed at improved image quality lead to an increase of patient dose, and, vice versa, measures aimed at a reduction of patient dose result also in reduced image quality.

(3) Limitation:
It is generally accepted that there is little use of dose limits in diagnostic radiology, as there are large ranges of doses due to the different complexity of the situations considered. It is, however, of concern that dose differences of up to two orders of magnitude for the same type of examination have been reported in diagnostic radiology (1). Therefore, more and more consideration is given to dose constraints for application in some common diagnostic procedures (3-5). These should be applied with sufficient flexibility to allow higher patient doses where indicated by sound clinical judgement.

Whereas for the justification of diagnostic radiological procedures patient doses do not play an important role, they are of major consequence for optimisation and limitation. Furthermore, the necessity to determine patient doses may arise routinely due to legal regulations (as, e.g. the German X-ray Ordinance) or in special cases, e.g. due to possible legal consequences of an individual examination. The various dose quantities considered for these specialities will be characterised in the following, and methods to determine these dose quantities will be described.

THE ROLE OF PATIENT DOSES IN X-RAY DIAGNOSIS

There are several aspects in x-ray diagnosis where patient doses are considered:

(1) Dose constraints:
A strict limitation of doses to patients comparable to the practice in other fields of radiation protection is unthinkable in x-ray diagnosis as this would adversely affect the care for the patient in special situations. On the other hand, it seems unnecessary that dose differences of orders of magnitude should occur for routine examinations. Therefore, dose constraints can be established in a sense that recommended dose values normally should not be exceeded for a certain examination of an average patient. These recommended values are based on the results of extended field studies where good radiographic technique and equipment were to be used. The dose constraints for each specific examination usually were then derived as the third quartile of all doses reported for this examination (3-5). It is important both for carrying out

such a field study and for testing of compliance with the recommended values in routine that the reference doses are of a simple nature and accessible to routine practice. Therefore, usually easily measurable dose quantities are used for this purpose as, e.g., entrance dose (free in air), entrance surface dose or dose-area product.

In this context, the demand for measuring the dose-area product in specified has recently arisen (6). The fields considered are extremely dose-intensive x-ray examinations, paediatric examinations (especially frequent examinations of premature babies), examinations in connection with surgical measures or in interventional radiology with extended fluoroscopy times. Furthermore, it was recommended to measure the dose-area product during the training of medical staff and at facilities equipped to switch to a performance involving high dose-rates. Although no attempt was made to establish dose constraints for these fields, the necessity was recognised to increase the awareness of the magnitude of doses involved; with this increased awareness it is hoped that also the sense of responsibility for the doses applied to patients would increase.

(2) Optimisation of examinations:
Optimisation in x-ray diagnosis means achieving a reasonable compromise between high image quality and low dose to the patient. For this purpose, special quality criteria characterising a good radiographic technique for different types of examinations were evaluated by CEC study groups (3,4). These include detailed requirements for the image as well as a set of technical parameters for the performance of the examination. As the recommended technical parameters are accepted to result in sufficient image quality, optimisation considerations can be based on the resulting patient doses. Here easily accessible doses as, e.g., entrance surface dose, are insufficient because the dependence of doses within the patient on the technical parameters varies with depth in the body and distance from the x-ray field. This is demonstrated by an example in Table 1. The increase of tube voltage from 90 to 125 kV for a chest examination of a female phantom reduces the entrance surface dose markedly, whereas organ doses decrease by much less or even not at all, depending on the position of the organ within the body.

Table 1: *Organ doses, entrance surface dose and effective dose from a postero-anterior chest examination for two different tube voltages, calculated for a female mathematical phantom (7). Focus-to-skin distance: 150 cm; field size: 35 cm · 35 cm; anti-scatter grid: 12/40; dose at image receptor: 5 µGy (Data from (8)).*

	Organ doses (µSv)			Organ doses (µSv)	
Tube voltage	90 kV	125 kV		90 kV	125 kV
Thyroid	22.6	22.8	Adrenals	146.6	116.5
Thymus	37.6	36.6	Kidneys	15.0	11.4
Breast	48.9	43.4	Red bone marrow	45.1	38.4
Oesophagus	79.0	59.4	Skeleton	120.3	86.6
Lungs	214.3	164.5	Skin	37.6	27.4
Liver	52.6	45.7	Muscle	37.6	27.4
Spleen	67.7	54.8			
Pancreas	45.1	38.8	Entrance surface	466.2	301.6
Stomach wall	22.6	22.8	Effective dose	47.5	38.2

Consequently, in a first step, a set of organ doses seems to be a more appropriate descriptor of patient exposure. On the other hand, a whole set of doses is rather unpractical, since single

organ doses are influenced by the altered exposure parameter to a different extent. There may occur situations where a change of one or more technical exposure parameters may result in the decrease of some organ doses and in the increase of some others. Then a list of organ doses does not allow an unequivocal decision which of the techniques considered is preferred with respect to patient dose. In these situations, a more condensed form of the information, preferably in one single number, is of advantage. For this purpose, the effective dose is of benefit, the sum of the weighted equivalent doses in selected organs and tissues of the body as introduced by the ICRP (1). Although this quantity should be used with caution in the field of x-ray diagnosis (8), it allows the ranking of various examination techniques with respect to patient dose in a rather simple and unequivocal way.

(3) Determination of doses to individual patients:
There are very few cases in X-ray diagnosis where a determination of doses to an individual becomes necessary. These are mainly: examinations in the pelvic region of pregnant patients; frequent examinations in the course of occupational diseases after which cancer occurs and the question arises with which probability this was caused by the examinations; unnecessary or inadequately conducted examinations entailing litigation; examinations leading to extremely high doses which might be followed by deterministic effects. In these rare situations, the determination of single organ doses is indispensable. In any situation where the probability has to be determined that a specific disease having occurred in an individual could have been caused by a certain exposure, the dose to the diseased organ is of major importance. In the case of deterministic effects, the knowledge of single organ doses can support further patient care.

(4) Determination of collective doses:
Doses of the whole or of groups of a population are determined to provide data for the following purposes: justification of examinations and risk-benefit analysis as, e.g., in the case of screening programmes; evaluation of the contributions from different examinations to decide where optimisation measures would be most necessary and effective; balancing of exposures due to statistical reasons and comparison with exposures from other sources. Here again, effective dose is of major importance, although the derivation and use of a slightly different quantity employing risk factors more appropriate to a patient population than those specified by the ICRP for a population representative of all ages and both sexes (1) would be highly desirable.

THE DETERMINATION OF PATIENT DOSES IN X-RAY DIAGNOSIS

In those cases where measurable dose quantities can be used, their determination is quite straightforward. Devices for the measurement of dose-area product are becoming more and more frequently available in common x-ray practice; if the field size applied during an examination is known, the entrance dose free in air can be easily deduced. The entrance surface dose can be measured directly on the patient using appropriate dosimeters without hindering the examination.

As soon as organ and tissue doses or effective dose have to be used, the situation is more complicated, as these doses are, in principle, unmeasurable. There is some indication that effective dose could be deduced from measured values of the dose-area product to within approximately 30% accuracy using a single conversion coefficient for groups of projections, field positions and beam qualities (9). Further investigations showed, however, that the numerical uncertainties of converting dose-area product to effective dose exceed 30% in many cases, when generally valid conversion coefficients are used (10). To refine this approach, it becomes

necessary to consider additional exposure parameters like tube voltage and field size, shape and position. Furthermore, this approach is not feasible for single organ doses.

A convenient method for the determination of organ and tissue doses are calculations using radiation transport codes together with computational models of the human body. The calculations are mostly based on the Monte Carlo method; that means that single particle histories are simulated whose exact course is sampled from known probability distributions of the influencing parameters; the dose quantities of interest are finally evaluated by averaging over millions of these single histories.

Types of computational models

The models of the human body used for this type of simulations are mostly so-called mathematical models, that means, mathematical expressions representing planes, cylindrical, conical, elliptical or spherical surfaces are used to describe idealised arrangements of body organs. This type of models was introduced by Fisher and Snyder (11) and further refined and extended by various authors. At the GSF, male and female adult mathematical models have been introduced (7). The best known representative of the mathematical models is that by Snyder et al. (12,13) which has been commonly called the "MIRD-5 phantom" due to being published in the MIRD Pamphlet No. 5. Referring to this, mathematical models are sometimes also called MIRD-type models.

More recently, tomographic models were developed, which use computed tomographic (CT) data of real persons to provide three-dimensional representations of the body. The first step for the construction of these models is to obtain a whole-body CT scan consisting of contiguous slices. The data are then processed using appropriate image processing software. Each organ or tissue is represented by those volume elements (voxels) which were identified as belonging to it from the CT slice images. The tomographic type of models (also called "voxel models") was introduced by two groups independently, approximately ten years ago (14,15). More recently, voxel models were constructed by various workers (16,17). At the GSF, five models of this type were constructed so far, two paediatric, two adult and a model of an Alderson-Rando physical phantom (18-20).

Comparison of mathematical and tomographic models

In mathematical models, organ shapes are reduced to a very simple form to limit the software and computational requirements. Consequently, the mathematical models are not designed to describe any individual in detail but rather to represent whole populations. On the other hand, tomographic models are constructed from CT data of real persons who might deviate significantly from reference data. The shapes of the body organs are determined by identifying all the voxels belonging to each organ. Thus, the shape of each organ is more realistic than for the mathematical models, although, being reconstructed from a specific individual, it might not be representative of large populations.

Mathematical models are usually rigid in size, whereas the external dimensions of tomographic models can be adapted to any size, for each of the three dimensions independently. All internal dimensions of the resulting scaled-down or scaled-up version of the original model are also changed with the same scaling factors. It is, however, important to keep the scaling factors within rational magnitudes; otherwise, considerable errors in the body proportions might be introduced.

In mathematical models, all skeletal components are homogeneously distributed in the skeleton, and there is no geometric representation of spongiosa. Usually, for estimating the dose to

the red bone marrow, the variation of the red bone marrow distribution among different bones and at various ages is considered (21,22), whereas the variation within single bones is not. In tomographic models, the amount of bone marrow and hard bone in each single skeletal voxel can be estimated, based on the CT data. Although it is not possible to identify functional bone marrow by this method or to model the complicated trabecular bone structure exactly, the distribution of bone marrow can be determined with the resolution of the CT scan, i.e. normally a few cubic millimetres.

Apart from these differences which should be kept in mind, the two types of models, mathematical and tomographic, are, in principle, equally suitable for the calculation of organ and tissue doses in x-ray diagnosis.

Studies performed

The mathematical models, primarily designed for use in internal dosimetry, were soon applied for external exposure conditions also. The first of the studies for diagnostic radiology was performed by Rosenstein (23), presenting organ doses for frequent x-ray examinations of adult patients. This was followed by work related to paediatric radiology (24,25) and fluoroscopic examinations of the upper gastrointestinal tract (26). The exposure parameters in these studies refer to the situation in the USA and are different from those considered for organ dose calculations due to examinations in Western European countries. The latter were represented mainly by studies performed at GSF and NRPB for conventional x-ray diagnosis (27-29), computed tomographic examinations (30-33) and, more recently, for paediatric radiology (34). Organ doses for fluoroscopic examination of the coronary arteries were evaluated in a co-operation of the Center of Devices and Radiological Health and GSF (35).

Adult tomographic models were used for the calculation of organ doses from dental radiography (14,36-38). The tomographic paediatric models constructed at GSF were used for organ dose calculations in paediatric conventional x-ray diagnosis (18,39) and CT examinations (40,41). Furthermore, work was performed to study the influence of patient size on organ doses in x-ray diagnosis (42,43). Table 2 shows, as an example, some selected organ dose conversion coefficients for CT examinations of paediatric patients.

Table 2: *Summed organ dose conversion factors for head and thorax scans of an eight week old baby and a seven year old child. The tube voltage is 125 kV, the angle of rotation is 360°. An asterisk as table entry means that the conversion factor is less than 0.005 (Data from (41)).*

	Organ dose per air kerma free in air			
	Head scan		Thorax scan	
Organ	BABY	CHILD	BABY	CHILD
Bladder	*	*	0.01	*
Breast	0.05	0.02	0.96	0.88
Colon	*	*	0.03	0.04
Liver	0.01	0.01	0.47	0.36
Lungs	0.05	0.05	0.95	0.77
Oesophagus	0.15	0.09	0.77	0.68
Ovaries	*	*	0.01	0.01
Red bone marrow	0.33	0.15	0.21	0.10
Skeleton	1.33	0.69	0.88	0.43
Skin (whole body)	0.22	0.12	0.26	0.16
Stomach	0.01	*	0.23	0.16
Thyroid	0.59	0.60	0.58	0.41

CONCLUSIONS

In radiation protection of the patient in x-ray diagnosis, the three principles introduced by the ICRP for occupational radiation protection should be applied also; it should be recognised, however, that in applying these principles a higher flexibility, compared to occupational radiation protection, is needed in order not to adversely affect the care for the patient in special situations. Whereas for the justification of diagnostic radiological procedures patient doses do not play a significant role, they are of major consequence for optimisation and limitation. Dose limitation is, to a certain extent, achieved by the observation of dose constraints which are defined in form of easily measurable quantities, e.g. entrance surface dose, entrance dose free in air or dose-area product. Optimisation in x-ray diagnosis means to achieve a reasonable compromise between high image quality and low patient dose. For optimisation purposes, the above mentioned easily accessible quantities are of limited value; in this context, organ and tissue doses as well as effective dose are the quantities of interest.

Calculations using Monte Carlo techniques together with computational human models are a very suitable method for the determination of organ and tissue doses from various radiation exposures in diagnostic radiology. The calculated dose values apply, strictly speaking, only to patients with the same body characteristics as the models used and to an exact replication of the exposure parameters simulated. Individual doses are strongly influenced by the body dimensions as well as by the irradiation conditions, such as field size, field position, focus-to-film distance, photon spectrum, etc. Accordingly, organ doses derived from the literature may deviate from the doses in real patients and should, therefore, be applied to individual situations with appropriate caution. An evaluation of the influence of patient size, as included in some of the above mentioned studies, improves the applicability of literature data to individual patients to a certain extent. One of the most powerful applications of organ dose calculations is to quantify the effects of changes in the exposure conditions on the doses to single organs, as these effects are largely independent of individual patient anatomy. Knowledge of the relationships between organ doses and certain technical exposure parameters allow to select an examination set-up, among several possibilities known to result in acceptable image quality, which minimises the dose to certain organs. Thus, dose calculations can play an effective role within optimisation in x-ray diagnosis.

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PRACTICAL APPROACHES TO DOSIMETRY FOR THE PATIENT AND STAFF FOR FLUOROSCOPIC PROCEDURES

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INTRODUCTION

Three recent developments that provide practical information on radiation dosimetry for patients or clinical staff for diagnostic or interventional fluoroscopy are presented. They are: (i) a Handbook for determining absorbed doses in tissues (for patients) from coronary artery procedures; (ii) recommendations on avoiding radiation-induced skin injuries (to patients) during fluoroscopically-guided interventional procedures: and (iii) a method for determining effective dose (to staff wearing protective aprons) from fluoroscopic procedures, using the results of personal monitors.

HANDBOOK OF SELECTED TISSUE DOSES FOR FLUOROSCOPIC AND CINEANGIOGRAPHIC EXAMINATION OF THE CORONARY ARTERIES

The Monte Carlo procedure and anthropomorphic phantoms (i.e., ADAM and EVA) used to produce the Handbook for examinations of the coronary arteries is the adaptation of the BRHGAM code (1) by the Gesellschaft für Strahlen-und Umweltforschung (2). The mathematical phantom for the reference male patient (ADAM) is a modification of the original MIRD-5 phantom (3), with the addition of an esophagus (4) and the addition of a supporting tabletop at the back of the phantoms (5). The reference female patient (EVA) is the ADAM phantom reduced uniformly to 83 percent of its original size, with the unique female tissues instead of the unique male tissues (2).

The simulated examinations of the coronary arteries are based on a series of distinct x-ray fields commonly used in coronary interventional radiology, derived from analyses of practice at the Institut de Cardiologie de Montréal (6,7). The views and arterial projections represented are generally applicable to a broad range of examinations following a variety of clinical protocols. For each arterial projection, a separate Monte Carlo calculation was made with ADAM and EVA for the relevant complex oblique x-ray field in divergent-beam geometry, with specific x-ray spectra representative of clinical practice (5).

The Handbook presents twelve tabulations, one tabulation for each of eleven arterial projections, and one tabulation that provides nominal data that can be used for an examination consisting of several views, in lieu of the individual tabulations. The nominal approach is acceptable for coronary artery examinations because all the views share the heart as a relatively small, common region intercepted by the central ray of each different x-ray field. The arterial projections are specified by view (e.g., RAO, right anterior oblique), angulation of image receptor (if different than 0 degrees) in the transverse and saggital planes, and location of the field center (left or right ventricle). Descriptions of the standard nomenclature used in the anatomy of the coronary arteries and to identify the arterial projections, the specifications for the irradiation geometry and complex oblique views, and the coefficients of variation for the Monte Carlo calculations are provided in the Handbook (5). Data are tabulated for the entrance skin in the primary field and 20 internal tissues or organs. The tabulated values (conversion coefficients) are tissue dose (mGy) per 1-Gy air kerma (free-in-air) at the skin-entrance plane.

In extended coronary artery examinations, cumulative absorbed doses to that portion of skin lying directly in the path of the incident primary field may approach or exceed the thresholds for deterministic injury. The entrance skin in the primary field is only a small fraction of the entire skin tissue; the extent is delimited and the location is determined by the collimation and irradiation geometry of the anatomical projections.

Except for the heart, the internal organs and tissues listed are those with which cancer, genetic effects or in utero effects have been associated (8). The heart surrounds the ventricle field centers and always lies within the field of view. It receives the highest absorbed doses per unit air kerma of the internal organs. The data for the heart are provided for reference purposes; there is no health effect yet established for absorbed doses in the heart for the ranges that occur in coronary artery examinations.

The data are presented for three beam qualities for each reference patient (i.e., ADAM, 2.5, 4.0 and 5.5 mm Al HVL; EVA, 2.0, 3.5 and 5.0 mm Al HVL). The range of beam qualities corresponds to that observed

in the study conducted at the Institut de Cardiologie de Montréal (7) and that observed in a nationwide survey of fluoroscopy practice in the United States (9). The beam qualities are without the presence of a supporting tabletop. For the usual ranges of kVp and aluminum filtration combinations used in fluoroscopy and cineangiography of the coronary arteries, the results should have uncertainties of less than 10 percent when half-value-layer (HVL) alone is used to describe beam quality (5).

When the actual diameter of field of view (FOV) at the image receptor plane differs from that used in the reference simulation by more than 20 percent, the following correction is recommended (5):

$$[\text{FOV(actual)}/\text{FOV(tabulated)}]^2.$$

The correction is not applicable to the entrance skin in the primary field, since the absorbed dose in that portion of skin is not dependent on the size of the area irradiated.

The tabulation for the nominal conversion coefficients is given in Table 1.

Table 1. Nominal Conversion Coefficients for Fluoroscopic and Cineangiographic Examinations of the Coronary Arteries (from reference 5): SSD = 60 cm; SID = 90 cm; FOV diameter at image receptor = 14 cm

	Tissue dose (mGy) per 1-Gy air kerma (free-in-air at the skin-entrance plane)					
	Male			Female		
HVL (mm Al)	2.5	4.0	5.5	2.0	3.5	5.0
Entrance Skin in Primary Field	1000	1120	1180	950	1090	1170
Brain	0.003	0.020	0.041	0.001	0.018	0.045
Thyroid	0.12	0.50	0.85	0.075	0.53	1.1
Thymus	2.2	6.5	9.9	1.6	6.7	12
Active Bone Marrow	6.1	12	16	5.2	12	17
Esophagus	14	33	47	11	32	51
Lungs	34	53	65	31	55	71
Breasts				3.0	9.4	15
Heart	30	62	86	23	63	95
Adrenals	36	62	78	32	64	87
Spleen	4.9	11	15	3.8	11	17
Pancreas	5.4	14	20	3.8	13	22
Stomach	2.4	6.3	9.3	1.7	6.3	10
Liver	4.4	9.7	14	3.5	9.9	15
Kidneys	3.2	6.8	9.3	2.7	7.1	11
Colon	0.071	0.31	0.56	0.043	0.32	0.67
Small Intestine	0.091	0.40	0.72	0.050	0.39	0.82
Ovaries				0.007	0.076	0.19
Uterus				0.005	0.071	0.17
Urinary Bladder	0.003	0.021	0.044	0.001	0.023	0.054
Testes	+	0.002	0.004			

+ less than 0.001 mGy absorbed dose per 1-Gy air kerma.

The Handbook can be used to perform a view-by-view or a nominal analysis of an examination. To use the nominal approach, the coronary artery examination is characterized in an overall sense. This permits a quick, but somewhat less accurate way to estimate nominal tissue doses for a complete examination without detailed specifications for the particular views applied clinically.

The complete examination is characterized with nominal values for four parameters:

(a) the beam quality (i.e., mm Al HVL);

(b) the total air kerma (free-in-air at skin-entrance plane) summed for all the fluoroscopic plus cineangiographic segments and for all skin-entrance planes;

(c) the field-of-view diameter at the image receptor plane; and

(d) the highest cumulative entrance air kerma (i.e., fluoroscopic plus cineangiographic) for any single skin location. Such a skin region may be irradiated in only one view or in multiple views that share a common locus of irradiation.

The example given below is for a left-heart study of an adult male entailing a left ventriculogram in biplanar mode followed by left and right coronary angiography. In a typical application of the nominal approach, the user will rely on estimated values of the parameters for the entire examination that may be truly nominal, developed at the level of detail and with the degree of accuracy available to the user.

Nominal Parameters for the Entire Examination (Example)
Half-value-layer: 3.6 mm Al
Total air kerma (free-in-air) for all skin-entrance planes: 1.6 Gy
Field-of-view (FOV) diameter at image-receptor plane: 20 cm
Highest cumulative entrance air kerma for any single skin location: 0.80 Gy

Notes: a correction factor of 2.0 is needed for the actual FOV (20 cm rather than the 14 cm in the reference tabulation); and linear interpolation is made between reference HVLs

Tissue	mGy per 1-Gy Air Kerma	Relevant Air Kerma (Gy)	FOV Correction	Tissue Dose (mGy)
Entrance skin in primary field (maximum)	1090	0.80	none	870
Active bone marrow (entire examination)	10	1.6	2.0	32
Lungs (entire examination)	48	1.6	2.0	150

The Handbook presents this same example evaluated by a method in which detailed specifications for the particular views applied clinically were obtained for the examination. The absorbed doses were:

Entrance skin in primary field (maximum), 830 mGy;
Active bone marrow (entire examination), 37 mGy;
Lungs (entire examination), 170 mGy;

which are close to the values obtained above by the nominal approach.

The following general observations for absorbed doses can be made for examinations of the coronary arteries:

(a) Since a variety of different views are used in any specific examination, no single area of entrance skin is uniquely irradiated throughout the examination. Therefore, the cumulative air kerma for all the views will be a significant overestimate of the maximum dose to any portion of the entrance skin. The largest single cumulative dose in a portion of the entrance skin occurs where a common region of skin is irradiated in multiple views. The maximum absorbed dose to a portion of entrance skin will need to be evaluated for each examination.

(b) For the fields used in coronary artery examinations, larger fractions of the heart, lung, esophageal and adrenal tissues are in the radiation fields than other internal tissues, although the heart is not subject to radiation risk at the doses involved. The magnitude of the absorbed dose in these tissues is much lower than the maximum absorbed dose in the entrance skin in the primary field (see value for the lungs in the example). The absorbed doses in other internal tissues are less than the absorbed doses in heart, lung, esophageal and adrenal tissues, some being an order of magnitude or more lower per 1-Gy air kerma (see Table 1).

RECOMMENDATIONS FOR AVOIDANCE OF SERIOUS X-RAY-INDUCED SKIN INJURIES TO PATIENTS DURING FLUOROSCOPICALLY-GUIDED PROCEDURES

An increasing number of invasive procedures, primarily therapeutic in nature and involving use of devices under fluoroscopic guidance, are becoming accepted medical practice. These procedures are performed by a variety of medical specialists and may provide significant advantages over alternate therapies in terms of improved clinical outcome and reduced patient risk. However, physicians performing these procedures should be aware of the potential for serious radiation-induced skin injury caused by long periods of fluoroscopy occurring with some of these procedures. Such injuries have been reported as a result of radiation exposure due to long fluoroscopic exposure times, high dose rates or both (10).

The types of injury to skin and adjacent tissues which result from x-ray irradiation are summarized in Table 2 along with the typical absorbed dose in the skin required to produce the injury. Appearance of the injury is dependent on variables other than cumulative absorbed dose in the skin, such as: the rate of delivery of the radiation; the fractionation of the absorbed dose; the age and characteristics of the exposed person; and the site on the skin of the exposure.

Table 2. Typical Threshold Doses for Radiation-induced Skin Injuries (adapted from reference 11)

Effect	Typical Threshold Absorbed Dose (Gy)	Fluoroscopic On Time (a) to reach Threshold (hours) at 0.02 Gy min^{-1}	at 0.20 Gy min^{-1}	Time (b) to Onset of Effect
Early transient erythema	2	1.7	0.17	hours
Temporary epilation	3	2.5	0.25	~3 weeks
Main erythema	6	5.0	0.50	~10 days
Permanent epilation	7	5.8	0.58	~3 weeks
Dry desquamation	10	8.3	0.83	~4 weeks
Invasive fibrosis	10	8.3	0.83	- - -
Dermal atrophy	11	9.2	0.92	>14 weeks
Telangiectasia	12	10.0	1.00	>52 weeks
Moist desquamation	15	12.5	1.25	~ 4 weeks
Late erythema	15	12.5	1.25	~6-10 weeks
Dermal necrosis	18	15.0	1.50	>10 weeks
Secondary ulceration	20	16.7	1.67	>6 weeks

(a) Time required to deliver the typical threshold absorbed dose to skin at the specified rate.
(b) Time after single irradiation to observation of effect.

The absorbed dose rate in the skin from the direct beam of a fluoroscopic x-ray system is typically between 0.02 and 0.05 Gy min^{-1}, but may range from 0.01 to more than 0.50 Gy min^{-1}, depending on the mode in which the fluoroscopic equipment is operated and the size of the patient. The times required to deliver the typical threshold dose shown in Table 2 are for fluoroscopic dose rates of 0.02 Gy min^{-1} (the usual dose rate for normal fluoroscopy for an average-sized patient) and 0.20 Gy min^{-1} (a dose rate near the maximum permitted for the high-level control mode of operation in the United States) (12). The times listed are for irradiation to a single skin area by a stationary, continuous fluoroscopic x-ray beam.

Other than the mildest symptoms, such as transient erythema, the injuries from the irradiation may not appear until weeks following the exposure. Physicians performing these procedures may not be in direct contact with the patients following the procedure and may not observe the symptoms when they occur. For this reason, it is recommended that information be placed in the patient's record which permits estimation of the absorbed dose to the skin, especially for patients who may receive a significant fraction of the threshold dose from single or multiple procedures. Physicians should counsel such patients on the possible symptoms and risks from these procedures.

Table 3 presents examples of the x-ray-induced skin injuries reported to the Food and Drug Adminstration in the United States that illustrate the range of procedures and the severity of the injuries. Details that would permit quantitative estimates of the absorbed doses to the skin were generally not available, since records were not maintained on the extent of fluoroscopic exposure times and other technical factors or the facility was unwilling to share additional information.

Table 3. Examples of Skin Injuries from Fluoroscopy Reported to the Food and Drug Administration (from reference 10)

Patient Sex and Age	Procedure	Nature of Injury	Fluoroscopic Exposure Time
male, 40	coronary angiography and PTCA followed by second coronary angiography	skin necrosis requiring 12 cm x 10 cm skin graft	unknown; estimated to exceed 120 min
female, ?	RF cardiac catheter ablation	7.5 cm x 12.5 cm second degree burn	unknown
female, 25	RF cardiac catheter ablation	skin breakdown 3 weeks post procedure	unknown; procedure time of 325 min
female, 34	RF cardiac catheter ablation	draining skin lesion on back 5 weeks post procedure	unknown; procedure time of 190 min
female, 62	balloon dilation of bile duct anastomosis	burn-like injury on back requiring skin graft	unknown
female, 61	renal angioplasty	skin necrosis requiring skin graft	unknown, procedure time of 165 min

PTCA is percutaneous transluminal coronary angioplasty; RF is radiofrequency

The Food and Drug Administration has issued a Public Health Advisory (13) with recommendations for avoiding such x-ray-induced skin injuries and additional advice (14) on recording information in the patient's record that identifies the potential for such injuries.

Recommendations to avoid serious x-ray-induced skin injuries during fluoroscopically-guided procedures

(a) Establish standard operating procedures and clinical protocols for each specific type of procedure performed. The protocols should address all aspects of the procedure, such as patient selection, normal conduct of the procedure, actions in response to complications and consideration of limits on fluoroscopy exposure times.

(b) Know the radiation dose rates for the specific fluoroscopic system and for each mode of operation used during the clinical protocol. These dose rates should be derived from measurements performed at the facility.

(c) Assess the impact of each procedure's protocol on the potential for radiation injury to the patient.

(d) Modify the protocol, as appropriate, to limit the cumulative absorbed dose to any irradiated area of the skin to the minimum necessary for the clinical tasks, and particularly to avoid approaching cumulative doses that would induce unacceptable adverse effects. Use equipment which aids in minimizing absorbed dose.

(e) Enlist a qualified medical physicist to assist in implementing these principles in such a manner so as not to adversely affect the clinical objectives of the procedure.

Additional advice on recording information in the patient's record that identifies the potential for skin injuries

(a) Each facility should establish a threshold dose for recording information. The Food and Drug Administration suggested a threshold absorbed dose in the skin of 1 Gy, but the facility may select another value (such as 2 Gy) based on its experience.

(b) Determine the fluoroscopically-guided procedures that will approach or exceed the selected threshold. The Food and Drug Administration stated that the list should include the following procedures:

Radiofrequency cardiac catheter ablation
Vascular embolization
Transjugular interhepatic portosystemic shunt
Percutaneous endovascular reconstruction

and any others that professional and medical specialty organizations suggest or the facility determines will approach or exceed the selected threshold.

(c) For these procedures, record an unambiguous identification of those areas of the patient's skin that received an absorbed dose that may approach or exceed the selected threshold. The facility may also wish to include in the patient or supplemental record an estimate of the cumulative absorbed dose (and an estimate of its uncertainty) to each irradiated area of the skin noted in the patient record, or sufficient data to permit estimating the absorbed dose to those areas of skin. Cumulative absorbed dose in skin refers to the dose accrued by any specific area of skin over the course of a single or possibly multiple procedures.

No consensus currently exists as to the most effective method for estimating skin dose. Absorbed dose in the skin from fluoroscopy may be estimated through: (i) direct measurements, such as placing radiation dosimeters on the patient during the procedure; or (ii) indirect means, such as collection of specific information for a patient on equipment technique factors combined with patient and procedure characteristics, or such as use of supplementary information obtained with measurement and recording devices attached to the x-ray equipment. Note, however, that the sum of all exposures (i.e., the cumulative value for air kerma) occurring in an entire procedure is likely to be a significant overestimate of the cumulative absorbed dose to a specific area of skin, except in the event that the x-ray beam is stationary during most of the procedure.

Each approach to dose estimation has advantages and disadvantages, and all of the approaches involve practical complexities. Fortunately, clinical decisions and patient management do not require highly accurate estimates of the cumulative absorbed dose to the skin. It is more important that the potential for approaching or exceeding the threshold for injury be recognized and avoided, if possible.

A METHOD FOR DETERMINING EFFECTIVE DOSE TO STAFF WEARING PROTECTIVE APRONS DURING FLUOROSCOPY AND INTERVENTIONAL RADIOLOGY

Clinical staff taking part in diagnostic and interventional procedures using fluoroscopy wear protective aprons to shield internal tissues and organs in the torso from scattered x rays. Use of the measurements from monitoring devices worn outside and above protective aprons as the record of effective dose (E) for these individuals results in significant overestimates of their actual risk (15).

Experimental determinations of E have been reported for simulated irradiation of clinical staff for conditions commonly encountered in fluoroscopy and interventional radiology (16). In that work, x-ray scatter radiation was produced at various x-ray tube potentials in the range of 70 to 110 kVp, with the x-ray tube in overtable or undertable position. The operational quantity, personal dose equivalent for strongly-penetrating radiation, $H_p(10)$, can be obtained from the film badge dosimeters that were placed on the neck and waist of a Rando phantom that simulated a clinical staff member. Absorbed doses to tissues and organs, when a protective apron was not present, were determined using numerous thermoluminescent dosimeters in the phantom. Absorbed doses to the tissues, when a protective apron was present, were estimated from the absorbed doses without a protective apron, as modified by transmission data for the appropriate x-ray tube potential and equivalent lead thickness of the apron. E was computed as described by the ICRP (8) for the noted range of x-ray tube potentials without an apron and with aprons having equivalent lead thicknesses from 0.1 to 0.5 mm.

The relationships between E and the $H_p(10)$ values obtained from the film badges are quite variable over the conditions studied. For example, for aprons of 0.3 and 0.5 mm lead equivalence, $H_p(10)$ for the neck film badge ranges from 21 to 72 times higher than the corresponding E, and $H_p(10)$ for the waist film badge ranges from 1.7 times higher (multiply by 0.60) to 67 times lower than the corresponding E (15). The fluctuations in kVp used during various fluoroscopy procedures render a direct use of the conversions for individual staff impractical. However, when a single personal monitor is worn at the neck outside and above a protective apron, dividing the $H_p(10)$ value for this personal monitor value by 21 (the minimum value in the range) will provide a conservatively high estimate of E. This modification gives appropriate credit for the protection afforded by the apron and does not overestimate the value of E by more than a factor of 3.4.

From the experimental results, one can produce an empirical formula for E (that uses the results of two personal monitors) of the form:

$$\text{estimate of } E = aH_1 + bH_2;$$

which is a weighted sum of the $H_p(10)$ values obtained with personal monitors worn under a protective apron at the waist (H_1) and at the neck above and outside a protective apron (H_2).

The procedure was to iterate values of the weighting factors a and b by trial and error until a desired approximation of E for radiation protection purposes is achieved for the clinical range of x-ray tube voltages, the two x-ray tube locations, and for aprons with 0.5 and 0.3-mm lead equivalent thicknesses.

The criteria for a desired formula for radiation protection purposes were: (i) minimize the underestimates of E, even at the expense of larger overestimates of E for some conditions, and (ii) obtain a close estimate of E at the combination most frequently encountered in clinical practice (i.e., 90 kVp, 0.5 mm lead equivalent apron and undertable x-ray tube).

The resulting formula, using the experimental conversion factors, is:

$$\text{estimate of } E = 0.5\ H_1 + 0.025\ H_2.$$

The estimates of E resulting from the formula are presented in Table 4. Over the stated range of conditions, the estimates range from 1.06 E to 2.03 E. The criteria for a desired formula are met. There are no underestimates and the largest overestimate is 2.03 E. The estimate for the most frequently encountered combination is 1.06 E. Use of the formula would be a simple and practical way to monitor E when both personal monitors are worn.

Table 4. Estimates for E Using Empirical Formula for Two Personal Monitors, Aprons Present, Over and Undertable X-ray Tubes (Derived from references 16 and 17)

Apron Thickness (mm lead equivalent)	Tube Voltage (kVp)	Estimate of E Relative to a Value of 1.0
0.5, overtable	70	1.49
	90	1.96
	110	1.13
0.5, undertable	70	1.10
	90	1.06
	110	1.34
0.3, overtable	70	1.56
	90	2.03
	110	1.26
0.3, undertable	70	1.27
	90	1.21
	110	1.44

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Molecular and Cellular Mechanisms in Radiation Carcinogenesis

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Substantial epidemiological data exist which implicate ionising radiation as a causative agent for a broad range of human cancers (1). To date, epidemiology has been able to provide good evidence of increased cancer risk for single acute radiation doses greater than 100 mGy. Understanding and estimation of radiation cancer risks below this level remain problematic and controversial. Developing an in depth knowledge of the mechanisms by which ionising radiation causes cancer should help to strengthen low dose risk estimates.

In recent years there have been significant advances in the understanding of many basic cellular processes including the regulation of cell proliferation. Unrestrained cellular proliferation is the key characteristic of cancer. The causes of this unrestrained proliferation lie in alteration of the normal patterns of expression of genes involved in proliferative control. Powerful tools for molecular genetic analysis are now available to help elucidate the genetic alterations in cancers. Cell biological and biochemical methods can be used to define the cellular consequences of these genetic alterations.

The basis for much of the understanding of genetic mechanisms in carcinogenesis has its origins in the study of chromosomes. Two broad categories of chromosomal rearrangement have been identified in human and animal tumours: translocations and deletions. Chromosome deletion events lead to the loss of genetic information while translocations cause the rearrangement of genes but not gene loss. Translocations are largely, but not exclusively associated with, haematological malignancies (see reviews 2,3). Deletions are more commonly associated with solid tumours (see 2,4) although deletions have also been observed in some human haematological malignancies such as acute myeloid leukaemias (AMLs, eg. 5,6). The consistent association of a particular chromosomal rearrangement with a specific tumour is indicative of the importance of genes on the involved chromosomes in the development of the tumour.

There is now a very large number of consistent chromosomal translocations associated with specific tumours (eg. 2,3). Among those identified are the t(8;14) in Burkitts lymphoma (BL), t(10;14) in T cell acute lymphocytic leukaemia (T-ALL), t(9;22) in chronic myelogenous leukaemia (CML) and t(15;17) in acute promyelocytic leukaemia (APML). Molecular analysis of these translocations has led to the identification of the genes involved and given some indications of the consequences of the translocations. Essentially two classes of translocation have been identified, those that lead to the over expression of one of the partner genes and those in which a novel fusion gene is formed from which a fusion oncoprotein is produced. Translocations of the over expression type are common in lymphocytic leukaemias. In T-cell leukaemias one of the genes is commonly a T-cell receptor (*TCR*) gene while in B cell leukaemias immunoglobulin (*Ig*) loci are frequently involved. These genes are actively expressed in the corresponding normal cell type. Genes which are otherwise silent (ie. not expressed) can become active when translocated into the vicinity of one of the highly expressed genes. Thus in Burkitts lymphoma the *MYC* gene is over expressed due to its translocation close to an *Ig* gene locus and in the t(10;14) in T-ALL the expression of the *HOX11* gene is driven by a *TCR* gene. Examples of gene fusions include *BCR-ABL*, involved in the t(9;22) in CML and *PML-RARA* in APML. These fusions lead to the production of proteins with abnormal functions.

Both the over expression and gene fusion translocations can be considered 'gain of function' mutations, ie. normal genes are expressed inappropriately or novel fusion genes are expressed. These events are in general genetically dominant, that is only one such change is sufficient to give an effect. The products from this group of proto-oncogenes are frequently involved in the processes of signal reception and transduction from the cell membrane to the nucleus (signal transduction pathway genes) or more directly in the regulation of gene activity (transcription factor genes). Gain of function mutations can also be on a smaller scale as in the *RAS* gene base pair mutations associated with rodent skin, breast and liver tumours (7,8,9).

Deletion events involve a class of genes known as tumour suppressor genes. Frequently, these genes are involved in the control of cell proliferation, differentiation or transcription. Consequently, upon the loss or mutation of such genes, proliferation may be enhanced of differentiation blocked. The classic examples of deletions in cancers are the del(13) involving the *RB1* gene in retinoblastoma and the del(11) in Wilms tumours in which the *WT1* gene is lost. In general these events are genetically recessive and so loss or mutation of both copies of the gene is necessary for a phenotypic effect. The gene which has been found to be most commonly lost or mutated in human cancers, *p53* (10,11), also acts as a tumour suppressor although some dominant negative mutations exist.

A single genetic alteration is rarely, if ever, sufficient to give rise to a malignant tumour. In solid cancers it is likely that 5-10 genetic or epigenetic alterations are required. In haemopoietic system cancers it is probable that fewer events are necessary. Where a very consistent genetic alteration is associated with a specific cancer, it is a good candidate for an initiating event. However, many of the genetic changes in tumours have a less strong and specific association. A gene very consistently altered in one tumour may also be involved in other tumour types but in a less specific fashion. One of the best studied human cancers is colorectal cancer in which consistent and early changes to tumour suppressor genes is followed by an accumulation of secondary events (12).

These basic principles of carcinogenesis have in the main been derived from the analysis of human tumour material. The identification of human tumours which are known to be associated with radiation exposure is very difficult. Therefore much of the current research on radiation carcinogenesis exploits animal models. At NRPB there is a strong interest in mechanisms of radiation-induced AML in the mouse. This system will be described to give a flavour of how molecular and cytogenetic methods can be exploited to understand mechanisms of radiation carcinogenesis.

AML can be induced to a maximum frequency of ~25% of CBA/H mice following a single acute x-ray exposure (13,14). Radiations of other qualities such as α-particles and neutrons can also induce AMLs in this inbred mouse strain (16,17). Cytogenetic analysis of radiation-induced AMLs by G-banding revealed a consistent chromosomal aberration, deletion or translocation of one chromosome 2 homologue (18). Similar aberrations have been found in AMLs induced in other mouse strains (19-21). Chromosmoe 2 aberrations can also be detected at 5 days - 1 month post irradiation in bone marrow cells (18), well before the average 18 month latent period required for full tumour development. Not only do chromosome 2 aberrations form early but they form in excess by comparison with other chromosomes - an approximately 2.5 fold excess of chromosome 2 events was noted at sample times between 1 and 6 months post-irradiation (22). In this study (22) it was found that all irradiated animals carried chromosome 2 aberrations, so while these aberrations are very early and possibly initiating events, they do not determine which individuals will develop AML. Similarly this experiment indicated that secondary events, undetectable at the cytogenetic level, contribute to leukaemogenesis.

Further resolution of the DNA sequences involved in mouse leukaemogenesis has required the development of a system for molecular mapping. By exploiting differences in allele sizes between inbred strains for certain abundant and widely distributed molecular markers known as microsatellites, mapping of deletions in AMLs developing in irradiated F1 mice has been possible (23). Chromosome 2 deletions

were detected in about 65% of AMLs arising in x-irradiated F1 CBA/H x C57Bl Lia mice. While the breakpoints involved in the deletions were not entirely consistent, clustering was observed and it has been possible to define 5 deletion types (23). In addition to defining breakpoints at the molecular level it has been possible to identify a minimally deleted region of about 30 cM in size. In the absence of strong breakpoint specificity, a mechanism of tumorigenesis involving abnormalities in gene expression seems unlikely, gene losses are probably of greater importance.

The microsatellite markers associated with breakpoints and the minimal deleted region can be used to isolate larger DNA clones in these areas. Such clones can then be exploited for further molecular and cytogenetic resolution of the AML associated genetic alterations. Using the technique of fluorescence *in situ* hybridisation (FISH) with these clones, it is possible to quantitate more accurately the frequency of these AML associated aberrations at very early post irradiation times. Thus, the size of the pool of radiation-initiated cells can be defined in individual animals. Through further genomic-wide screening of AMLs by microsatellite analysis, additional target genes for acute myeloid leukaemogenesis may be identified. FISH experiments as described above should enable the assignment of a temporal position for such changes in the leukaemogenic process.

One further issue in radiation carcinogenesis which can now be addressed is that of genetic predisposition. Within human populations certain rare cancer prone individuals and families exist. Some of these cancer prone syndromes are associated with the inheritance of genes which encode elements of DNA damage processing systems. Examples of this category are xeroderma pigmentosum (24), hereditary non-polyposis colon cancer (25) and ataxia telangiectasia (26). Inheritance of classic tumour suppressors such as *RB1* and *WT1* underlie familial retinoblastoma and Wilms tumour respectively (see review, 4). Proto-oncogenes at present seem rarely to be associated with cancer predisposition. The only clear example here is the *RET* proto-oncogene involvement in multiple endocrine neoplasia type 2A (27). As yet there are no clear cases of a human gene or genes predisposing to radiation-induced cancer. However, with the recent identification of a gene mutated in ataxia telangiectasia (28) this situation may change. The differences in tumour incidence between strains of mice have been exploited to map tumour susceptibility loci (eg. 29). Such strategies can also be applied to radiation-induced cancers. It has proven possible to generate artificial mouse mutants susceptible to radiation carcinogenesis (30).

To summarise, the advances in the understanding of the process of carcinogenesis in general and radiation carcinogenesis in particular can benefit mechanistic modelling of radiation cancer risk. A knowledge of radiation cancer associated genetic alterations and their temporal sequence allow the construction of valid models which can be exploited in risk estimation. Furthermore, the target gene alterations identified can be used for screening of individuals for preneoplastic lesions and in some case suggest rational bases for therapy.

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IONIZING RADIATION-INDUCED DNA DAMAGE AND ITS REPAIR

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The deposition of energy in the cell by ionizing radiation results in damage to DNA, both directly and indirectly, as a effect of free radical formation. As a consequence, a variety of DNA lesion are induced, including single- and double-strand breaks (SSB and DSB, respectively), and base damages. All organisms have highly efficient mechanisms for the recognition and repair of DNA damage. The cellular response to ionizing radiation, in addition to DNA repair, includes other safeguards such as cell cycle regulation and mechanisms involved in scavenging of free radicals which are produced by ionizing radiation. Therefore, a tremendous complexity of the cellular defense mechanisms against ionizing radiation can be expected.

To dissect and elucidate these mechanisms, X-ray-sensitive mutants have been investigated in a wide variety of organisms. In humans only two hereditary disorders are shown to have clearly an increased sensitivity to ionizing radiation: ataxia telangiectasia (AT) and Nijmegen Breakage Syndrome (NBS) (reviewed in ref. 1 and 2). The recent isolation of the *ATM* (*AT Mutated*) gene for several previously established complementation groups indicate that one gene is responsible for AT (3). The existence of patients with characteristics of both, AT and NBS indicate that these two disorders are closely related, although recent studies suggest that they are distinct, since the gene(s) defective in NBS is not located at the site of the *ATM* gene (4 and 5).

As a genetic approach to analyze the mammalian cellular response to ionizing radiation, in addition to the human mutants, many X-ray-sensitive mutants have been isolated in cultured rodent cells, and at least eleven complementation groups have been identified (reviewed in ref. 6). Amongst these groups, one group has been suggested to be defective in the gene homologous to the AT gene (7).

Recently, a fruitful interaction between researchers working in different fields, such as radiobiology, biochemistry, immunology and somatic cell genetics has led to the discovery that V(D)J recombination, the process responsible for the formation of the immunoglobulin and T-receptor genes, utilize elements of the DSB repair machinery. Studies with four groups of X-ray-sensitive rodent cell mutants (groups 4, 5, 6, and 7), have led to the identification of the *XRCC4* (8), *Ku86, Ku70*, and *DNA-PK*$_{cs}$ genes (reviewed in ref. 9 and 10). Recently, mutations in XRCC4 and Ku86, have been identified in the hamster mutants of group 4 and 5, respectively, providing the direct evidence that the *XRCC4* (8) and *Ku86* (11) genes, are responsible for the observed phenotype of these groups of mutants. Sofar, mutations in DNA-PK$_{cs}$ which are responsible for the defect in *scid* mouse cells, have not been identified.

Rodent mutants have served as a tool for the isolation of the *XRCC1* and *XRCC3* genes (12 and 13) which are involved in DNA single-strand break repair. The *XRCC1* gene product is required for normal activity of DNA ligase III (14), and the *XRCC3* gene function remains unknown.

In addition to the defective repair of DNA lesions induced by ionizing radiation, defects in cell cycle progression might lead to the increased X-ray sensitivity. The phenomenon of radioresistant DNA synthesis (RDS) after γ-irradiation was the first sign of a defect in cell cycle control in AT cells. A major insight into the nature of the product of the *ATM* gene is indicated by the observation that the carboxyl terminus of this protein is similar to the catalytic subunit of phosphoinositide 3-kinase (PI 3-kinase) (3). This protein is implicated in the response to DNA damage. The ATM protein shows the functional homology to the products of several genes such as, MEC1, rad3, mei-41, TEL1 and DNA-PK$_{cs}$ which play a role in cell-cycle control in the presence of DNA damage (reviewed in ref. 15 and 16).

A recent finding that a gene located on human chromosome 4q enhances the level of inhibition of DNA synthesis after gamma-irradiation (17), indicates that the rate of DNA synthesis is regulated by numerous genes, including *ATM, NBS*, and the gene localized on chromosome 4q. The identification of hamster mutants showing RDS (18) should be helpful in the identification of these genes. Since the gene on chromosome 4q inhibits DNA synthesis after ionizing radiation without correcting the X-ray sensitivity, in terms of cell killing or chromosomal aberrations, it is indicated that RDS is not responsible for these biological consequences of RDS (19).

The isolation of the genes involved in NBS, as well as in the remaining complementation groups of rodent X-ray-sensitive mutants, and the recognition of their precise role, should further elucidate the mechanism of the cellular response to ionizing radiation and its involvement in cancer.

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Biological indicators for radiation exposures

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The experience with accidental radiation exposures has frequently shown that the physical dosimetry does not give sufficiently good results for dose estimations. Therefore the determination of biological effects is necessary in order to get some additional information about the absorbed radiation dose. Several biological systems have been studied under these circumstances and interest for biological indicators for radiation damage has increased appreciably. There are two main reasons for this interest:

1. Biological indicator can be used in addition or even in substitution to physical dosimetry. It is an advantage that biological indicators can be determined for biological dosimetry after a radiation exposure has taken place which is in contrast to physical dosimetry.

2. Biological dosimetry gives not only information about the range of the absorbed (radiation dose) but it is also possible to get information about the individual radiosensitivity which can vary considerably from person to person. Therefore biological indicators are also a measure of the biological-medical radiation damage. By determination of such indicators it is possible to give a prognosis about possible radiation damage. It is further possible to start therapeutical procedures under these conditions earlier.

A number of methodologies have been studied for these purposes:

1. Electronspin resonance,
2. Biochemical indicators,
3. Chromosomal aberrations,
4. Hematopoetic changes,
5. Studies of spermiogenesis.

The most efficient and widely used techniques are those which measure chromosomal aberrations. Several methodologies are used in this connection: The evaluation of structural chromosomal aberrations in metaphases,

premature chromose condensation, the determination of micronuclei and recently the use of molecular biological techniques like fluorescence in situ hybridization (FISH).

A useful biological indicator should fulfil the following features:

1. The measured radiation effect should be dose dependent in a certain dose range. It would be desirable that this dose range covers radiation doses of occupational exposures at working places (20 - 50 mSv) up to accidental exposures of several sievert.

2. A certain specificity should exist for the changes by ionizing radiation.

3. It should be possible to get some information about dose rate effects.

4. The results should be available within a few days after radiation exposure.

5. The radiation damage should be measurable over a certain time period.

6. It should be possible to determine a partial body irradiation and to measure the localization of radiation exposure.

7. The biological material which is needed for the determination should be easily obtained.

These demands are maximal and cannot be fulfilled by any indicator up to now. Ionizing radiation induces free radicals in many biological materials. These free radicals can be quantitatively determined by electronspin resonance. The measurements can only be performed with materials with a low water content. Several studies have shown that teeth are very well suited for such studies. But also in hair and bone free radicals can be measured. After the nuclear accident in Tschernobyl this technique has been used for dose estimation in several cases. It is also possible by this technique to measure radicals in bricks, stones and other material in order to estimate local radiation doses.

A number of studies have been performed in order to use biochemical systems for dose estimation. In this connection the excretion of metabolites after the

break down of proteins and nucleic acids has been mainly investigated. It was observed that in a dose range between 0.5 to 2.5 Gy a good dose response can be obtained for taurine and other amino acids as well as with metabolites of DNA. However, there is a very high individual variability with respect to these parameters and therefore it is difficult to obtain a reasonable dose estimation with the biological systems which have been used up to now.

Most experience for biological dosimetry has been obtained with cytogenetic techniques through which chromosomal changes can be measured. Most widely the determination of the socalled dicentric chromosomes has been used. These chromosomes are formed after chromosome breakages and subsequent connection of these breaks in such a way that a chromosome with two centromeres results. The formation of dicentrics shows a certain specificity for ionizing radiation. The conventional technique has been found useful for dose ranges between 0.1 to several sieverts. The dependence on radiation quality and dose rate has been studied. Again the spontaneous rate of dicentrics can differ individually considerably. This has to be taken into consideration and complicates dose evaluation in the low dose range. Usually chromosomal aberrations are determined in metaphases of proliferating lymphocytes from blood after cell proliferation has been stimulated. Dicentric chromosomal aberrations have to be measured within a few weeks after a radiation exposure. Otherwise the cells with such chromosomal aberrations get lost and an estimate is no longer possible.

For these disadvantageous reasons a new technique has been introduced: The fluorescence in situ hybridization (FISH). With this technique it is possible to paint specific chromosomes by the corresponding DNA probes and to study not only instable chromosomal aberrations like dicentrics or chromosome breaks but also to determine stable chromosomal aberrations like translocation, deletious etc. As DNA probes for all human chromosomes are available today this technique can be widely used. With this technique it is possible to determine chromosomal aberrations even are several years after a radiation exposure. Thus chromosome aberrations could be measured in survivors after the atomic bombing in Japan even today. Further the potential of this technique allows to study a number of fundamental radiobiological questions.

Besides these techniques for which a lot of experience and expertise is necessary an easier method has been developed: The determination of

micronuclei. With this technique it is not necessary to push the cells into the metaphase but micronuclei can be determined also in interphase cells. However, the cells have to go through mitosis before micronuclei are expressed. Micronuclei originate also from chromosome breaks with acentric fragments and from whole chromosomes which are located in the cytoplasm.

During last years a technique has been developed which includes a proliferation control. Cytochalasin B has been used in concentrations which allow the division of the cell nucleus but not the division of the cell itself. Under these conditions cells with two cell nuclei develop. The micronuclei which consist of chromatin particles in the cytoplasm are only determined in such bionucleated cells. The specificity of the expression of micronuclei is less than with dicentrics. Cells with micronuclei get lost with in a few weeks like cells with dicentrics. Therefore an exact measurement can only performed within a few weeks after radiation exposure. Although the sensitivity for the micronucleus determination is less than for other cytogenetic techniques it has been found very useful that the results are obtained quickly and comparatively easily. Further the micronucleus test can be used with an automated cell system.

With all these cytogenetic techniques it has been shown that besides estimates of radiation dose it is also possible to get a measure of the individual radiosensitivity. For instance in a number of persons with genetic syndromes like ataxia telangiectasia, Franconi anaemia and others which show a high degree of radiosensitivity it has been found that the expression of chromosomal aberrations including micronuclei is increased. Most of these genetic syndromes are expressed only in persons who have the mutation in both genes (homozygotes), the genetic mechanism is recessive and not dominant. However, it has been found that an increased radiation effect can also be seen in heterozygotes (mutation only in one gene) although the increase of the radiation effect is less in these individuals than in the homozygotes. This variability of individual radiosensitivity certainly will play a significantly larger role for radioprotective measures in the future. Therefore it is desirable to have such techniques available which give an information about the extent of radiation damage and to use this information for the decision whether therapeutic measures have to be applied or can be excluded.

It can be suggested from these experimental experiences that the comparative simple micronucleus assay is used for screening if many persons have been exposed by an accident. Under these circumstances quite a number of results can be obtained in a very few days after blood samples have been taken from the exposed individuals. The micronucleus assay can be used again in a dose range of 0.1 to several sieverts. As with chromosomes evaluations of micronuclei have been described for fast neutrons, also dose rate effects have been studied. After such a screening the blood of the exposed people can be investigated by looking for conventional chromosomal aberrations or using the FISH technique. Especially in the low dose range these methodologies certainly can give more exact data. In one accident it was possible to measure the radiation effect by the micronucleus test and the formation of dicentrics. Comparable results were obtained.

With these techniques it is certainly possible to get an information about a radiation exposure including the individual radiosensitivity which will be very useful for judgements of accidental situations.

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HEALTH ASPECTS OF CELLULAR MOBILE TELEPHONES

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INTRODUCTION

Cellular mobile telephones are one of the main topics among health aspects of electromagnetic fields. In many countries, the number of people opposing communication towers is on the rise. Lawsuits against telecommunication and power line companies have been filed. All this makes people doubt the safety of electromagnetic fields.

With respect to cellular phones, there are two scenarios:

☆ Exposure of the operators of hand-held terminals (HHT).

☆ Exposure of the general public from base stations (BS).

In the first case, the transmit antenna of the HHT is very close to the human body. For normal operation, the distance will roughly be 2 - 3 cm. The transmitter power of the HHT is comparatively low, but there is a considerable fraction of the radiated electromagnetic energy penetrating the tissue. Considering the second case, BS transmitter powers are by a factor of 100 - 1000 higher, but the distance between antenna and the human body is by a factor of 1000 - 100000 greater, as far as areas of unrestricted public access are concerned. As the power density of an electromagnetic wave decreases inversely proportional to the square of the distance, exposure of the public is always significantly (by many orders of magnitude) lower than exposure of operators of HHTs.

Some well-known interaction mechanisms of microwave radiation with the human body have been very well-established today. In some other areas, there is still a need for further research. This paper summarizes present knowledge on human safety with mobile telephone systems.

TECHNICAL CHARACTERISTICS OF CELLULAR MOBILE TELEPHONE SYSTEMS

Cellular mobile telephone systems can be categorized with respect to several parameters: Digital or analogue modulation, carrier frequencies, transmitter power, etc. Table 1 shows a survey about the technical characteristics of some selected, important systems used around the world today. We find that

- both analogue and digital systems are used today, but the trend clearly goes towards digital systems. With respect to biological effects, the key difference between the two kinds of modulation is that analogue systems use continuous transmission, while digital systems use pulsed transmission.
- Transmitter frequencies of HHTs are in the ranges of 441-456 MHz, 810-849 MHz, 872-915 MHz, 1.429-1.453 GHz, and 1.710-1.785 GHz. With respect to biological effects, a different carrier frequency makes a difference in the amount of energy absorbed in the tissue (see chapter about the heating effect below).
- Mean transmitter powers of HHTs never exceed 1 W, for most digital systems it is below 0.6 W. Moreover, most systems operate at even reduced power when the transmission channel is sufficiently good.

The data given in Table 1 summarizes the key characteristics of cellular telephones with respect to health aspects.

RELEVANT BIOLOGICAL EFFECTS

Based on present knowledge, the following biological effects can be considered relevant for human exposure to electromagnetic radiation from mobile telephone systems:

Local heating of tissue due to absorption of electromagnetic energy:

This effect occurs while using hand-held terminals. Results of scientific investigations will be discussed below. The effect does not occur with exposure of the general public by the electromagnetic radiation from base stations, as far as areas of unrestricted public access are concerned.

Interference of the electromagnetic field from hand-held terminals with electronic medical implants such as cardiac pacemakers or defibrillators:

The results of scientific investigations will be discussed below. Interference caused by the electromagnetic radiation from base stations does not occur, as far as areas of unrestricted public access are concerned.

Other effects that are known from in-vivo or from in-vitro experiments seem not to be relevant or have not been sufficiently well established yet to allow a conclusion to be made.

Changes in human EEG:

One researcher (1) reported a five-fold increase in the power density of the 10.7 Hz (α) - spectrum of the human EEG. Changes were observed during and after exposure with 217 Hz - modulated radiofrequency fields of a power density of less than 0.1 $\mu W/cm^2$. The carrier frequency was 150 MHz, the duration of the square-wave pulses were 100 μs. We have repeated the experiments described in (1), the same was done at several other research

Table 1. Selected public cellular mobile telephone systems that use HHTs

System	Countries	Mobile transmitter frequencies [MHz]	Mobile transm. mean rms output power [W] [1]	Modulation	Multiple Access	Analogue / Digital
NMT 450	Scandinavia, Denmark, France, Austria, Switzerl.	441-456 [2]	1	PM speech FSK contr. sign.		A
ETACS	U. K., Italy, Austria, Spain, Malta, Ireland	872-905	0.45 [3]	PM speech FSK contr. sign.	SCPC/ FDMA	A
NTT	Japan	915-940 [2]	1			A
NMT 900	Scandinavia, Denmark, Switzerland	890-915	1	PM speech FSK contr. sign.	FDMA/ TDMA	A
GSM	Europe, Asia, Pacific	880-915	0.25 [4]	GMSK	TDMA	D
CDMA	North America	824-849	0.6	QPSK	DS/CDMA	D
NADC	North America, Asia, Pacific	824-849	0.2	/4 DQPSK	FDMA/ TDMA	D
JDC (PDC)	Japan	810-826	0.3	/4 DQPSK	FDMA/ TDMA	D
JDC (PDC)	Japan	1429-1453	0.3	/4 DQPSK	FDMA/ TDMA	D
DCS 1800	Europe	1710-1785	0.125	GMSK	TDMA	D

[1]) All systems except NMT 450 use mobile power control. Example: GSM in Austria: Power reduction in 2 dB-steps as soon as -94 dBm (÷ 0,4 pW) is reached at the mobile receiver input.

[2]) Summary of frequencies used in different countries or regions, values rounded.

[3]) 0,6 W ERP = standard, approximately 1,5 dB antenna gain assumed.

[4]) Units with 2 W peak power. The standard would allow 5 W for HHTs, but there are no 5 W terminals on the market.

PM ..... phase modulation
FSK ..... frequency shift keying
GMSK ..... gaussian minimum shift keying
QPSK ..... quadrature phase shift keying
DQPSK ... differential quadrature phase shift keying
FDMA ..... frequency division multiple access
TDMA ..... time division multiple access
DS/CDMA direct sequence code division multiple access
SCPC ..... single channel per carrier

laboratories throughout the world. All results were negative, the findings of (1) were not confirmed, e. g. (2). Thus, at present, reactions to the observations reported in (1) would not be justified.

Influences on cell membrane permeability:

The transport of the cations potassium (K^+), sodium (Na^+) and calcium (Ca^{++}) from the interior of the cell to the intercellular space can be increased by amplitude modulated fields (1 - 100 Hz) in the frequency range of 50 MHz - 3 GHz (3, 4, 5). This has overall significance for the general cell function. The effect seems to be restricted to certain „windows" of power density and frequency. It seems to be a true non-thermal effect, as the results have been obtained even at specific absorption rates below 1 mW/kg. So far there has not been any mechanism demonstrated that could explain the phenomenon. Furthermore, the relevance of the effect for human health is totally unclear. Thus, the preliminary results about influences on cell membrane permeability cannot be used to derive conclusions regarding the use of mobile telephone systems at present.

Influences on nerve tissue:

The electrochemical potential of brain- and peripheral nerve cells can be affected by radiofrequency radiation (4, 5). It seems to be impossible to distinguish between thermal and non-thermal effects. The literature available today does not permit any conclusions regarding exposure with mobile telephone systems.

Influences on the immune system:

Several studies show that it has been possible to increase the level of macrophage activity or to inhibit the activity of natural killer cells by exposure to radiofrequency radiation. Effects seem to be mainly due to thermally induced stress when the whole-body temperature is increased by 1 degree Celsius or more. A chain reaction is then initiated from the hypothalamus. Results are not conclusive. At present it is not possible to derive any conclusions regarding mobile telephone systems.

Mutagenic and carcinogenic effects:

It has not been possible to demonstrate independent mutagenic effects (genetic damage) at cellular levels, even under weak hyperthermic conditions. An independent initiation effect on the DNA level with direct DNA damage has not been positively demonstrated after exposure to radiofrequency fields. The activity of the enzyme ornithine decarboxylase and the activity of nonspecific protein kinases can be affected by strong radiofrequency fields. Therefore, such fields cannot be regarded as an initiator of a carciogenic process on the cellular level. A possible promotion has not been proven, but, according to some published, preliminary results of investigations, it cannot be excluded (6). These studies are very controversial. The majority of the studies was done using specific absorption rates higher than those occuring with normal use of mobile telephones. On the basis of data available today it is not possible to derive conclusions regarding mobile telephones.

Most of the data regarding the effects mentioned above are inadequate for drawing any conclusions about possible health effects of cellular telephones. Further investigations will be necessary before judging the relevance of the above mentioned effects.

EXPOSURE OF OPERATORS OF HAND-HELD TERMINALS

Heating effect

Hand-held terminals are operated at distances between antenna and biological tissue of approximately 5 - 30 mm. The parts of the tissue (head) that are located close to the antenna absorb a significant amount of electromagnetic energy. This absorption of energy leads to a temperature rise in the tissue. Supply of energy and cooling by blood flow can compensate, if a certain amount of power per mass (specific absorption rate, in watts per kilogram) is not exceeded, and if the rate of blood flow (volume per time, in liters per minute) is sufficiently high. This criterion will be fulfilled for very low power transmitters (numerical values will be given below) and tissue with a high rate of blood flow. An example for this is the brain, which has a metabolism that is eight times the body average. The criterion might not be fulfilled for tissue with a low rate of blood flow such as the eye. Another relevant criterion is the sensitivity of the organ that is irradiated. Critical organs could be

- the eye (located at the body surface; high absorption due to protein; no blood flow);
- the inner ear (located very close to the antenna of the hand-held terminal; no absorbing tissue (only bone) between antenna and cochlear; high absorption in the liquid of the cochlear, which contains the hair cells; thermally isolated location in bone with only minimal blood flow);
- the hypothalamus (well-cooled by blood flow, surrounded by brain tissue with a high rate of blood flow; possibly critical organ because it controls a variety of functions in the body, e. g. circulation of blood, breathing, etc.);
- pineal gland (well-cooled by blood flow; possibly critical organ because it produces the tumor-inhibiting hormone melatonin).

Due to the heterogenity of the human head, the field distribution in the tissue is very complex. Today, complicated numerical techniques are used to calculate local specific absorption rate (SAR) and local temperature rise. Furthermore, results of measurements of electric field-strength and temperature rise in homogeneous and heterogeneous phantoms are available. The following, simple equations are used to determine SAR, once the field-strength has been calculated:

$$SAR = \frac{\sigma |E|^2}{\rho} = c \frac{dT}{dt}$$

where

$|E|$ ..... absolute value of electric field-strength in the tissue

σ ..... conductivity of the tissue; typical values of σ at 900 MHz are between 0.2 S/m (cortical bone) and 2 S/m (aqueous humour)

ρ ..... density of the tissue; for tissue with high water content, $\rho \approx 1$

c ..... heat capacity of the exposed part of the body; for tissue with high water content, $c \approx 4.0\text{-}4.2$ $J.kg^{-1}.°C^{-1}$

dT/dt .. time derivative of tissue temperature.

Published *results about SAR* are sufficiently consistent to draw reliable conclusions: The SARs in various kinds of tissue have been experimentally determined in phantom materials (7), see Figure 1. There is a strong frequency-dependence. The values in Figure 1 are valid for a typical separation distance of 2,5 cm between antenna and body (normal operation of a HHT) and irradiation from a half-wave dipole.

Tissues:

- ● muscle
- ■ brain
- ○ fat
- □ eye

Irradiation with half-wave dipole,
25 mm from phantom surface
7 W transmitter power

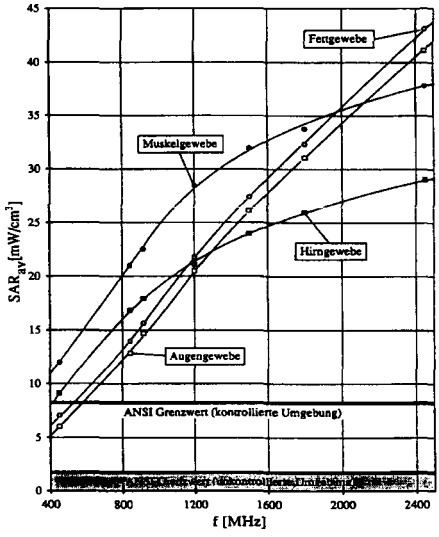


Figure 1: Specific absorption rates in various kinds of tissue, averaged over 1 g (7)

The *distribution of SAR in the head* has been studied by several authors. Figure 2 shows histograms of the SAR-distribution in a horizontal plane through the eyes at 800 MHz (8). The calculation was done using the finite-difference time-domain (FDTD) method. The model of the head consisted of 126.000 cubic cells with a side length of 3,2 mm each. The head is irradiated by a plane wave from the front. Maximum absorption occurs in the eyes. Another local maximum occurs in the center of the head. For the adult (Fig. 2a), the SAR in the center of the head is lower than the SAR in the eyes. For the child (Fig. 2b), the SAR in the center of the head is approximately the same as the SAR in the eyes. When the dipole is located in front of one eye, maximum absorption also occurs in this eye. The maximum in the center of the head disappeares at this near-field exposure. At higher frequencies, the fraction of power penetrating the head is less than at lower frequencies, absorption primarily occurs at the surface and within the first few centimeters of tissue.

Figure 3 shows the maximum values of SAR, averaged over 10 g of tissue, as a function of distance between antenna feed-point and the body (10). At a distance of 2 cm (realistic worst-case for normal operation) and a position of the HHT at the side of the head, the maximum is about 3 W/kg per W for 900 MHz and 4 W/kg at 1.8 GHz. A larger averaging volume results in lower SAR-values, a smaller volume results in higher SAR-values. According to another study, averaging over the eye in an exposure situation where the transmitter is held in front of this eye results in a maximum of approximately 6 W/kg/W at 900 MHz and 11 W/kg/W (adult) to 18 W/kg/W (child) at 1,8 GHz (9). From Table 1 we can see, that transmitter power is less than 1 W for all systems except NMT and NTT. Therefore, in particular for the digital systems, SAR will be lower than the values mentioned above. For, e. g., a GSM handy with 0.25 W average power, worst-case SAR in the eye will be 1,5 W/kg for 2 cm distance. If the antenna is brought even closer to the head, SARs will be higher: For a distance of 3,2 mm and averaging over 1 g of tissue, a local SAR as high as 30.4 W/kg per W has been calculated (9). Such a position of the HHT at the head anyway does not correspond to normal operation.

A more difficult task than the calculation of local SAR is the *calculation of local temperature rise*. Only very few results are available at present. In a recent study (11), the basic relation between SAR, temperature change and transmitter power has been shown, see Figure 4. The graph is valid after 30 seconds of exposure. For such short exposure time, temperature rises linearly with energy deposited in the tissue. Later, thermal equilibrium will be reached. The final amount of temperature rise will depend on a variety of parameters such as the kind of tissue, environmental temperature, cooling by blood flow, etc. In another study (12), the spatial SARs, averaged over one cm^3, were calculated for the human eye, and then the heat conduction equation was solved using the implicit alternating-direction (IAD) algorithm. Irradiation from a half-wave dipole at frequencies of 840 MHz, 915 MHz, 1500 MHz and 1800 MHz led to temperature rises as shown in Figure 5a. For a continuous transmitter power of approximately 0,7 W and a distance of 2 cm between antenna and surface of the eye, maximum temperature differentials reach approximately 0.2° C at 840 MHz, 0.25° C at 915 MHz, 0.6° C at 1500 MHz and 0.65° C at 1800 MHz. For a distance of 6.5 mm and a CW transmitter power of approximately 0.8 W, maximum temperature elevation reaches approximately 1° C at 840 and 915 MHz and approximately 2.2° C at 1500 and 1800 MHz. A maximum absolute temperature of 39° C at 1500 MHz was calculated in the posterior region of

a) Head of an adult

b) Head of a child

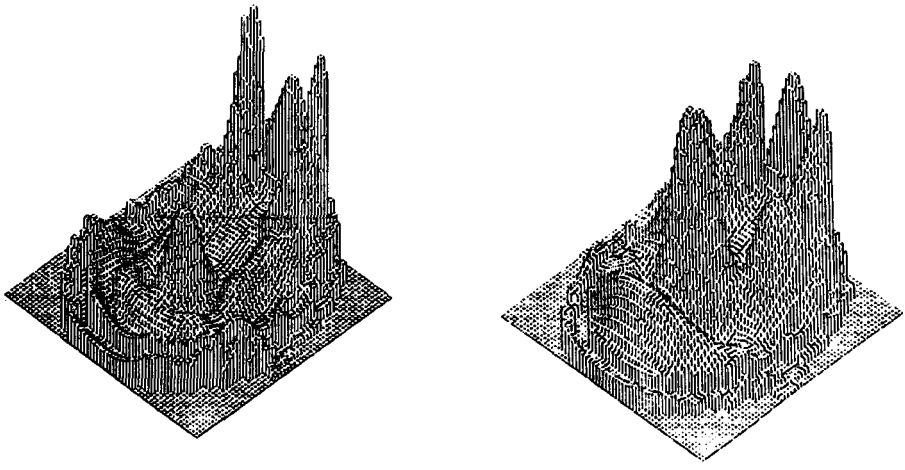


Figure 2. Histograms of the SAR in an anatomical model of the human head that is irradiated by a plane wave from the front at 800 MHz, horizontal cross-section through the eyes (8).

Asterisks: HHT and antenna horizontally beside the left ear (chassis at front, antenna at back of the head)
Diamonds: HHT and antenna vertically in front of the left eye (antenna up)
Triangles: HHT and antenna vertically in front of the left ear (antenna up)

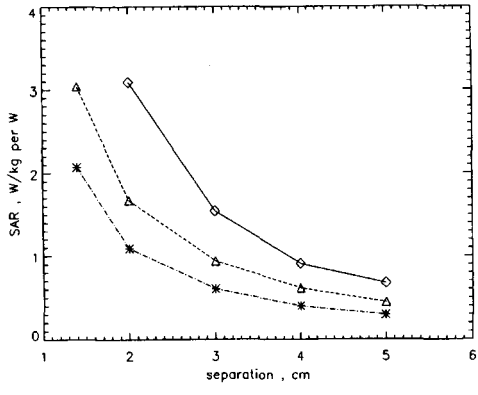


Figure 3. Maximum values of SAR, averaged over 10 g of tissue (10)

the eye for 6.25 mm distance. Lower transmitter power will result in lower temperature rise and lower absolute temperature, see Fig. 5b: For 2 cm distance and 1500 MHz, temperature rise for a peak transmitter power of 0.7 W and a duty cycle of 1:8 (GSM) is only about 0.04 °C, where it has been about 0,6 °C for 0.8 W CW. Thus we can estimate that the maximum temperature rise in the eye for, e. g., a 2 W - GSM handy with 0.25 W average power at 2 cm distance will not exceed 0.1 to 0.2 °C.

Induction of cataracts in the eye has been shown to occur at temperatures of 41° C or more. Considering the worst-case data given above (12), there seems to be a sufficient safety factor between temperatures occuring with the use of HHTs and temperatures that must be regarded hazardous for the eye.

Temperature elevations in other, possibly critical organs, such as the inner ear, the hypothalamus, etc., have not yet been precisely calculated. This is an important, remaining task that should be carried through, before final judgements about thermal effects, exposure limits, or restrictions on transmitter power can be made.

Interference to Medical Implants

Electromagnetic fields can interfere with electronic apparatus. We have studied 9 scientific publications and symposium contributions about experiments with cardiac pacemakers and defibrillators, e. g. (13, 14, 15, 16). In these studies, a total of 363 tests have been carried out using actual implants in phantoms. In 116 cases interferences occured, this corresponds to 32 %. Among these, 40 (11 %) caused inhibition, 33 (9 %) caused asynchronous pacing, 36 (10 %) caused synchronization, and 7 (2 %) caused combinations of the three effects. All tests were done at very small distances between HHT and surface of the phantom, i. e. representing a HHT held close

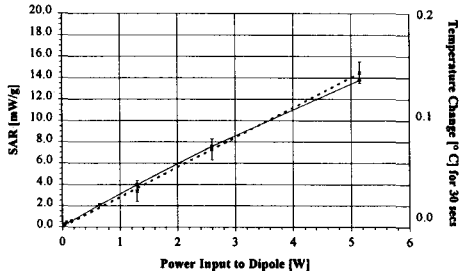
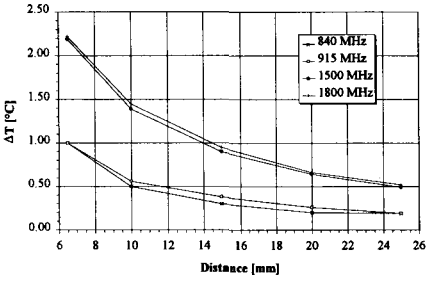
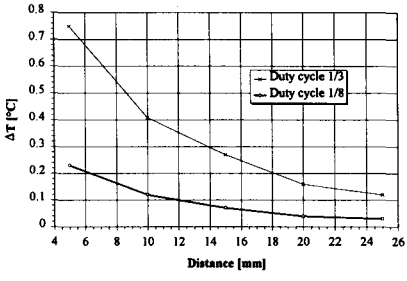


Figure 4. Typical values of specific absorption rate and temperature change versus transmitter power, measured in a phantom irradiated by a half-wave dipole at 840 MHz (11)



a) 4 different frequencies, peak transmitter power approximately 0.8 W



b) 2 different duty cycles at 1500 MHz, peak transmitter power approximately 0.7 W

Figure 5. Maximum temperature differentials in a model of the human eye for various distances between a dipole antenna and the body surface (12)

to the chest. The above mentioned statistics do not mean that one third of all pacemaker patients have to expect interferences when using a mobile telephone. Most of the tests were carried out in-vitro, and worst-case configurations were used. Enlargement of the separation between HHT and the pacemaker eliminated the interference in all cases. Recommendations for the safety distance range between 10 cm and 40 cm.

Interference to a pacemaker can mean a severe health hazard to the patient:

- Switching to the asynchroneous mode during normal heart activity can trigger vertricular fibrillation, if the impulses appear within the vulnerable phase of the heart. This certainly indicates serious danger.
- Synchronization can lead to wrong stimulation of ventricles. This can make patients feel unwell and effects of weakness and fainting can be expected.
- Inhibition would also result in weakness and fainting.

A major reason for the interferences is the amplitude-modulation of the mobile telephone signals. Therefore, the majority of interferences were found with digital systems. The most dangerous pulse repetition rates are 2 - 8 Hz. Pacemakers can interprete such signals as signals from the heart, whereas they usually filter out components at, e. g., 217 Hz, the normal pulse repetition rate of GSM. Frequency components at 2 - 8 Hz occur during signaling or for instance in the DTX-mode of GSM (discontinuous transmission). In this mode, the number of frames is reduced as long as the user is silent. Also the ringing phase poses a risk to the pacemaker patient. As analogue systems also use digital signaling, interferences are not restricted to digital systems. For analogue systems, the dialing phase appeared to be most critical.

Different kinds of pacemakers show very different reactions to exposure to mobile telephone signals. It is very important that pacemaker patients receive detailed information about possible interferences. Keeping the HHT away from the chest and not carrying it in the breast pocket could be a general safety recommendation for pacemaker patients. Moreover, doctors responsible for implantations should give recommendations for measures to be taken in case of undesired exposure.

EXPOSURE OF THE GENERAL PUBLIC

Cellular nets use cell sizes of typically 1 - 10 km in diameter. The trend goes towards smaller size cells, a higher number of base stations, and consequently lower transmitter powers at both HHT and BS. Antennas of BSs are located on masts on the top of high buildings in order to ensure good transmission. Antennas are constructed

so that their directivity enables maximum radiation in the horizontal plane and minimum radiation towards the ground. The surrounding of such an antenna must be an obstruction-free space for at least several tens of meters. Thus, exposure in areas with public access can never be in the main beam of the antenna.

General public exposure to radiation from BSs is very comparable to exposure from television (TV) stations. The highest-frequency TV channels in Europe are at 860 MHz, which is within the frequency range used for cellular phones as well. Figure 6 shows electric field-strengths measured in a typical urban environment (Vienna, 4th district). FM radio stations, TV stations and cellular telephone signals typically range between 60 and 100 dBμV/m, corresponding to 0,01 - 0,1 V/m. Local maximum values can reach 1 V/m. The public has been exposed to such signals for decades. Looking at the ambient frequency spectrum shown in Figure 6 we can see that signals from cellular telephones are an additional contribution to the total exposure, but they appearently add just a very small percentage.

Let us try to evaluate the SAR for this exposure situation. For far-field exposure at 900 MHz, a field-strength of 0,1 V/m causes a whole-body averaged SAR of approximately 10^{-7} W/kg. In the frequency range of 30 MHz to approximately 300 MHz, a field-strength of 0,1 V/m causes approximately 10^{-6} W/kg. If we were to add up 1000 radiofrequency signals, total SAR would still be below 1 mW/kg, whereas the present, well-agreed limit value for the general public is 80 mW/kg. Thus it is evident, that there is a large safety factor between actual signal levels and recommended safety limits.

Also, TV transmitters have been well-accepted through the years and there were no diseases or health hazards reported.

Entering the near-zone of a BS antenna by, e. g., climbing an antenna mast, requires seperate safety considerations. Here, field-strengths can exceed the safety limits. Careful operation is required as is the case at any radio transmitter or at any industrial location where electromagnetic fields are applied. In areas of unrestricted public access, such situations should not occur.

From the above considerations we can conclude that, based on present knowledge, exposure of the public to the electromagnetic radiation from base stations appears to be no health risk.

88-108 MHz: FM radio stations
170-230 MHz: TV channels
470-860 MHz: TV channels

120 dBμV/m ÷ 1 V/m

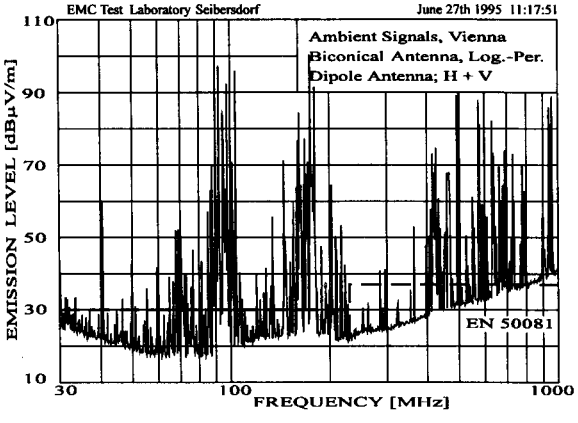


Figure 6. Ambient signals measured in Vienna, 4th district

SUMMARY AND CONCLUSIONS

The maximum, localized specific absorption rate in the head of a user of a modern cellular telephone reaches 6 W/kg per W transmitter power with normal use (hand-held terminal in a position at the head, where one can speak into the microphone and listen to the loudspeaker; distance between antenna and body surface not less than 2 cm). The strongest hand-held terminals available on the market today use 1 W average transmitter power, most types use less. For a typical GSM handy, e. g., maximum specific absorption rate is 1.5 W/kg in such an exposure situation. The maximum local temperature rise in possibly critical organs in the head, such as the inner ear, the hypothalamus, etc, associated with a certain specific absorption rate cannot be precisely specified yet. Results of scientific investigations are available for the eye. Here, for a distance of 6.25 mm (worst-case), a maximum absolute temperature of 39 °C has been numerically calculated for a continuous transmitter power of 0.7 W at the highest frequencies used in cellular mobile communications today (calculation for 1500 MHz, data at 1800 MHz are almost identical). This maximum occured in the posterior region of the lens. Cataracts are known to occur at temperatures of 41 °C or more, thus there seems to be a sufficient safety factor. The temperature actually occuring with normal use of cellular telephones is well within a range that the human body is frequently exposed to. A

healthy body is able to properly react by thermoregulation. Based on today's knowledge, there is no indication of hazard when hand-held terminals are used in a normal way.

As a precaution, a warning should be expressed regarding a possible exposure situation such as the antenna of a hand-held terminal being pressed against the eye or another part of the head for several minutes or longer. In this case, specific absorption rates can reach 30 W/kg per W or more, which could mean a hazard due to excessive, local heating. This should anyway be treated as a special case of misuse.

A number of further biological effects of microwave radiation has been observed during in-vitro and in-vivo experiments. The relevance of these effects for human safety is totally unclear at present, thus these effects cannot be used to derive safety criteria. The studies are very controversial, clear indications of hazards have not been demonstrated. As a precaution, such effects should however be investigated further. If reliable results become available and they have an impact on human health, they must be accounted for in relevant safety guidelines and standards.

An explicit warning must be expressed to any person with electronic, medical implants. For switched-on hand held terminals a safety distance between antenna and implant should always be kept, as long as immunity of the device has not been clearly demonstrated. The required safety distance should be specified for each type of pacemaker, defibrillator, etc. Typically, it is of the order of 10 - 40 cm. Patients should avoid carrying a hand-held terminal in their breast pocket. Operation at the ear (normal use of the telephone) has been demonstrated to be safe for most types of pacemakers.

Public exposure to radiation from base stations does not cause any hazard. The level of exposure is far below the threshold for well-established effects. Moreover, the public has been exposed to very similar electromagnetic fields (broadcast radio and television) for decades and no adverse effects have been reported.

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ESTABLISHED BIOLOGICAL EFFECTS OF EXTREMELY LOW FREQUENCY (ELF) FIELDS, CURRENT PROTECTION CONCEPTS AND RESEARCH NEEDS

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INTRODUCTION

Public concern about possible health hazards from exposure to electromagnetic fields is growing. However, although several epidemiological studies suggest a weak association between the exposure to extremely low-frequency (ELF) fields and an increase in various kinds of cancer, a final risk assessment of long-term continuous exposure to ELF fields is not possible so far. There has not been a definite proof yet that the electric and especially magnetic ELF fields occurring at working places or in every-day life are mutagenic or carcinogenic. Questionable points are those concerning statistical evaluation, insufficient determination of the field strength during exposure, dose-effect relationship, inadequate demarcation of concomitant factors, and - most important - the absence of known interaction mechanisms (for further details see contribution of J. Stolwijk in this book). A final clarification of the question of possible late effects requires further elucidation. The non-stochastic ELF field effects, therefore, are of major importance for the derivation of standards (7).

INTERACTION MECHANISMS

Direct coupling mechanisms: Electric fields

Biological material is a much better conductor than air, therefore an electric field is distorted by the presence of a living being in the field.

Due to polarisation effects electric fields act on the outer surface of the body and lead to hair movements and sensory effects. In addition, time varying electric fields induce surface charges on an exposed body which result in currents inside the body, the magnitudes of which are related to the surface charge density. Depending on the exposure conditions, the surface charge density can vary greatly resulting in a non-uniform distribution of currents inside the body. For sinusoidal electric fields the magnitude of the currents inside the body increases proportionally with frequency. The induced current density distribution varies inversely with the body cross-section and may be relatively high in the neck and ankles of people.

In a wide variety of materials the current density $\underline{J}$ is directly proportional to the field strength $\underline{E}$, $\underline{\mathbf{J}} = \sigma \underline{\mathbf{E}}$, σ is the frequency dependent electrical conductivity of the medium.

Direct coupling mechanisms: Magnetic fields

The permittivity of biological material and air is about the same, therefore magnetic fields, unlike electric fields, penetrate biomaterial without any loss. Time varying magnetic fields induce electric fields and those cause currents inside the body. The internal tissue current density $\underline{J}$ is related to the external magnetic induction $\underline{B}$, using the Maxwell equations, its absolute value can be approximated by : $\mathbf{J = 0.5\ R\ \sigma\ dB/dt}$, where R is the radius of the inductive loop perpendicular to the direction of the magnetic induction.

The appropriate inductive loop radius depends on the orientation of the magnetic field relative to the body. Theoretically, the largest current densities will be induced in the peripheral tissues and will decrease linearly towards the center of the body as the inductive loop radius decreases.

Indirect coupling mechanisms

ELF electric fields interact with biological bodies through electric charges induced on ungrounded metallic objects, such as cars, trucks, cranes, wires and fences. Two types of interaction may occur:

- a spark discharge between the object and the person touching the object;
- the passage of current to ground through a person coming into contact with such an object; the magnitude of the current depends on the total charge on the object and on the person's impedance to ground. This charge, in turn, depends on the frequency, the electric field strength, the object geometry and its capacitance.

Interferences with medical devices

In addition to direct and indirect coupling mechanisms of low frequency fields, interference with active and passive medical devices has to be taken into account. For persons with implanted electronic devices, such as pacemakers, insulin pumps or cochlea implants and for persons with ferromagnetic implants, e.g aneurysm clips or protheses, magnetic fields may pose potential health risks. For example pacemakers in demand-mode may be activated; if the pacemaker rhythm then is asynchronos to the heart activity, this may pose a certain risk. For worst case situations the interference field strengths for 50 Hz fields were determined to be 1.7-6.8 kV m^{-1} resp.13.8-55 μT (8). Very slow fields may also inhibit continuously working pacemakers, especially for frequencies of some Hz, which is close to the heart activity.

BIOLOGICAL EFFECTS

Cellular Studies

Extremely low frequency electromagnetic fields induce time varying distributions of electric charges in tissues and electrical currents. The local electrical field strength respectively the local current density are responsible for any possible effects. Basically all organisms can be influenced by charges, because all molecules contain charged groups and the change of electric potentials plays an important role during chemical and biochemical reactions. In order to examine the interaction of electromagnetic fields and biological systems, the cell, the cell membrane or molecules are possible sites to be inspected.

The cell membrane is a barrier between the cytoplasm with a specific composition of ions and the extracellular medium. The lipid bilayer therefore shields chemical substances and charges. The cell has a negative potential with respect to the extracellular side, in most cases the potential difference is about -70 mV. Consequently there is an electric field strength of about 10 V m^{-1} across the 8 nm of the membrane. Natural biogenic electrical fields result from potential differences at cell membranes e.g. action potentials of nerves and muscle cells. Close to such cells, field strengths up to 200 V m^{-1} can be measured. In the periphery of the heart or brain even a field strength of 50 mV m^{-1} or current densities of 10 mA m^{-2} are possible. It is speculated that these relatively large natural fields are part of an extracellular communication between electrically excitable cells (10).

If the current densities in tissue, induced by electromagnetic fields, are large enough to depolarize the membrane potential up to the excitation threshold, a well defined biological reaction will follow. The mechanisms and thresholds of nerves and muscles are well known in classical electrophysiology. At levels of induced current density exceeding 100 to several hundred mA m^{-2} for frequencies between about 10 Hz and 1 kHz, thresholds for neuronal and neuromuscular stimulation are surpassed. The stimulation for single nerve- and cardiac muscle cells are well above 1 A m^{-2} (11). The threshold current densities increase progressively at frequencies below several Hz and above 1 kHz.

In contrast there is little known about the consequences and the possible mechanisms of interactions between induced, weak electrical fields and biological systems. The induced fields are permanent present and often possess higher frequencies than endogenous fields do (4).

At the moment the hypothesis is that the extracellular surface of the membrane is the target for interaction, it is based on the results of numerous cellular studies using weak magnetic field exposures. It is agreed that receptors in the membrane will be influenced and highly specialized transduction pathways can transform information into the cell. The primary process of the field - protein - coupling mechanism in not understood.

The membrane receptors are specified to transduce one single input signal (e.g. enzyme molecule) to a dominant output message. This unique biological amplification of small signals, lead to the development of many different specific receptor types. Therefore field induced modulation of this transduction pathway leads to different types of reactions without systematical relation.

In experiments, with current densities above 10 mA m^{-2} the following basic effects were produced

- influence of regulation and messenger systems of the cells,
- interference with protein synthesis,
- influence on cell growth and differentiation,
- complex physiological phenomena.

But also exposure with weak magnetic fields below 100 μT (inducing current densities less than 2 mA m^{-2}) may modulate some biological reactions. An evaluation of the various studies with weak fields, is difficult as the results of other labs were not reproduced and the effect found do not differ clearly enough from normal biological variations. Additionally, other experimental parameter proofed to be more effective stimulants.

Animal studies

Many biological endpoints have been examined using exposed rodents or other mammals. There is little evidence to suggest that low level exposure causes any consistent effect on the heart, lungs or immune system.

Various investigations have shown that power frequency fields may affect circadian rhythms via the hormone melatonin (12, 18).

There is some concern about the possible adverse effects of exposure to fields on reproduction and development of embryos and fetuses of chickens but not of mammals (9).

It has been suggested that magnetic fields may increase the risk of cancer incidence by reducing the oncostatic potential of melatonin (14, 18). Studies on carcinogenesis excluded the possibility of initialising cancer by electromagnetic fields, however fields may have a promoting or a co-promotor capacity leading to increased cellular proliferation (9).

Electric Fields - Studies On Volunteers

ELF electric fields can be perceived because of the field-induced vibration of body hair, or the occurrence of spark discharges on contact with clothes or grounded objects. The threshold for perception by hair vibration shows wide individual variation: Only 5% of exposed subjects are able to detect fields as low as 3-5 kV m^{-1} (2, 16). The threshold depends on the relative positions of the head, trunk and limbs, simply as a consequence of the different perturbations of the incident field.

Above a certain threshold, the current to ground is perceived by the person as a tingling or prickling sensation in the finger or hand touching the charged object, for frequencies below about 100 kHz, and as heat at higher frequencies. Although these effects are not considered to be a hazard, hair vibration and tingling if sustained can become an annoyance (5).

A severe shock can be experienced at levels much higher than threshold. The threshold current depend on frequency and on surface of contact area. The thresholds for women are about one third lower and the thresholds for children about 50% lower than the equivalent thresholds of men (3).

Of greater biological significance may be the occurrence of capacitive spark discharges (microshocks) which are generated when two objects of different potential come into close proximity and the electric breakdown field strength of the air is exceeded. These spark discharges will continue as long as the air gap and the potential difference is maintained. The current flows across a very small area of skin and results in a high current density which may be perceptible, irritating or painful. Exposed people may demonstrate stress reactions in the presence of repeated spark discharges with increased nervousness and inability to continue to work.

The threshold for the perception of spark discharges by 10 % of a group of volunteers close to an earthed object has been reported to be 0.6 to 1.5 kV m^{-1} at 50 to 60 Hz, with a similarly defined threshold for annoyance of 2 to 3.5 kV m^{-1} (2). Sensitivity appears to depend on such factors as skin hydration, body location and skin temperature.

There are also extensive laboratory tests on volunteers with better defined exposure conditions, but limited exposure time. The laboratory tests include field exposure times from 3 h to 1 week and field strengths of up to 20 kV m^{-1}. The variables examined were reaction time to optic and acoustic stimulation, physiological factors, EEG, ECG, blood pressure, pulse frequency, body temperature, blood status, biochemical parameters of blood and urine, enzymes and metabolic factors. No significant changes were found.

Magnetic Fields - Studies On Volunteers

At frequencies below approximately 100 kHz, interactions of time varying magnetic fields with biological systems and potential hazards can be considered in terms of induced currents and current densities. The use of induced current densities, is appropriate for the assessment of acute, immediate effects, while it may have some limitations for the complete evaluation of long-term effects. The waveform of the electromagnetic field and peak instantaneous field strengths are important factors to be considered in the response of biological systems. Generally, for frequencies above 100 to 1000 Hz, the thresholds for stimulation effects increase with frequency, up to frequencies where thermal effects dominate (> 100 kHz).

The ability of people to perceive 50 or 60 Hz magnetic fields has been studied by several groups. Visual phenomena (magnetophosphenes) caused by magnetic fields depend not only on the frequency and flux density of the magnetic field, but - with eyes open - also on adaption condition or background illumination. The sensitivity thresholds were found to be frequency dependent with a minimum threshold (maximum sensitivity) of approximately 5 mT at 20 Hz, although threshold values as low as 2 mT have been quoted, rising at lower and higher frequencies. The threshold current density at 20 Hz is estimated to be 10 mA m^{-2}. The investigations indicate that visual phenomena are not released by the magnetic field, but rather by the induced currents influencing the bipolar ganglion cells of the retina. This also explains the agreement between magneto- and electrophosphenes which may result from alternating currents produced by electrodes (5).

Following the development of clinical magnetic resonance imaging (MRI) diagnostic techniques, patients are routinely exposed during examination to intense (up to several mT), time-varying longitudinal and transverse gradient magnetic fields, pulsed at frequencies below about 10 kHz. Experiments have been carried out using volunteers in order to identify thresholds for peripheral nerve stimulation. Small, involuntary muscular twitches of the nose, and sometimes of the lower back and thigh, were reported. Threshold values depend on the orientation of the magnetic field relative to the body and have been estimated to range between

about 25 and 60 T s^{-1}, the lower values corresponding to transverse magnetic field orientation and to current densities exceeding 500 mA m^{-2} (5).

The exposure of volunteers for several hours to 50 or 60 Hz fields of up to 5 mT had no effect on a number of clinical and physiological tests, including haematology and blood biochemistry, ECG, heart rate, blood pressure and body temperature; elevated triglyceride levels were ascribed to other, confounding factors (5, 17).

It may be assumed on the basis of all known effects that man is influenced by low-frequency magnetic fields (50/60 Hz) of more than 5 mT. Volunteers exposed to intense ELF magnetic fields (up to 100 mT) at frequencies between 5 and 50 Hz have reported „indisposition and headaches" during exposure above 60 mT. Changes were also recorded in the visually evoked potential response at this magentic flux density (corresponding current densities of 100 mA m^{-2}, although visual acuity itself was apparently unaffected. Reversal of VEP's were observed that are reported not to be immediately reversible.

CURRENT PROTECTION CONCEPTS

Health implications and basic restrictions

Any discussion on the restriction of exposure has to be built on a certain protection concept. For non-ionizing electromagnetic fields usually the health threshold based system is applied: that is adverse health effects are identified together with the type and magnitude of the fields which cause this effect.

Given the inadequacy of the available data on carcinogenicity, exposure limits for electric and magnetic fields are currently based on well established mechanisms and experimental findings related to acute effects. The published guidelines are based on the limitation of electric fields or current densities induced in the body by exposure to electric or magnetic fields. Nevertheless, by considering the merits of the induced current approach for risk assessment, one should not mistakenly conclude that alternative explanations for other phenomena are impossible.

The following observations can be made for induced current density ranges and magnetic flux densities induced by sinusoidal homogeneous fields (17):

- Below 10 mA m^{-2} are the naturally occurring endogeneous currents of the body.
- Between 10 mA m^{-2} and 100 mA m^{-2} there are well established effects on the visual and nervous system.
- Between 100 mA m^{-2} and 1000 mA m^{-2} changes in the excitability of nerveous tissue is observed and there are possible health hazards.
- Above 1000 mA m^{-2} nerve stimulation and stimulation of the heart including extrasystoles and ventricular fibrillation can occur (acute health hazards).

Based on these data, levels of exposure are recommended which lie well below the levels of nerve excitation. In order to chose an approriate safety margin for such a recommendation, the following factors have to be considered:

- It is difficult to correlate precisely tissue current densities with external fields.
- There is a wide difference in perception thresholds, e.g. between children and grown-ups.
- Extrapolation from acute short term effects to long term exposure.
- Combined effects of exposure to fields and other chemical or physical agents of the environment.
- Medicated persons may have different responses.

Furthermore, experiments are conducted for specific frequencies only so that the whole frequency range and especially all possible modulations cannot be covered with sufficient data. The available set of data mostly is not obtained from experiments with human beings.

In the frequency range between 10 Hz and 10 kHz, for levels of induced current densities above 100 mA m^{-2} the thresholds for possible acute health hazards are surpassed. Given the

considerations above a safety factor of 10 was chosen and it was decided, that for human exposure only current densities below 10 mA m^{-2} should be allowed.

This basic restriction is a factor of 10 below serious acute effects like reversal of VEP and below the lowest threshold of changes in central nervous system excitability. Furthermore this basic restriction is a factor of 100 below the threshold of definite health hazards (ventricular fibrillation).

Then there are additional reasons to adopt lower exposure limits for the exposure of the population than for the ocupationally exposed persons. The main reason are the following: the general public comprises individual persons of all ages and different health status, individuals or groups with particular health susceptibility may be included in the general population. Exposure of the general public may last all day long, where as occupational exposure is restriced, usually to 8 hours a day for 5 days a week. Finally the public cannot be expected to take any risks or to accept any annoyances e.g. from contact currents. Thus for the exposure of the general public another factor of 5 was chosen, which leads to maxium current densities of 2 mA m^{-2}, corresponding to the current densities naturally occurring in the tissues and organs of the body.

The current densities should be averaged over a cross-section of 1 cm^2 perpendicular to the current direction. This averaging seems to be sufficient to include spatial peak values due to differences in the conductivity and in view of the fact that the effects of current densities are occuring at the cellular levels and that the safety zone from stimulating effects is sufficiently large.

Derived exposure levels

In practice, values of „field" quantities are required since the basic quantities cannot be measured directly in the human body. The derived exposure levels is the permissible value for the electric or magnetic field strength which is derived from basic restrictions. Derived exposure levels are given to provide a method of assessment for the effect of electromagnetic fields and to demonstrate compliance with the basic restrictions of exposure. Said exposure levels have been specified in such a manner that the basic restrictions are not exceeded even if the most unfavourable conditions of field exposure are taken as a basis.

Since the first attempt to relate external exposure to internal current densities (1) several numerical and measuring methods were developed for the derivation of field strength exposure limits from the basic restriction. For both methods considerable simplifications have been used up to now that did not account for phenomena such as the inhomogeneous distribution and anisotropy of the electric conductivity and other factors.

Magnetic field models assume that the body has a homogeneous and isotropic conductivity and apply simple circular conductive loop models to estimate induced currents in organs and body regions, i.e., for the head by using the relation $\mathbf{J} = \pi \mathbf{R} \sigma \mathbf{f} \mathbf{B}_0$ (amplitude of the field $\mathbf{B}_0 = \sqrt{2} \cdot \mathbf{B}_{rms}$). More complex models use an ellipsoidal-shaped model (11) to represent the trunk or the whole body for the estimation of induced currents resulting in higher current densities which occur at the periphery of a body. Results are summarized in table 1 for 50 Hz. The exposure level of 0.5 mT for occupational exposure or 100 µT for general public exposure recommended by (7) corresponds to maximum current densities of about 8 mA m^{-2} or 1.6 mA m^{-2}, respectively, in peripheral regions of the trunk. Referring to more realistic calculations based on an anatomically and electrically enhanced model (15) result in higher current densities (i.e., 0.5 mT induce 12 mA m^{-2} at 60 Hz).

Electric field models have to take into account that depending on the exposure conditions, size, shape and position of the exposed body in the field, the surface charge density can vary greatly resulting in a variable and non-uniform distribution of currents inside the body. Thus for sinusoidal electric fields below about 10 MHz the factor of proportionality σ between the

current density and the external field strength should be replaced, so that, $\underline{J}=A\ f\ \underline{E}_{ext}$, where $\underline{E}_{ext}$ is the external unperturbed electric field strength and A is a „shape factor" for human body parts of specific shape and a specific orientation.

The factor A can be determined by model calculation (including measurements) for different parts of the body: for the trunk (average) about $6.7 \cdot 10^{-9}$ S Hz^{-1} m^{-1}, for the ankle about $40 \cdot 10^{-9}$ S Hz^{-1} m^{-1} for the neck about $10 \cdot 10^{-9}$ S Hz^{-1} m^{-1} and for the head $2 \cdot 10^{-9}$S Hz^{-1} m^{-1}. These numbers show that the induced current density distribution varies with the body cross-section and may be relatively high in the neck and ankles of people. The electric field strengths which approximately procuce current densities of 10 or 2 mA m^{-2}, respectively, in different body parts are listed in Table 1. The exposure level of 5 kV m^{-1} for continuous general public exposure recommended by (7) corresponds to a current density of about 2.5 mA m^{-2} in the neck and about 1.7 mA m^{-2} in the trunk (average) if the E-field vector is parallel to the body axis.

Table 1: 50-Hz-electric field strengths and magnetic flux densities producing approximately current densities of 10 mA m^{-2} or 2 mA m^{-2}, respectively, in different body parts (6)

	Trunk (average)	Head	Neck	Ankles, when both feet are grounded	Whole Body (ellipsoidal model)	Head	Wrist/ Ankle
	Shape factor in 10^{-9} S $Hz^{-1}m^{-1}$				$\sigma = 0.2$ S m^{-1}		
Parameters for calculation	A = 6.7	A = 2	A = 10	A = 40	a =0.2 m b =0.85 m c =0.1 m	R=0.075m	R=0.03m
Current density	Electric field strength (kV m^{-1})				Magnetic flux density (mT)		
10 mA m^{-2}	30	100	20	5	0.6	2.5	6
2 mA m^{-2}	6	20	4	1	0.12	0.5	1.2

RESEARCH NEEDS

Recently research on interaction mechanisms of electromagnetic fields has increased enormously. In spite of the large amount of publications there still is no convincing answer, how weak electromagnetic fields in the body can lead to an interaction even below electric stimulation thresholds.

Based on numerous studies with exposure to weak magnetic fields it is hypothesized by now that the induced currents and voltages in the extracellular medium interact with electrochemical processes at the cell surface.

Some biological effects can only be evoked by fields with a narrow set of parameters therefore reaction windows were postulated. Experimental data then have to be used to test the existence of nonlinear intensity-reaction-relations. Only after the proof of this hypothesis a transfer to other biological systems is possible.

The weak fields obviously do have a very differential interaction with the metabolism of organisms. Various effects only lead to a minor temporal modulation of biological reactions. The measurement of different physiological, parameters showed heterogeneous and contradictory results.

Therefore there still is a need for fundamental research to find and to define one or more primary mechanisms. For example it should be found out, if it is the magnetic field or the induced electric field which is responsible for a reaction.

Only with the knowledge of the primary reactions, a systematic concept of research projects can be established for examining the pathway of the biological signal to the manifested effect and for testing the possibility of transduction with subsequent amplification cascades. It has to be elucidated for example which physiological functions are involved and if cells have different sensibilities.

Studies on immune and nervous systems then have to show if those single cells effects can influence the circulation system or the embryonic development or the whole organism have to be tested with exposure conditions.

Central nervous system responses to induced electric current should be further investigated including, for example, *in vitro* brain slice studies of weak electric current effects.

There is a particular need for studies on the possible carcinogenicity of electromagnetic fields. The experimental work should include co-carcinogenesis experiments, particularly looking at co-promotion in animal and cellular models, studies of effects on cell signalling and proliferation, Ca^{2+} uptake and gene expression, and further study of effects on melatonin and its possible role in the suppression of mammary tumour growth.

Further investigations of possible weak ELF field interaction mechanisms particularly of experimentally testable hypotheses such as magnetic field effects on radical pair interactions have merit as well as studies of weak electromagnetic fields interactions with the optical system, in relation to effects on the production of melatonin by the pineal gland.

After the field induced changes or modulations of biological systems are scientifically established, it has to be found out, whether these evoked effects are potentially pathogenic.

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BIOLOGICAL EFFECTS OF ELECTROMAGNETIC FIELDS

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INTRODUCTION

The effects of electromagnetic (em) fields on biological systems were first observed and exploited well over a century ago. Concern over the possible health hazards of human exposure to such fields developed much later. It is now well known that excessive exposure to em fields may have in undesirable biological consequences. Standards were introduced to determine what constitute an excessive exposure and how to avoid it. Current concern over the issue of hazards stems mainly from recent epidemiological studies of exposed populations and also from the results of laboratory experiments in which whole animals are exposed in vivo or tissue and cell cultures exposed in vitro to low levels of irradiation. The underlying fear is the possibility of a causal relationship between chronic exposure to low field levels and some forms of cancer. So far the evidence does not add up to a firm statement on the matter. At present it is not known how and at what level, if at all, can these exposure be harmful to human health. This state of affair does not provide a basis for incorporating the outcome of such research in exposure standards.

This paper will give a brief overview of the research in this field and how it is evaluated for the purpose of producing scientifically based standards. The emphasis will be on the physical, biophysical and biological mechanisms implicated in the interaction between em fields and biological systems. Understanding such mechanisms leads not only to a more accurate evaluation of their health implications but also to their optimal utilisation, under controlled conditions, in biomedical applications.

INTERACTION OF EM FIELDS WITH PEOPLE

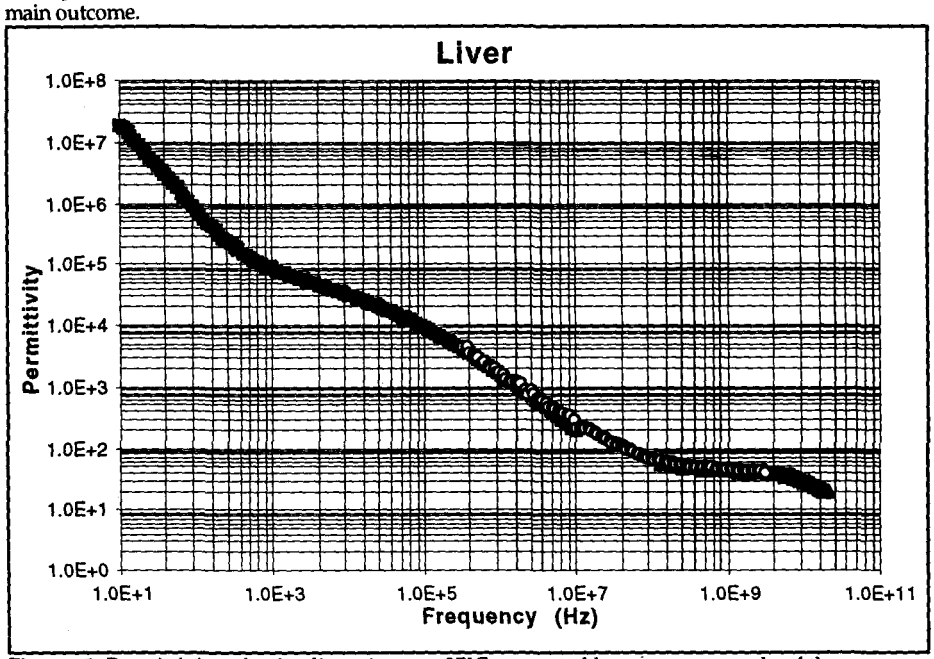
Interactions between em fields and people occur at all levels of organisation. The coupling of external fields with the body is the first step leading to further interactions at the cellular and molecular level. The initial coupling is a function of numerous parameters including field characteristics as well as the size and shape of the body and its electrical properties. The coupling is most efficient when the size of the body is of the same order of magnitude as the wavelength of the field and when the long axis of the body is in the direction of the field. This statement is based numerous dosimetric exercises in which the shape of the body was approximated to a geometric form such as a homogeneous ellipsoid to facilitate calculations and on later verifications using more realistic models (1). A consequence of the primary interaction is that internal fields are induced inside the body. The quantification of these fields and their spatial distribution is the concern of dosimetry.

The internal fields will, in turn, induce local fields at the level of the cells, in the extracellular space, within cells and across cell membranes (2). The quantification of these fields and the determination of local current pathways is termed microdosimetry (3). Microdosimetry can be extended to include interactions at the molecular level using concepts of em energy absorption and the establishment of internal thermal gradients (4).

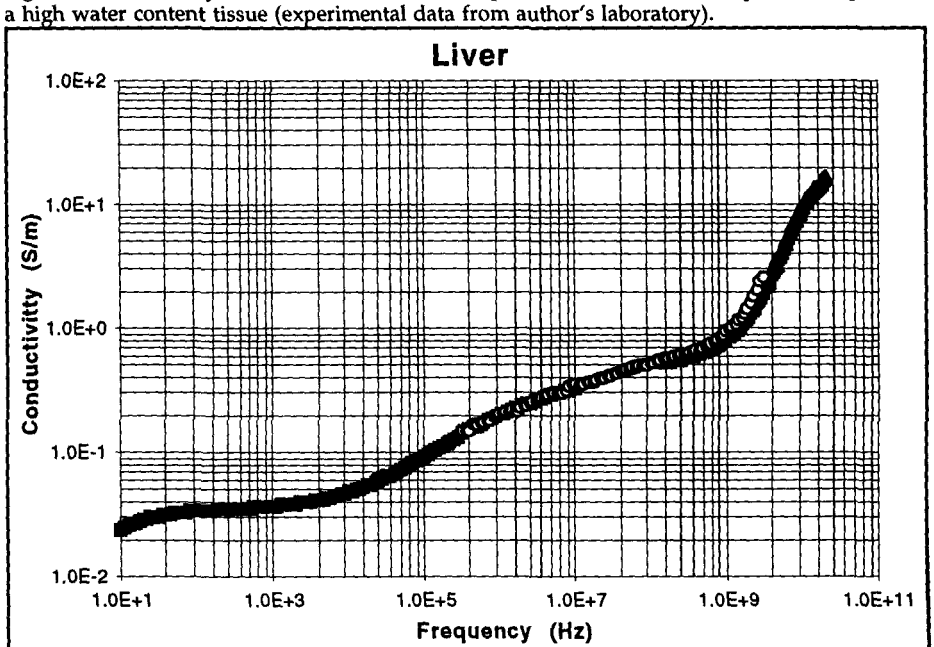
Electromagnetic fields interact with matter through forces generated on charges. Internal electric fields act on bound and free charges in the body tissue causing polarisation, molecular orientation and the establishment of ionic currents. There is little, if any, direct interaction with a magnetic field, instead, time varying magnetic field generate electric fields with the usual consequences.

The electrical properties of tissues (relative permittivity ε' and total conductivity σ) can be considered a measure of the interaction of the tissue with an electric field. For most tissues, the relative permittivity is highly frequency dependent from hertz to gigaherts with values reaching 10^7 below 100 Hz decreasing to less than 50 above 1 GHz (Fig. 1). The corresponding conductivity

values (Fig. 2) increase with frequency is steps that mirror the fall in permittivity. Implicit in this qualitative analysis is that strong direct interactions are likely at low frequencies (high permittivity and low conductivity) while at high frequencies, the interactions are dominated by the high conductivity of tissues making energy absorption from ionic and polarisation currents the main outcome.



Figures 1: Permittivity of ovine liver tissue at 37°C presented here as an example of the spectrum of a high water content tissue (experimental data from author's laboratory).



Figures 2: Conductivity of ovine liver tissue at 37°C presented here as an example of the spectrum of a high water content tissue (experimental data from author's laboratory).

BIOLOGICAL EFFECTS

Cells and tissues exist in a background of bioelectric fields. For example, some electrically active cells sustain a transmembrane potential of up to 0.1 V (inside negative), cell communications initiate action potentials which are pulse like signals lasting a few milliseconds. Currents induced by external fields add to and interfere with these ambient fields. At frequencies below 1 kHz, induced currents flow mainly through the extracellular fluid, they affect the electrical environment of cells, may cause changes in the transmembrane potential, and, if sufficiently intense, stimulate electrically excitable cells. Current densities of the order of 0.1 Am^{-2} are capable of stimulating nerve and muscle cells (5) while higher currents have more serious consequences, this compares with endogenous current densities of between 1 and 10 mAm^{-2}. Interactions at or below the threshold for stimulation are not isothermal, energy is absorbed but the resulting thermal load is negligible by comparison to the thermal fluctuation of the body. The threshold for stimulation increases proportionally with frequency, the energy dissipated by that currents increases at a faster rate. At about 1 MHz, thermal damage to the cells may occur at current densities below the stimulation threshold. Interactions resulting in thermal effects are described in terms of the power absorbed per unit body mass or specific absorption rate (SAR). People are accustomed to receiving thermal stimulation and, provided that these are not too large, the body can deal with them by invoking thermoregulatory responses. The threshold SAR for the onset of thermally induced biological effects is about 4 Wkg^{-1}. This level of SAR may give rise to a temperature elevation of about 1 or 2 degrees and may cause behavioural changes or result in a reduction of performance of learned tasks in experimental animals. These effects are consistent with the rise in temperature. The biological effects associated with higher levels of SAR are well documented and have been extensively reviewed (6, 7) they include modification of the action of drugs, changes in the secretion of hormones, developmental abnormalities as well as transient effects on heat sensitive systems such as spern cells and blood forming tissues. The batabase of biological effects is consistent with a strong correlation between the SAR and the severity of the resulting biological effect.

There is a growing body of evidence describing subtle biological responses to specific low intensity fields below the threshold for thermal manifestations. . A common feature to most of the experimental studies is the lack of dose-response relationship. The concept of low level interactions leading to significant biological effects has been challenged on theoretical grounds (8). The main argument is that such fields are likely to be masked by thermally generated electrical noise. Several non linear interaction mechanisms have been proposed to describe some of the experimental results in terms of signal amplification from resonant or cooperative interactions at the site of the cellular membrane. The absence of dose response relationship together with the lack of well defined mechanism makes it difficult to plan new experiments and even to repeat old ones in different laboratories under identical exposure conditions.

Two examples of well conducted studies will be briefly reported to illustrate the type of research that needs to be replicated in different laboratories. Degenerative changes caused by low level microwave irradiation in the retina, iris and corneal endothelium of primates were first reported in 1985 (9) followed by several studies by the same research group over a number of years (10). The effects were observed with continuous irradiation but pulsed microwaves were found to be as effective at lower power levels. Pre-treatment of the eye with the glaucoma drug timolol maleate further lowered the threshold for damage to an average SAR of 0.26 Wkg^{-1}. Although the authors did not measure intra ocular temperatures in the animals, the results suggest that a mechanism other than significant heating of the eye is involved. Another example of biological effects arising from acute low-level exposure to microwaves involves single-strand breaks in DNA in brain cells from rats (11). No significant effect is observed immediately after irradiation with pulsed microwaves but a dose related effect was observed 4 hours post exposure equivalent to 0.6 and 1.2 Wkg^{-1} whole body SAR. With continuous irradiation increase in DNA single strand breaks were observed immediately as well as 4 hours post exposure. The study suggests that microwave irradiation may increase the rate of DNA breaking or inhibit the repair processes in the cells.

A study by Chou et al (12) illustrates investigations of the effect of chronic exposure to low levels of microwave . The aim of the study was to investigate the effects of long-term exposure to pulsed

microwave radiation. An interesting and important feature of the study was the exposure of a large sample of experimental animals (rats) throughout their lifetimes in order to monitor them for effects on general health and longevity. Statistically the results were negative overall for effects on general health, longevity, cause of death, and lesions associated with ageing and the incidence of benign tumours. Some positive results on hormone levels and changes in the immune system were transient. A statistically significant increase of primary malignancies in exposed rats compared to controls was reported but tumour incidence was lower than historically expected in both groups. The authors state that, in the light of other parameters in the study, it is conjectural whether the excess reflects a true biological influence. Moreover, in the absence of reduced longevity, the biological significance of this effect is questionable. Overall, the results indicate that there were no definitive biological effects in rats chronically exposed to microwaves. The cost of repeating lifetime animal studies is not trivial in many respects, nevertheless, such studies need independent verification. A number of long term exposure studies are currently underway, it will be interesting to see if they have sufficient similarities in the experimental design to enable a meaningful comparison of the data output.

Understanding and eventually predicting the biophysical responses to low levels fields is a challenging task. Below 100 kHz, a prerequisite is the quantification of induced currents at the site of interaction. Such currents are different from those calculated in the primary interaction of external fields with the body assuming tissues to be uniform media of known dielectric properties. In principle it should be possible to extend the same dosimetric techniques to the cellular and molecular levels, in practice, it is very difficult to model with accuracy complex biological tissues including cells and their associated bioelectric fields. The theoretical considerations are not trivial even for random non biological systems (13). The answer may well rest with the development of experimental microdosimetric field measurement techniques (2,14) or indeed with a dual approach of experimental and computational methods.

The estimation of induced current density is important in the frequency range where direct action of the field on cells is anticipated. A different approach to microdosimetry should be adopted at higher frequencies where energy absorption is the prime consideration. For example, dielectric studies can yield relevant information on the polarisation of cells and molecules and help predict the distribution of energy absorptions at the molecular level (4,15). Experimental techniques include the use of molecular probes to provide spatial and temporal estimation of temperature and specific energy absorption at the cellular and subcellular levels (16). Progress rests with a multidiciplinary approach including experimental and theoretical investigations.

IMPLICATIONS FOR STANDARDS

Standards should, ideally, be based on rigorous scientific evidence that a physical agent is capable of causing harm under identifiable conditions. Standards, by their very nature, introduce controls and are bound to have an economic impact. However, when the science is clear cut, all other considerations are forsaken in favour of scientific data of health hazards and their corresponding thresholds. The scientific base underpinning em exposure standards is well established with respect to acute effects, but the issue is clouded by uncertainties provided by the growing database of low levels effects. It is important that such studies be continued and that their health implications, if any, be determined before their incorporation into standards. Equally important is to resist using the prevailing uncertainties as an excuse to adopt policies of 'prudent avoidance' and the like. Prudent avoidance measures may be, and indeed mostly are, harmless in their own right, their incorporation into guidance documents is however tantamount to the rejection of science as basis for protection.

CONCLUSIONS

The importance of microdosimetry has been highlighted to help elucidate the mechanisms of interaction responsible for low level effects. At low frequencies, the emphasis is on estimating the current at the cellular, at microwave frequencies energy absorption should be understood at the

molecular level. The argument points to the importance and the relevance of extending dosimetric studies to include thermal modelling for both human and animal exposures. Because the biological effects are related to the temperature rise, thermal modelling would enable better extrapolation to human from animal experiment and also help our understanding of the mechanisms involved in the interaction.

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BASIS OF CONCERN: THE POSSIBLE LUNG CANCER RISK FROM RESIDENTIAL RADON

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INTRODUCTION

Our concern about the lung cancer risk from indoor radon arises from three issues. (1) The measured large variation range of the radon concentration in the indoor air of dwellings; (2) the rather high alpha dose to the critical bronchial cells from inhaled short-lived radon progeny; and (3) the clear evidence of an excess lung cancer risk, increasing with the exposure to radon progeny, which follows from epidemiological studies of Rn-exposed underground miners as well as from animal experiments. - On the basis of the latter findings, the International Agency for Research on Cancer has classified radon with its progeny as a carcinogenic agent to humans.

To quantify the possible lung cancer risk from indoor radon three different types of approaches can be taken into consideration: (1) Direct, well-designed epidemiological studies in population groups exposed residentially; (2) the transfer of exposure-risk models which follow from Rn-exposed miner's data, to the residential radon exposure of the general population; and finally (3) the so-called "dosimetrie approach" which proceeds from the observed excess risk of lung cancer among the atomic bomb survivors (Life Span Study), following a single, short-term dose to the lung by external radiation.

Starting with a summary of the population exposure to residential radon and its progeny, in the following the results and uncertainties of these risk approaches are outlined.

EXPOSURE TO RESIDENTIAL RADON

Survey measurements of the radon concentration in dwellings have been carried out in many countries. The results which are summarized in the UNSCEAR 1993 Report, yield a variation range from a few up to 200 000 Bq m^{-3}. The observed variation can be approximated by a log-normal distribution function with geometric standard variation in the range of 2-3. The cumulative distribution of the national mean values is shown in figure 1. The national arithmetic mean values cover a range from 10 to 110 Bq m^{-3}, compared with a mean value in outdoor air of about 10 Bq m^{-3}.

On the basis of these data UNSCEAR adopts for the world population a population - weighted, arithmetic mean value of the radon concentration in dwellings of 40 Bq m^{-3} (geometric mean 26 Bq m^{-3}). Taking into account a mean equilibrium factor F=0.4 for Rn progenies in indoor air this corresponds to a mean equilibrium - equivalent concentration c_{eq} of 16 Bq m^{-3}. Assuming a mean occupancy factor indoors of 0.8, it follows for the world

population a mean annual exposure indoors (exposure=concentration x residence time) to radon progeny of about $1.13 \cdot 10^5$ Bq h m^{-3}, expressed in terms of the equilibrium-equivalent exposure, or $6.28 \cdot 10^{-4}$ J h m^{-3}=0,177 WLM in terms of potential alpha energy exposure P_p. With a life expectancy of 65-85 years follows a lifetime exposure of 12-15 WLM.

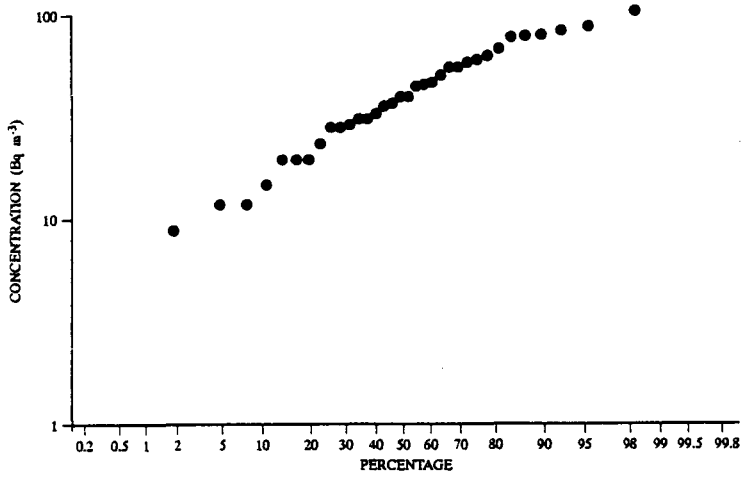

Fig.1. Population-weighted cumulative distribution of the mean radon concentration indoors which results from national surveys in 34 countries (UNSCEAR 1993).

From the observed variation of indoor radon it can be estimated that worldwide in about 2 % of dwellings the annual exposure might be more than a factor 5, and in about 0,2 % more than a factor 20 higher than the worldwide mean annual exposure value. - In this context it seems important to note that the epidemiological studies of Rn-exposed miners yield a statistical significant excess of lung cancer at cumulative occupational exposures above 50 WLM.

EPIDEMIOLOGICAL APPROACH FROM MINER'S DATA

Epidemiological studies of Rn-exposed miners in the USA, Canada, Europe, China and Australia give strong evidence for an the induction of lung cancer by inhaled radon progeny. Recently a joint analysis of the data from 11 cohort studies of male underground miners has been conducted (Lubin et al. 1994a). Figure 2 shows the excess relative risk (ERR) of lung cancer as function of the cumulative exposure P_p (in WLM units) to radon progeny which follows from the combined data of these studies. The analysis yields as best fit a linear relationship up to exposure levels of about 400 WLM with a slope ERR/P_p=0,005 WLM^{-1} (95 % confidence interval 0,002 - 0,010 WLM^{-1}). This relative risk coefficient is modified, however, by various factors. It decreases with attained age and declined with increasing time since exposure. Furthermore the data indicate an increase of the relative risk coefficient with decreasing exposure rate or with long duration of exposure (see figure 3). This inverse exposure-rate effect has been sometimes interpreted in terms of a reciprocal correlation. From an extended analysis of the data it has been recently concluded, however, that this effect decreases with decreasing cumulative exposure and obviously disappears at exposure levels below about 50 WLM which are of relevance for the population exposure (Lubin et al. 1995a).

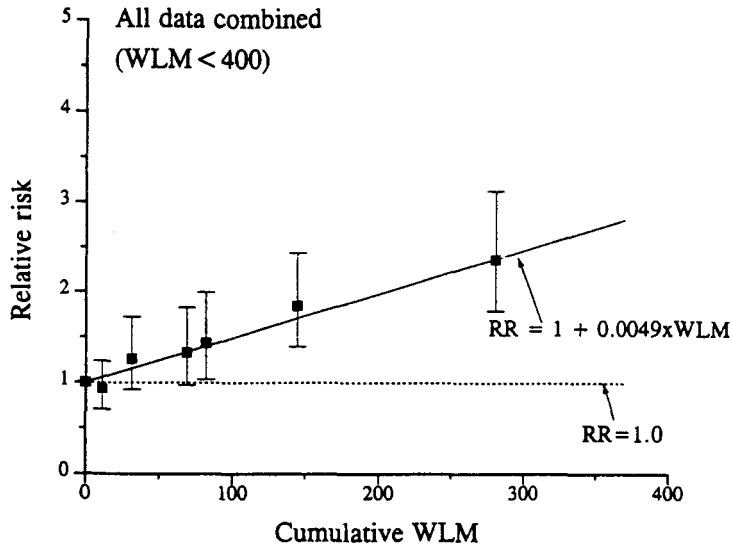

Fig.2. Relative lung cancer risk (RR) as function of the cumulative exposure (in WLM) to radon progeny which follows from the combined data of 11 cohort studies of underground miners. The mean values and their 95 % confidence interval are plotted at the mean exposure in each exposure category. The closed line indicates the best linear fit to the data (from Lubin et al. 1994a).

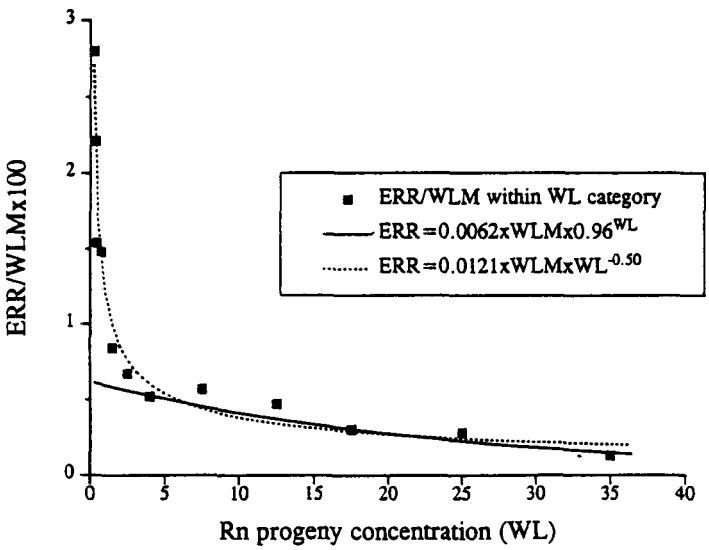

Fig.3. Estimated mean values of the relative risk coefficient ERR/P_p (WLM) within Working-Level (WL) categories as it follows from the joint analysis of 11 cohort studies of underground miners. The curves give the results of different models (from Lubin et al. 1994a).

A further uncertainty concerns the influence of tobacco smoking. The available few data from miners give little quantitative information on this subject. In previous studies (ICRP 1987, BEIR IV-NRC 1988) a multiplicative model was assumed. This means that the relative risk coefficient ERR/P_p, would be equal for nonsmokers and smokers. The now available data from some groups of Rn-exposed miners indicate that the interaction between smoking and radon is somewhat less than multiplicative but more than additive. Lubin et al. (1994a) assumed in their smoking-adjusted model an ERR/P_p-value which is a factor 0,9 lower for smokers and a factor 3,0 higher for nonsmokers, compared with the average value derived from the miner's data. The uncertainty of this adjustment factor for nonsmokers is, however, quite large. A reanalyses of the data of the Colorado uranium miners yields an ERR/P_p-ratio between nonsmokers and smokers (20 cigarettes/day) of somewhat less than 2 (Moolgavkar et al. 1993).

In table 1 the central lifetime risk estimates of the attributable relative lung cancer risk (ARR) from residential radon exposure, averaged over males and females, are listed which should be expected on the basis of the approach from miner's data. In addition to the model of Lubin et al. (1994a) the results of previous estimates (ICRP 1987, BEIR IV-NRC 1988) are given. The ARR-values refer to a lifetime exposure to radon progeny at a constant rate of 0,18 WLM/year as it follows for a mean Rn-concentration indoors of 40 Bq m^{-3} like it was assumed by UNSCEAR.

Table 1. Attributable relative risk (ARR) of lung cancer from residential exposure to radon progeny at a constant mean level of 0.18 WLM per year; central estimates of different risk models derived from miner's data with dosimetric adjustment for indoor exposure.[a]

Relative risk model	Smoking influence	Population	ARR, in percent	
			nonsmokers	smokers
ICRP 50 (1987) const.RR-model	multi-plicative	Reference population[b]	6.8	6.4
BEIR IV (NRC 1988) TSE model	multi-plicative	USA	7.0	6.5
		Germany	6.0	5.7
Lubin et al. (1994a) TSE/AGE/WL-cat model	multi-plicative	USA	12.0	11.3
		Germany[c]	7.3	7.0
	submulti-plicative	USA	28.5	10.3
		Germany[c]	18.8	6.5

a) Dosimetric adjustment factor K for indoor exposure:
0.8 for age < 10 y; 0.7 for age ≥10 y (NRC 1991);
b) refering to a population with a life expectancy of 70 y for males and 75 y for females.
c) estimate for the population in West-Germany, taking into account mortality data 1985-89 Steindorf et al., 1995).

The ARR-values given in table 1 are averaged over females and males. Assuming a multiplicative influence of smoking the attributable risk which follows from the new joint analysis of 11 miner's studies (Lubin et al. 1994), is about a factor 1.7 (US population) or 1.2 (German population), respectively, higher than the values resulting from the BEIR IV-model (NRC 1988). As it should be expected the values, adjusted for a submultiplicative influence of smokers, lead to considerably higher ARR-values for nonsmokers. It must be kept in mind, however, that this risk approach from miner's data involves large uncertainties. The statistical 95 % confidence interval of the input data covers a range given by a factor from about 0.3 to 2.5. Additional systematic uncertainties concern mainly the assessment of miner's exposure, the exposure-rate effect, the influence of smoking, and, last not least, the simplifying assumptions made for the transfer of the risk models from underground miners to the residential exposure of the general population, particularly to females and children.

DIRECT RESIDENTIAL RADON STUDIES

Direct well-designed, epidemiological studies on the carcinogenie effect of radon progeny in indoor air are desirable. Two different types of studies have been conducted: (1) So-called "geographie" or "ecological" studies which look for a correlation between the lung cancer rate in certain geographical areas and the mean radon exposure in these areas; and (2) case-control or cohort studies in which individual radon exposures of lung cancer cases are estimated and compared with those of appropriate controls.

The available results of geographic studies are summarised and reviewed in reports of Stidley and Samet (1993) and of UNSCEAR (1994). A further comprehensive study in which data from 1601 US counties are compared, has been published recently (Cohen 1995). - Summarising, 9 of these studies indicate a positive correlation, 6 studies show no correlation and 5 studies yield a negative association between lung cancer frequency and estimated radon exposure. Taking into regard the general methodological problems and limitations of these studies, the value of such geographical studies is questionable (Stidley et al. 1993, IARC 1995). An exception might be studies which compare the lung cancer frequency in areas with strongly different concentrations of indoor radon, like the study in Umhausen/Tirol in Austria (Ennemoser et al. 1994).

Case-control studies are more appropriate than geographical studies. The results of 12 published case-control studies are summarized and reviewed in the UNSCEAR 1994 Report and in papers of Lubin et al. (1994b; 1996 in press). Most of these studies involve, however, only a relative small number of lung cancer cases, which leads to large statistical uncertainties. This is demonstrated in figure 4 (from Lubin et al. 1994b) where for 7 of these studies the observed relative lung cancer risk is plotted as function of the radon concentration indoors. The three larger studies are the nationwide Swedish study (Pershagen et al. 1994), the study in Winnipeg/Canada (Letournean et al. 1994) and the study in New Jersey/USA (Schoenberg et al. 1990). The Swedish study (1360 cases, 2847 controls) yields for the concentration cohort from 140-400 Bq m^{-3} a mean excess relative risk of 0.3 (95 % CI: 0.1-0.6), compared with the cohort below 140 Bq m^{-3}. The study in Winnipeg (750 cases, 750 controls) shows no significant correlation between lung cancer and indoor radon exposure. The New Jersey-Study (433 cases, 402 controls) indicates a poitive trend after adjusting for smoking and age, but there are only a few cases with higher radon exposure in this study.

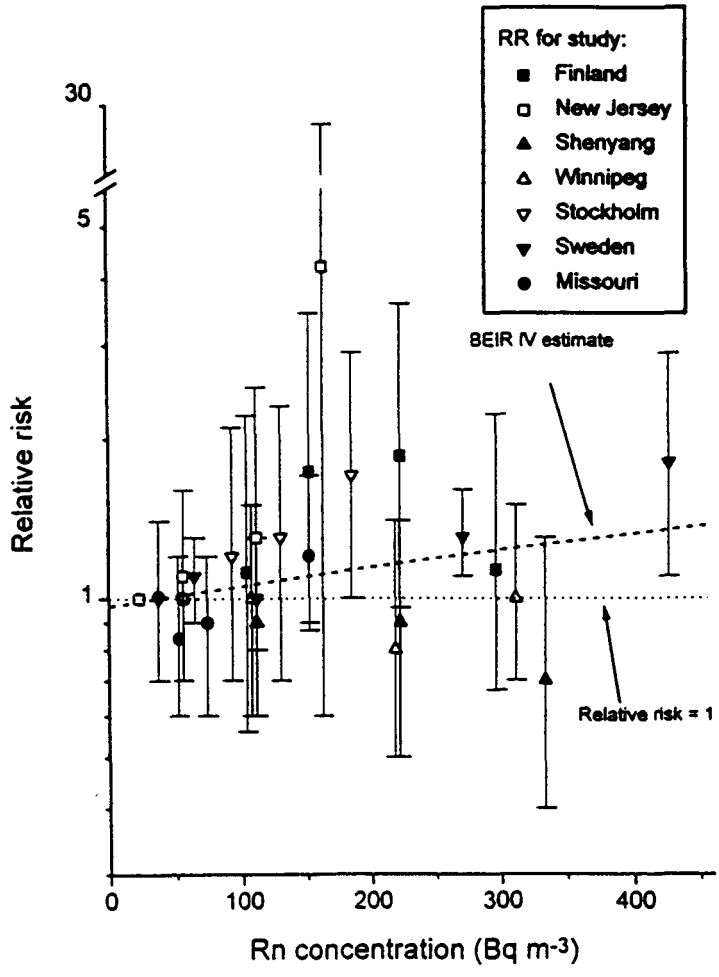

Fig.4. Relative lung cancer risk in different intervals of indoor radon concentration (mean values and 95 % conficence intervals) resulting from 7 residential case-control studies (from Lubin 1994b, 1996). The dashed line shows the increase which should be expected on the basis of the BEIR IV-model derived from miner's data, refering to a residential exposure from age 35-65.

However, all of the available case-control studies have limitations that hinder a clear interpretation of their results. This concerns particularly the assessment of the individual lifetime exposure, taking into regard the residential mobility, and the assessment of the individual smoking habits. It should be kept in mind that in these studies the radon exposure of most lung cancer cases was relatively low and therefore the attributable risk from radon will be small compared with the lung cancer risk from smoking. To illustrate the influence of errors in exposure assessment, Lubin et al. (1995b, 1996 in press) have simulated case-control studies. The results reveal the substantial contribution of these errors in explaining the inconsisteney of the present case-control studies. Lubin concludes: "These simulations imply that it is unlikely that case-control studies alone will be able to determine precise estimates of risk from indoor radon. Also implied is that even future efforts at pooling epidemiological studies may not adequately adress issues of risk from residential radon exposure" (Lubin 1995b).

FINAL CONCLUSIONS

Without doubt ionizing radiation is a carcinogenic agent for humans. This general statement is also valid for the induction of lung cancer from inhaled radon with its progeny. The problem is to quantify the risk from residential radon exposure. The available direct geographical and case-control studies are inconsistent and enable no conclusion. Like in the past risk estimates for residential radon must therefore continue to be based on the approach from epidemiological data of Rn-exposed miners. In spite all uncertainties this approach indicates that, apart from smoking, inhalation of radon with its progeny in dwellings seems to be the most important cause of lung cancer in the general population.

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1996
International Congress
on Radiation Protection

PRIORITY AND PROGRESS

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INTRODUCTION

Since 1980, Sweden has a system for limiting radon in dwellings. Before that, the Swedish Radiation Protection Institute (SSI) had informed about the hazard for about ten years (1,2) without any attention from the mass media. This information was based partly on the results of the measurements carried out by Hultqvist in 1955 (3) and my measurements in the early 1970s (4), which both showed rather high radon concentrations. The risk estimation was based on the experiences from miners especially in non-uranium mines.

Before 1978

Hultqvist measured gamma radiation in 986 dwellings in central Sweden built before 1946. Of these dwellings 287 were studied with regard to radon using ionisation chambers. The resulting radon levels are shown in Table 1. He found high levels in some of the alum shale concrete dwellings but also in a few dwellings built of other materials with very poor air exchange rates and in basements. Limits were discussed but not introduced because a good air exchange rate was judged to be necessary for other hygienic reasons and the ventilation rate was important for the concentration of radon. Another reason was that natural radiation was excluded in the recommendations from ICRP (the International Commission on Radiological Protection) and also in the Swedish radiation protection Act.

Table 1. The radon concentrations in air in four towns in central Sweden in the early 1950s, in houses built before 1946 (3). Both single-family houses and apartments in multi-family houses are included.

	Radon concentration[a], Bq m^{-3}			
	Unaired houses[b]	Aired houses[c]	Weighted average	Highest value
All dwellings	77	36	62	592 (1740[d])
Wood	15.2	15.5	15.2	63
Brick	47	26	40	303
Alum shale concrete	133	67	116	592 (1740[d])

a) 1 Bq m^{-3} = 0.027 pCi/liter
b) No thorough airing was carried out since the day before the sampling of air for measurement.
c) Dwellings were aired the morning before the sampling of air for measurement.
d) This value was not included in the original sample of dwellings. The ventilation rate was extremely poor.

In the 1970s we tried to interest the Swedish national boards responsible for indoor health in the radon problem, but with little success. We saw the radon concentration increase when residents saved energy by decreasing the ventilation rates. At that time we could not use the

idiation protection Act to specify limits or recommendations for natural radiation. We issued booklet for the public and presented it through a press release. One journalist, whom we ontacted said that it was not ethical to inform the Swedish people about such a problem efore Christmas. We also tried to bring up the radon problem internationally but the answer vas "Maybe you in the Nordic countries have a problem with your cold climate, alum shale concrete, and with inhabitants who never open windows, but we don't have that problem." Bo Lindell said at that time: "if you measure houses in your country you will find some houses with high radon levels. Even in a tent you will find levels higher than outdoors". We informed the company which produced the alum shale concrete that the material would not be permitted when limits for radium and thorium concentrations in building materials would be introduced. Subsequently, in 1974, they stopped the production. The material had then been produced since 1929.

1978-1989

However, after we had found several thousands of Bq m^{-3} in houses in 1978 (Table 2), the government granted Bo Lindell's request for a government commission to work with a limitation system for radiation exposure in dwellings. The Commission received a fund for

Table 2. The average concentrations are given for six houses built on abandoned waste from alum shale mining in past centuries in Sweden and for five reference houses. The average for each house of concentrations of radon and radon progeny together with air exchange rates and gamma radiation are given. The measurements were made in October 1978 (5).

Ground	Radon Bq m^{-3}	Radon progeny Bq m^{-3} EER	Air exchange rate h^{-1}	Gamma radiation μSv h^{-1}
6 houses on alum shale waste	400 - 1600	260 - 1030	0.33 - 0.47	0.0.06 - 0.09
5 reference houses	37 - 93	25 - 52	0.14 - 0.35	0.03 - 0.05

Table 3. Radon concentrations in single-family houses and apartments in multi-family houses in the 1976 Swedish housing stock (6). The measurements were made 1980-82.

Type of dwelling	Arithmetic mean, Bq m^{-3}	Geometric mean, Bq m^{-3}	>800 Bq m^{-3}	Highest value, Bq m^{-3} Average in a dwelling
Single-family	122	69	2 %	3310
Multi-family	85	53	1 %	920

targeted research. As one result, a random sampling was carried out in 1980-82 in the 1976 housing stock (6), Table 3, mitigation methods were investigated, a calibration room for testing radon detectors used for searching radon houses was built at our institute (7). This research also implied that the knowledge about radon, earlier concentrated to our institute, was undergoing a rapid growth in other organisations. Much effort was put into informing

health inspectors, builders, residents etc. The recommendations of the Swedish Radon Commission were used by the responsible national boards to set requirements on less than 400 Bq m^{-3} of EER[a)] for existing dwellings and 70 Bq m^{-3} EER for newly built dwellings. It was also recommended that the level should be decreased in the range 70 - 400 Bq m^{-3} EER when this could be done with simple methods.

The Swedish Radon Commission also issued recommendations on the distribution of responsibility between the authorities. The National Board of Health and Welfare is responsible for existing buildings, the National Board of Housing and Planning is responsible for future building and SSI is responsible for correct measurements and for giving advice to the other national boards. The local health and building boards are responsible for health in their region, which means that they are responsible for finding buildings with high levels and for radon risk mapping.

1990 - 1995

The limit (action level) for existing buildings was decreased to 200 Bq m^{-3} EER in 1990 and workplaces were included with the exception of mines etc.. In 1993, the limit was changed to 400 Bq m^{-3} of radon gas as a consequence of an interim report to the government (8) together with a proposal for more efforts to decrease the indoor levels. When the radon concentration is higher than the limit, the owners of single-family houses can receive a grant to cover 50% of the cost up to SEK 15 000 (about $ 2000) for measures to decrease the level. It is recommended to decrease the levels when they are higher than 200 Bq m^{-3} of radon gas. That level is now also the limit for newly-built buildings. The 1990 random sample study of the 1988 housing stock (9) found that about 4% of the dwellings had radon concentrations above the limit for existing buildings, 400 Bq m^{-3}, and about 14% had levels above 200 Bq m^{-3}, the limit for newly-built buildings.

MEASUREMENT PROTOCOLS AND CALIBRATIONS

The first measurement protocols were issued in 1980. New protocols were established in 1988 and another version was published in 1994. We now have two separate protocols: one for legal purposes and one recommendation for short-term advisory measurements e.g. in connection with real-estate transactions. The measurement companies can become accredited on a voluntary basis. We are discussing the certification of radon measurements for consultants and how to put more pressure on the nuclear track film laboratories to become accredited. The protocols require one annual calibration at our institute or another about which we provide information.

We have a radon and a radon progeny standard which is the basis for our radon calibration rooms with a known concentration of radon and radon progeny. The results compare well in international comparisons.

BUILDINGS INVESTIGATED

Measurements are ordered from measuring companies by the local health and building authorities and by private persons. Therefore, we have no statistics of the number of dwellings and workplaces measured unless we ask the local health and building authorities. The last time the municipalities were asked was for the period up to 1992, when 151,600 dwellings and workplaces had been measured. Together with measurements by others, about 240,000 cases had been measured. A further 50,000-100,000 dwellings and workplaces have been measured between 1992 and June 1995 (10).

DWELLINGS MITIGATED

Approximately half of the dwellings, about 15,000, which up to 1992 were found to have levels higher than 400 Bq m^{-3}, had been mitigated at that time. A further 10,000 dwellings are estimated to have been mitigated since then (10). The mitigation methods are described in a handbook (11). The mitigation methods mostly work very well, in a few cases various attempts have to be made to decrease the concentration. New mitigation methods have been developed to decrease the concentration in houses when the building material is the major radon source. One is a wallpaper containing charcoal (12). In a research study it gave very good results, but when used on full scale to mitigate whole homes the results have been less encouraging.

It has been found that the low radon concentrations after different types of mitigation are not always sustained according to the first part of a follow up study (13). A few years after the mitigation the radon concentration had risen by more than 30% in about 25% of the cases. Fifteen dwellings showed at least double as high radon concentration as immediately after the mitigation. This change was not specific to any given method. The second part of the study three years after the first part gave the same resulting pattern, but not for the same dwellings.

The limit (action level) for existing buildings is in principle compulsory. In a building with more than 400 Bq m^{-3} the radon exposure is deemed insanitary. For buildings with rented dwellings, the practice is that the local public health boards can impose penalty if nothing is done in buildings with levels exceeding the limit. For detached houses where the owner lives with his/her family the practice has been only to inform about the risks when the level is above the limit. However, one municipality required measures when children lived in such houses and one house owner has appealed against the decision. The case has been tried in court several times and has not yet been finally decided. Hitherto, the local health board has been found to be right.

NEW BUILDINGS

A decrease in the average radon concentration in dwellings built since 1981 to half of the average before that time can be seen from the random sample study of Swedish dwellings, see Fig. 1. This is partly because the radium rich alum shale-based concrete has not been produced since 1975, but half of the decrease depends on the improvement of ground constructions to decrease the inflow of radon in soil as a result of the 1980 Swedish Building Code together with information and training. The limits given in the Building Code are compulsory but are

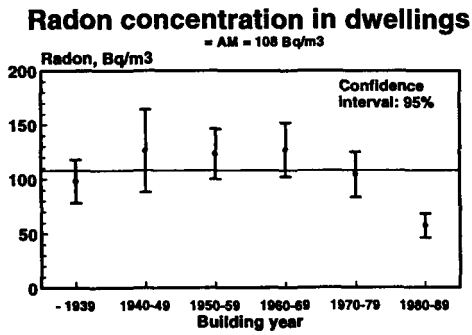

Figure 1. The radon concentrations in the 1988 housing stock (9). The measurements were made in 1990 in a random sampling of Swedish dwellings in single-family and multi-family houses.

applied in different ways. One way of interpretation is that when buildings are finished they are existing houses. Another is that all buildings built after the limitation system came into force should have concentrations below the limit.

WORKPLACES

Systematic investigations of workplaces (other than mines, etc.) have been started by the National Board of Occupational Safety and Health. Earlier, schools and nurseries were investigated by the local public health boards and, when needed, usually mitigated. Few other workplaces above ground level have been mitigated. The limit for workplaces is the same as for dwellings, 400 Bq m^{-3} radon gas. For miners and other underground workers the limit for radon progeny concentrations is 2 MBq h y^{-1} (equivalent to 2 200 Bq m^{-3} of radon gas for 2000 working hours per year and F=0.5[b]). Revision of the limit is being pursued.

WATER

Over the past few years there has been great interest in radon in drinking water. Already in the beginning of the 1960s von Döbeln and Lindell (14) studied the retention in the body for radon following ingestion. On this basis, they calculated the absorbed dose to various organs in the body and found about the same values for adults as were later found based on more experimental work. They also measured the radon concentration in tap water in some towns etc. in central Sweden. The maximum value found was 1200 Bq/l. Most levels were less than 100 Bq/l.

About 10 000 private wells are estimated to have radon concentrations exceeding 1000 Bq/l, based on a study in the 1980s of random sampling of private wells and public water plants (15). Also some public water plants have high concentrations. It has been recommended since 1984 that reduction of radon levels should be considered when the concentrations are between 100 and 1000 Bq/l. It is recommended that radon levels exceeding 1000 Bq/l should be decreased. The National Food Administration has proposed limits for radon in drinking water in 1995. More than 100 Bq l^{-1} would be a compulsory action level for public water plants. 500 Bq l^{-1} would be a voluntarily action level for private wells. Water with more than 1000 Bq l^{-1} would not be suitable for drinking. The interest in decreasing the radon concentration in drinking water seems to be much higher than that for decreasing the radon concentration in the air of the dwellings.

Over the past few years several companies have developed different types of radon removal equipment for private wells. Some of the companies guarantee that the radon concentration in the water after installation will not be higher than 100 Bq l^{-1}. Investigations of conventional water cleaners have shown that the influence on the radon concentration is small (16,17). If limits are established, measurement protocols will be given and accreditation of the measuring companies will be required.

GENERAL EXPERIENCES

The scope of the information problem is illustrated by the fact that around 1000 official persons in 284 municipalities, around 100 000 building professionals, and a large portion of the more than 8 million population, had to be informed about finding on this issue.

One Swedish report deals with the growth, dissemination and implementation of knowledge regarding radon in buildings (18). The project was initiated by the Scientific Advisory Board of Swedish Council for Building Research because the work with radon was

seen as a success compared with other problems. The experiences of the radon work was generalised to general health problems obtained from buildings.

Risk perception and communication about radon have been studied by Lennart Sjöberg at the Center for Risk Research, Stockholm (19). He found among other things that the health risk seemed to be underestimated and that the possibility of decreasing the risk by one's own handling was also underestimated.

TRAINING AND INFORMATION

What are we doing in Sweden on basis of all this experience? Radon issues are now dealt with in most upper secondary schools and at universities. Further training of employed professionals has taken place via external courses of 1-3 days. A radon campaign started in the autumn of 1995. Our booklets have been rewritten, articles for the daily press are written, and the local health and building authorities have received updated information. The main message has focused on the health hazard, on where to order a test, and on mitigation. The demand for training courses has increased and our institute gives courses in measurements (one basic course, and one for accreditation), a course for radon risk mapping, one for mitigation, and one for testing and mitigation of drinking water. Seminars in different parts of the country are planned. A training course on radon given by the European Radiation Protection Education and Training (ERPET) and SSI was held in Stockholm 11 - 15 September 1995.

During the last years our institute has given the high schools the possibility to order nuclear track films, which the pupils themselves place in yoghurt mugs with a sheet of plastic. This is based on an idea from Bristol University. We etch the films chemically after exposure and give a calibration factor, so that the pupils can count the marks on the film and calculate the radon concentration.

CONCLUSIONS

The interim report to the government in 1993 and the suggestions for future work stimulated an interest in the mass media, an interest which has remained. This interest has increased the number of measurements and also the number of mitigated buildings. Most Swedes know something about radon indoors today. Our experience is that it takes a ten year period to disseminate knowledge to health inspectors, builders, inhabitants etc. However, after 15 years with a radon limitation system only about 10-20% of the expected number of dwellings with levels higher than the limit have been mitigated.

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[a] EER stands for the Equilibrium Equivalent concentration of Radon. EER is derived by multiplying the radon gas concentration by the equilibrium factor F, which in most countries is around 0.3-0.5.

[b] F stands for the equilibrium factor.

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applied in different ways. One way of interpretation is that when buildings are finished they are existing houses. Another is that all buildings built after the limitation system came into force should have concentrations below the limit.

WORKPLACES

Systematic investigations of workplaces (other than mines, etc.) have been started by the National Board of Occupational Safety and Health. Earlier, schools and nurseries were investigated by the local public health boards and, when needed, usually mitigated. Few other workplaces above ground level have been mitigated. The limit for workplaces is the same as for dwellings, 400 Bq m^{-3} radon gas. For miners and other underground workers the limit for radon progeny concentrations is 2 MBq h y^{-1} (equivalent to 2 200 Bq m^{-3} of radon gas for 2000 working hours per year and F=0.5[b)]). Revision of the limit is being pursued.

WATER

Over the past few years there has been great interest in radon in drinking water. Already in the beginning of the 1960s von Döbeln and Lindell (14) studied the retention in the body for radon following ingestion. On this basis, they calculated the absorbed dose to various organs in the body and found about the same values for adults as were later found based on more experimental work. They also measured the radon concentration in tap water in some towns etc. in central Sweden. The maximum value found was 1200 Bq/l. Most levels were less than 100 Bq/l.

About 10 000 private wells are estimated to have radon concentrations exceeding 1000 Bq/l, based on a study in the 1980s of random sampling of private wells and public water plants (15). Also some public water plants have high concentrations. It has been recommended since 1984 that reduction of radon levels should be considered when the concentrations are between 100 and 1000 Bq/l. It is recommended that radon levels exceeding 1000 Bq/l should be decreased. The National Food Administration has proposed limits for radon in drinking water in 1995. More than 100 Bq l^{-1} would be a compulsory action level for public water plants. 500 Bq l^{-1} would be a voluntarily action level for private wells. Water with more than 1000 Bq l^{-1} would not be suitable for drinking. The interest in decreasing the radon concentration in drinking water seems to be much higher than that for decreasing the radon concentration in the air of the dwellings.

Over the past few years several companies have developed different types of radon removal equipment for private wells. Some of the companies guarantee that the radon concentration in the water after installation will not be higher than 100 Bq l^{-1}. Investigations of conventional water cleaners have shown that the influence on the radon concentration is small (16,17). If limits are established, measurement protocols will be given and accreditation of the measuring companies will be required.

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IRPA 9
1996 International Congress on Radiation Protection
14-19 April 1996, Vienna

SYMPOSIUM «RADIATION PROTECTION AT NUCLEAR FACILITIES»
Wednesday, April 17, 1996 from 11.00 to 12.30 hours

INTRODUCTION

Laurent STRICKER
Senior Vice President, Safety and Technology Issues
Electricité de France, Paris

Operation and maintenance of nuclear installations imply the exposure of workers, and to a lesser extend of the public, to ionizing radiation. This exposure is by far the smallest contribution to the collective dose received by the population compared with other sources. However, because this contribution can be potentially important from the point of view of the radiation protection of individuals, since the beginning of the practice in the 1950s, it always deserved a particular attention by those in charge of its operation. It is of course beyond the capability of this short symposium to cover all aspects related to radiation protection at nuclear facilities and the session has been deliberately prepared having in mind occupational exposures.

In spite of the general increase in electricity generated, the most striking feature, as far as occupational radiological protection is concerned, is the ongoing improvement of performances over the last two decades. The 1993 UNSCEAR report [1] estimates that during the last 15 years the average annual collective dose has not substantially varied, but the reduction of the effective collective dose per unit electrical energy generated was about 50% and the estimated average individual dose by 30%. These remarkable performances can be observed in almost all types of installation of the nuclear fuel cycle. Figures 1 and 2 below illustrate the evolution of collective exposure in the case of nuclear reactors. Since the early eighties there is a continuous and significative reduction of the collective dose per unit even with the progressing number of reactors put into operation. This trend is even older and more important when using the collective dose per unit of electricity produced as performance indicator.

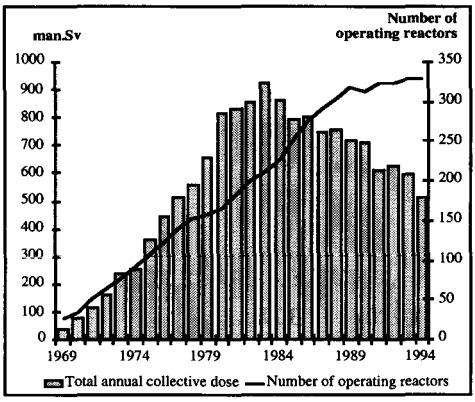

Figure 1 : Number of operating reactors and total collective dose per reactor in OECD countries [2].

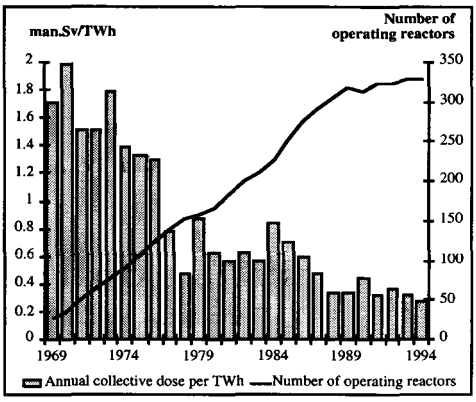

Figure 2 : Number of operating reactors and total collective dose per TWh in OECD countries [2]

As far as individual exposures are concerned, the picture is less clear. The global improvement of the situation over the last decades is not questionable, however the difficulty to follow accurately all transient workers operating in several installations, who are forming in certain countries a significative part of the work force in the nuclear industry, does not allow to present reliable statistics. Some utilities and countries have developed systems to overcome this difficulty and it is possible to present particular situations to illustrate the evolution with individual exposures. As an example Figures 3 and 4 present the performances in terms of annual individual doses for workers in French nuclear power plants since an operational dosimetry network connecting all plants (the so-called DOSINAT system [3]) has been put into operation.

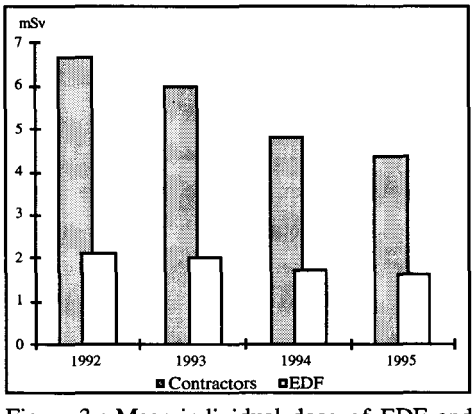

Figure 3 : Mean individual dose of EDF and contractor workers from 1992 to 1995

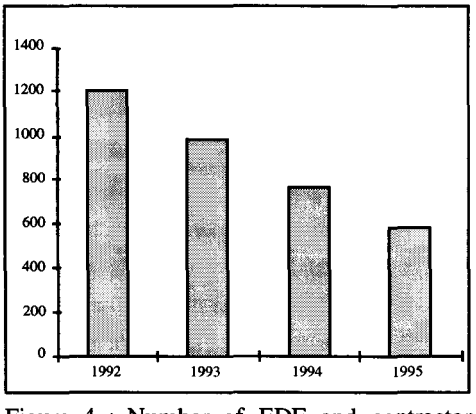

Figure 4 : Number of EDF and contractor workers with annual individual doses higher than 20 mSv

The general evolution of occupational exposure at nuclear installations is of course resulting from a combination of many different factors either of technical or organisational nature. As an introduction to this symposium, I would like to briefly delineate some of these factors that will be presented in a more analytical and detailed manner by the three invited speakers.

The emphasize put since the seventies on the ALARA principle is certainly the driving force to explain the general reduction of occupational exposures observed over the last two decades. The willingness to maintain or to reduce doses as low as reasonably achievable taking into account economical and social considerations has progressively transformed the radiation protection organisation and culture. The rationale in applying ALARA in nuclear installations is to achieve a level of exposure low enough to guarantee a sufficient protection of the workers but preserving at the same time the economical viability of the installations.

The role of regulation in this evolution was certainly important to initiate the process and maintain a high level of commitment of all parties involved. Different views exist in the various countries about how the ALARA principle can be practically implemented. In some countries regulators have clearly played a key role for the spreading of the ALARA approach, in some others, utilities have been the initiators. Whatever the national context, it is now recognized that a large flexibility should be left to operators to develop their ALARA programmes according the specificity of the technical, economical and social context. The role of regulation and regulators with regard to the practical implementation of ALARA will be developed by Mr Thommy Godås and Mr Lars Malmquist from Sweden.

The collective exposure in installations is dependent of ambient dose rates at working places, the duration of the work to be performed in these ambient dose rates and the number of workers involved. Any attempt to better control the doses that the personnel finally received involves to intervene either at the level of the sources or the management of the work. The control of sources has always received much attention in all types of installations and considerable efforts have been done to reduce the radiation field build-up. In light water reactors for example, different techniques have been developed to reduce or replace cobalt in structural materials, to control water chemistry, to decontaminate systems and or to install shieldings for particular components. In recent years, progress has been also made in the field of work management. ALARA programmes are giving more and more importance to non-technical factors that can influence the duration of work in radiation fields. Selection of maintenance options, planning and co-ordination of operations, training of workers, have proven to be efficient ways for reducing the number of manhours in radiological environment [4]. This last aspect will be presented in more details by Mr. David Miller from the United-States.

All together, the various actions developed year after year to either reduce sources or the duration of exposed work combined with the evolution of materials and techniques have resulted in most installations in a significative decrease of individual and collective exposures. The evolution of the collective dose associated with the replacement of steam generators at nuclear power plants, as shown on Figure 5, is certainly one of the most significative example to illustrate the global impact of all the above mentioned actions implemented to control exposures.

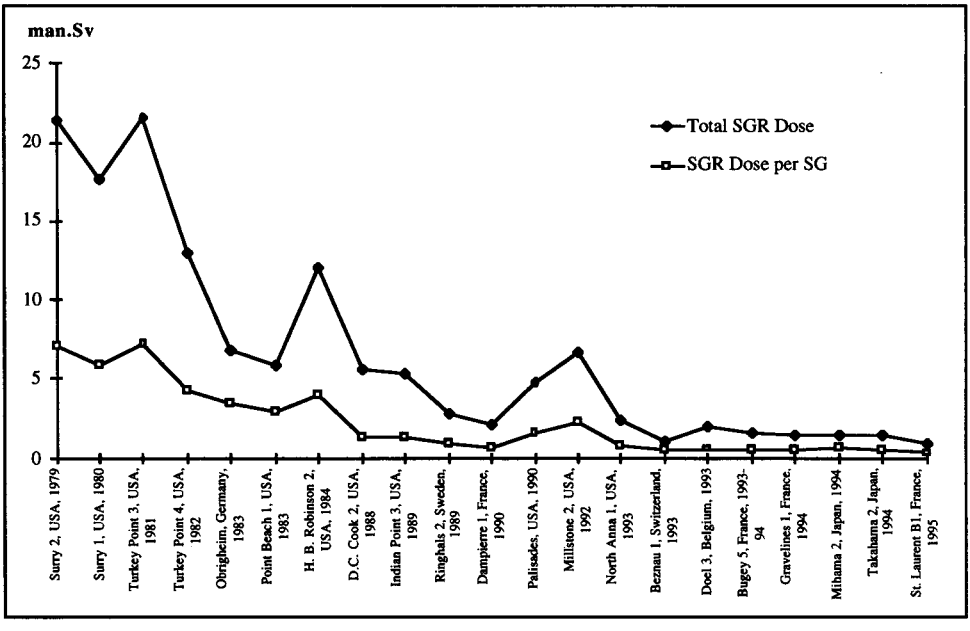

Figure 5 : Evolution of the total collective dose associated with the replacement of steam generators at nuclear power reactors from 1979 to 1995 [2]

The ALARA approach, by putting forward the a priori estimation of exposures, the follow-up of doses and the analysis of past experience with radiation protection, calls for the implementation of systematic and coherent dosimetric systems in order to monitor ambient dose rates and to track personnel doses at the operation and even task levels. Important improvements have been made over the last decade in the field of electronic dosimetry which now allow to obtain a detailed picture of where, when and how doses are received by the various categories of workers during the operation and maintenance phases of installations. It seems now indispensable to fully develop the operational dosimetry to complement the older generation of personal dosimetry which is not in position to provide the adequate and necessary data for the

preparation, follow-up and past experience analysis of operations that are needed in an ALARA perspective.

This last aspect is of prime importance. Large efforts have been deployed over the last recent years to improve at all levels (utilities, national or international organisations) the establishment of past-experience data bases. The International System on Occupational Exposure (ISOE) developed by the Nuclear Energy Agency from OECD is a good example of such effort [2]. Beyond the collection and analysis of the data of most operating reactors in the world, the merit of such a system, like some others, is to give the opportunity to professionals to regularly meet and exchange information and experiences.

Beyond the implementation of appropriate technical and organisational measures, the success of the ALARA approach in nuclear installations is closely related to the spreading of a «radiological protection culture» aiming at promoting the commitment of all those involved, from the top management down to operators through a clear delineation of their respective responsibilities [5]. The motivation of operators during all phases of the work, from planning to past experience analysis, and their sensibilisation through education and training to their individual and collective responsibilities as far as exposures are concerned is of prime importance for maintaining doses as low as reasonable. It is possible here to draw a parallel with the «safety culture» as defined and promoted by the International Atomic Energy Agency in its INSAG-4 document [6]. Mr Sollet from Spain will developed this last point in more details in its presentation but I will say a few words about another important aspect related to the involvement of contractors for improving radiological protection in installations. Because of their important role at the maintenance level, especially in nuclear power plants, contractors must be closely associated to the radiation protection programmes to establish common goals through an ongoing dialogue. In this perspective Electricité de France has introduced a partnership with its contractors within the framework of professional organisations.

To conclude these short introductory remarks I would like to emphasise some of the challenges the nuclear industry is now facing as far as occupational radiation protection is concerned. First, the decreasing trend for collective and individual exposures per unit of energy produced have certainly not reached their asymptote and further progress should be possible at reasonable costs. For the next future, one of the main objectives will be for operators to maintain individual exposures below the 20 mSv range for the personnel being exposed for periods longer than 5 years.

The contribution of dismantling of older installations will progressively become more important in the total exposure of workers. The experience with dismantling is still limited and efforts need to be made to prepare this future activity with the same level of protection for the

workers that will be involved as the one reached with the workers in operating installations. New installations will be designed and constructed in the future and this must be done based on the lessons of the past. The future reactor now in preparation at the European level in cooperation between Germany and France (The so-called European Pressurised Water Reactor (EPR)) is a good example of how ambitious radiation protection objectives can be set at the early phase of a project based on past experience and a careful evaluation of alternative options.

Reduction in occupational exposures at nuclear installations is not only desirable because it directly contributes to the health and safety of the personnel but also because it enhance the safety and the reliability of installations by making their operation more efficient and economical. As a responsible for the safety and the technical operation of the French reactors I am strongly convinced that maintaining a good level of protection of workers at nuclear facilities and, when feasible, improving it, is a guarantee of the quality of operations and therefore of the general safety of the installations. The long-term production of nuclear-powered electricity requires from all operators of the nuclear fuel cycle a solid relationship with their environment mainly based on trust. The demonstration that it is possible to operate the fuel cycle while maintaining individual and collective exposures as low as reasonable achievable plays certainly an important role for meriting this confidence.

Finally, I would like to emphasize that the IRPA-9 Conference, with the participation of so many colleagues from all over the world is an excellent opportunity to exchange experience and views about the various facets of the issues that will be raised during this short symposium, and I hope that fruitful contacts will be made during the course of the Conferences.

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INTRODUCTION

The term "safety culture" first came into use after the Chernobyl accident, and in the course of its brief 10-year existence, the expression has been the source of inspiration for a very large number of texts, procedures and plans which, in many cases, have given rise to specific activities and objectives.

In the field of radiation protection, a monumental effort has been made to implement optimization criteria or ALARA in nuclear installations, and such endeavors have managed to bring about a considerable reduction in the collective doses associated with the operation of nuclear power plants throughout the world.

These programs for designing radiation protection on the basis of optimization criteria and of applying concepts such as efficiency, costs, benefits, rationalization, etc., must be the foundations upon which to create an effective overall radiation protection culture that should covers the whole organization, and which at the same time succeeds in ensuring that the concept of radiation protection transcends the basic framework of activity in the installations, and enters other spheres of the organization until it reaches top level management and its aims are considered to be as important as economic or productivity goals.

BACKGROUND

The primary aim of radiation protection is to prevent the occurrence of harmful health effects in individuals being exposed to ionizing radiation, without unduly limiting the beneficial effects that the activities which give rise to such exposure have upon individuals and society.

The conceptual framework of radiation protection is established in the recommendations set out by the International Commission on Radiation Protection (I.C.R.P.), adopted in national regulations, this in turn being based upon the E.U. Directives, in the case of the countries which are members of the European Union, and the publications issued by the International Atomic Energy Agency (I.A.E.A.) in Vienna.

The radiation protection model for nuclear installations is based on the basic principles of safety and protection announced by I.N.S.A.G. and I.C.R.P., which are outlined below:

- To reinforce "Safety Culture" in order to enhance positive attitudes and bring about an improvement in the behavior of both individuals and the organization.

- Nuclear safety, radiation protection for the workers and the general public and the preservation of the environment are fundamental basic principles which govern the operation of the nuclear installations.

- All activities that involve exposure to ionizing radiation, must yield sufficient benefit to society and to the exposed individuals to outweigh the radiation health

detriment it causes or could cause.

- Radiation sources and installations should be provided with the best available protection and safety measures under the prevailing circumstances.

- The design of the installation and the operating procedures, shall be in accordance with the principle of defense in-depth, so that any potential failures can be compensated by the safety and prevention measures.

- The safety of an installation and protection of the workers and the population, must be ensured by sound management and engineering, quality assurance, training and qualification of personnel, comprehensive safety assessments and attention to lessons learned from experience and research.

- The principle which governs interventions undertaken in the event of an emergency must be justifiable.

These basic principles were later formulated by the I.A.E.A. in a draft of a Safety Fundamentals of the Safety Series collection in two (2) main objectives of radiation protection and safety and eleven (11) basic principles that are derived from those two, concerning practices and actions to which we will be making reference.

- **Objective of Protection**

  To prevent the occurrence of determinist effects in the individuals by keeping the doses below the relevant thresholds and to ensure that all reasonable steps are taken to reduce the occurrence of stochastic in the population at present and in the future.

- **Objective of Safety**

  To protect individuals, society and the environment from harm by establishing and maintaining effective defenses against radiological hazards from sources.

The radiation protection system for practices, is based on four (4) principles:

- Justification of practices
- Dose limits
- Optimization of protection
- Prevention and mitigation

The interventions are governed by two (2) basic principles:

- Justification of intervention

- Optimization of intervention

Three (3) principles of implementation are drawn up from the above, concerning the following:

- Location of sources
- Design and construction
- Operation and use of sources

Finally, details are given of two principles regarding the organization and management of radiation protection and safety:

- Legal framework
- Responsibilities

These basic principles of radiation protection are not new, they were all drawn up, either in identical or similar form, many years ago and form part of the manuals and operating procedures for the installations. Workers are already familiar with them but, and this is the paradox, they have not been able to create a radiation protection culture among the organizations that perform or develop them.

Although it has been in existence for a much shorter period of time, the "nuclear safety culture" has well managed to create, or is managing to weave an effective culture tissue among both organizations and individuals, so that safety can proceed with the ongoing process of improvement. One of the reasons of the success has been its novelty in the formulations and the wish of not lag behind. Whatever the reasons are for the success of the implementation of the "nuclear safety culture", the fact is that it must serve as an example to be followed, and that such a safety culture should likewise be extended to the concepts of radiation protection in such a way that it is no longer possible to distinguish one concept from the other, or at least, that the organization develops ongoing ways of improving radiation protection.

CULTURE OF RADIATION PROTECTION

Without attempting to define culture, it is worth mentioning certain characteristics that are common to any definition thereof, and which can be summarized as follows: a set of principles, values and beliefs jointly shared by all members of an organization, together with standards and expectations that influence the behavior and attitudes of the members of such an organization.

Applied to radiation protection, this concept could then be defined as the entire set of technical and social standards, together with the behavior and attitudes of the individuals concerned with a reduction in the exposure of workers and the general public to ionising radiation.

The element that distinguishes this definition from other general definitions of the radiation protection concept, and that which gives it its cultural nature, is the part that includes the attitudes and the behavior of the individual.

If this definition is correct and the initial thesis that no effective radiation protection culture yet exists, it is also true to say - or at least the conclusion to be drawn is - that either the basic principles are not shared by all levels on the hierarchical scale, a possibility which would have to be ruled out on grounds which are self-evident, or that the behavior and attitudes of the constituents of the organization are not sufficiently active where radiation protection is concerned.

In the light of the above - What are the factors that might exert a positive influence on the behavior of individuals in the field of radiation protection? Without fear of error but without wishing to provide an exhaustive account, the following factors influence such behavior:

- The perception of risk
- Attitudes towards work
- Work group dynamics
- Attitude towards technology
- Attitude towards the organization, hierarchy, leadership, procedures and working habits
- Personal motivation and expectations of improvement

We will now go on to analyze these factors as a whole, together with other conditioning elements regarding radiation protection, with a view to specifying all that is necessary for the creation of a radiation protection culture.

REGULATORY ASPECTS

It is the case in nearly all countries, that the practical application of the principle of optimization of radiation protection ALARA (As Low As Reasonably Achievable) is developed within the organizations, at the same time as the safety culture is being developed.

The driving force behind the application of the principle of optimization has not been equally applied in all countries and organizations, sometimes this being due to the regulatory bodies and at other times because of factors inherent to the organization itself.

In countries where standards are more clearly defined, the initiative generally corresponds to the regulatory bodies, but acceptance and implementation have been

brought about more through imposition than a general consensus that has been convinced of its necessity and, as a whole, the process tends to be rather laborious and time-consuming, probably due to the lack of flexibility of the regulatory practices to be applied to all circumstances. One single and valid implementation pattern does not appear to exist for all organizations.

However, what is relevant in this case, is that all the effort made in implementation must be used and situated in a more general radiation protection framework, in which optimization will naturally be the most important current criterion, but without excluding the attitudes and behavior of individuals.

Other countries, which have less prescriptive controls and a more performance based, the application of the optimization criterion, are allowing the utilities, in a more flexible climate, to find the tools needed for using their specific cultural elements and to create an effective and efficient "ALARA culture" in the organization, and they have a more suitable shape from which to expand upon the objectives already achieved, thereby establishing a genuine radiation protection culture.

STRATEGIC FORMULATIONS

A first step towards ensuring a successful radiation protection culture is to expand its scope of activity. Radiation protection must not only exist within a nuclear power plant framework, but should also operate at the highest decision-making levels of the company.

The current scenario in which nuclear activity takes place, makes the safe operation of nuclear power plants and radiation protection for the workers, the general public and the environment, an ever-increasing necessity. The obligations that those responsible for the operation of nuclear installations have to the workers and to society in general, based upon essential values and quality management attributes, require a clearly defined professional code of conduct, that not only guarantees the responsibility of the professionals to society in such a way that nuclear safety and radiation protection are constantly being perfected, but also by participating in society at the same time, having a role that is clearly subordinate to the latter where service is concerned. In this way, radiation protection becomes a requirement regarding the strategic outlines and company policy, as an ongoing attempt is made to modernize and improve management.

The maximum body for company representation and decision-making, which we shall refer to as the General Management, is a key element for initiating, implementing and maintaining a radiation protection culture, because jurisdiction as regards the allocation of resources, falls upon this body, which is likewise responsible for defining the culture of the entire organization.

The General Management must also include the permanent objective of protecting the workers health, the prevention of accidents, occupational illnesses and diseases and the protection of the environment, as an integral part of its strategic plans. This ongoing preoccupation of nuclear power plants must materialize itself in paying

attention to the constant improvement of radiation protection in nuclear installations.

Such corporate undertakings must take the form of a few clearly defined and easily definable aims, well known to all the organization members, and sufficient resources must be allocated to guarantee their fulfillment. In short, radiation protection should also become part of the management and decision elements of the General Management.

These main company strategies and obligations must bear the following in mind:

- Economic and social factors that affect the protection of the workers and the general public alike, given that the purpose of radiation protection is to provide an acceptable level of protection, without limiting the benefits that nuclear installations give to society and without forgetting their social dimension.

- The integration of radiation protection criteria into the general production, quality and continuous improvement targets, by means of a radiation protection policy based on predictions and goals, the measurement of the results and an analysis of experiences. It has to be possible to measure these goals by establishing performance indicators that must reflect the results of the company's policies.

- This integration of the elements of radiation protection at top level management, ought not to be interpreted as implying a modification to the existing organizational structures, but rather that a natural integration process should take place. This would require that the General Management be provided with an advisor in radiation protection so that the latter could provide the former with constant and continually updated information as regards the achievement of the objectives and a monitoring of all relevant activities.

PLAN MANAGEMENT POLICY AND ORGANIZATION RESPONSIBILITIES

The second step in a commitment towards radiation protection culture concerns the management of the installation.

The Management of the installation must promote the initiatives that lead to the establishing of a radiation protection network involving all the workers. The principle that the workers themselves must be the beneficiaries from radiation protection as well as being those responsible for it, must no longer be a mere maxim, and has to become a reality.

It has been proved that a nuclear power plant operating with a small collective dose has better indicators and performances, as well as being more economical and safer than other plant with the same power but with a higher dose. The dose is, in itself, indicative of the quality of the operation in a nuclear installation. The lower the dose the greater the quality.

A lower dose means less time spent in the radiation zones, more reliable equipment, a greater availability of personnel, lower maintenance and operating costs, more time devoted to training and, in all probability, shorter outages times and thus greater revenues.

Therefore, from a power plant perspective, the plant management provides the main impetus for radiation protection culture, and is also its most enthusiastic adherent, and defines the radiation protection policy as an integral part of its operational goals, allocating the resources required for successfully carrying out this policy and involving all management levels to this end. So, by way of example, the use of tools for cost-benefit analysis can make it possible for a decision to be taken to reduce power, with a view to making an urgent maintenance intervention in operation.

There are a variety of ways in which other managerial levels can be involved. One of the most effective ways is to make an associate between the specific objectives and the dose. The dose is also converted into an additional goal for each department.

Each department has a maximum annual limit that must be efficiently and effectively managed, and reported to Management for their activities and achievements, in the same way as with a budget.

Therefore, the radiation protection service (R.P.S.) for the installation is not alone in its responsibility for this, because the duties must be shared among all those involved in the organization of the plant. This is a natural way for the principles of radiation protection to filter through and throughout the entire installation, and to bring about an integration in the existing organization, operation procedures and radiation protection maintenance criteria, as well as enhancing the creation of interdisciplinary groups for the purpose of programming, planning, carrying out and reviewing/revising all the work.

Of course, this process of change cannot take effect without the involvement of the radiation protection service on the site. If it is the workers who are the protagonists when it comes to radiation protection, they must be offered every assistance in making their tasks easier. The term customer-oriented service must be the aim of the service and has to play an active role to make the work easier for others at all the stages involved in those plant activities in which the worker dose is a common element.

The all too well known mentality of acting as a sort of police force within the controlled zone must be abandoned, and such activities should be replaced by systems of surveillance, advice and support for the tasks. The bureaucratic concept of passing through a control point for the radiation protection service to authorize the work must also be abandon. The Radiation Work Permit must cease to be a Permit, and to form part of the Work Order, thereby enhancing the smooth-running of the radiation protection measures. Furthermore, the R.P.S. ought to provide the executing services with all radiological information concerning the plant, so that these services can plan their tasks well in advance and succeed in obtaining a more efficient management of the personnel in their charge. The use of radiation and dosimetric data bases at the disposal of all personnel, together with the presence, in the controlled zone, of all mimics updated in

real time plus radiological information for the premises will help well-trained foremen to manage both their own radiation protection and that of those in their charge.

WORKER ATTITUDE

If the controlling bodies act in such a way as to enhance the natural integration of the radiation protection criteria into the management of nuclear installation in the widest sense of the term, so that the radiation protection culture becomes an undeniable fact, only worker involvement will remain to be achieved.

The worker thus becomes a basic element in the framework of the radiation protection culture. If active worker participation does not become a reality, the whole process is doomed to failure. Therefore, it is essential to modify the behavior and attitudes of the workers, so that they themselves not only carry out the company's radiation protection policy, but also feel themselves to be fully integrated in the plans and their developments.

Nevertheless, worker involvement is a complex process that is still at the blueprint stage, and a wide range of factors come into play, such as: economic, cultural, social, technical, interpersonal relationships, etc.

In the light of the above, no attempt is being made to establish universal recipes for actions to be taken in order to bring about active worker involvement in the setting up and development of the cultural process. However, the following list of conditions and pre-requisites have to form part of management activities if the workers are to be involved:

- Risk Perception and Attitude towards Technology

  The non-tangible nature of ionising radiation risk, even the expression to define them - stochastic and determinist risks -, the social pressure that tends to magnify such risks, an unawareness of the technology, the possible existence of an inherent yet subconscious tendency to reject, etc., may well have caused the workers to fail to understand the reasons for radiation protection standards and thus not take part in defining them, leading them to a situation in which they complacently accept such rules without really knowing why. Therefore, it would appear obvious that the initial training and any later re-training, must always place emphasis on these basic concepts and explain the risks of ionising radiation in a clear and unambiguous way, by way of analogies involving other more easily recognizable risks. All positive experience obtained in the prevention of professional risks, must be put at the disposal of the radiation protection service and the radiological risks must be included in a more global management of professional risks, given that the activities involved in radiation protection are not exclusive to the technicians, but also require the participation of all those involved in nuclear activities and are of equal importance at all organizational levels.

- Behavior of Management and Senior Staff

  If the workers are to be involved in the culture of radiation protection, management must be able to imbue them with a perception of its own desire to promote the principles, not only of willingness to have such a policy, but also that management itself sees it as being necessary. The existence of Committees at the highest level which deal with radiation protection questions on a routine basis, the frequent presence of senior staff in radiation zones verifying the conduct of the workers and showing an interest in the development of the work being carried out in the radiation zones, making sure that radiation protection matters are as important as safety, economic and operational objectives, etc., must be activities that are conducive to an improvement in worker participation where radiation protection is concerned.

- Education and Training

  Reference has already been made to some aspects of worker training. Recipes that cannot be justified must be abandon, and training activities must be focused upon the general philosophies of protection and correct practice in radiation protection for the personnel and the general public. Familiarity with the codes of conduct in radiation zones and the carrying out of high quality work, are the logical consequences of training programs that have properly assimilated.

- Worker Participation in Management

  The existence of work-outs and improvement groups that integrate the active participation of the workers and in which new lines of action are proposed that, once approved, receive the whole support of management, should help to achieve worker involvement when it comes to developing radiation protection. Seeing that these initiatives are finding an increasingly prominent place in the annual plans of the units, provides a climate that is favorable to achieving more active worker participation in the implementation of a radiation protection culture.

- Information and Communication

  Regular worker participation of the monitoring of the goals of radiation protection in a clearly visible and manifest way, the preparation of information sheets, posters, awareness campaigns, a prompt reply from management to the suggestions made by workers, etc., will mean that the information and communication will flow from management to the workers and vice versa, again creating an atmosphere which is conducive to motivating the workers and gaining their support, when it comes to consolidating and crystallizing radiation protection culture.

CONCLUSIONS

From the above, some conclusions can be drawn in order to establish an efficient radiation protection culture among organizations for the operation of nuclear power plants.

- The organization ought to practice a realistic and open-minded sense of self-criticism, when analyzing the specific weight and scope of the radiation protection in the structures of the company, as well as ensuring that Management really is involved in the radiation protection culture, and in the process of deciding how to extend this discipline beyond the strict confines of the Plant itself; these should be the pre-requisites to defining the starting point from which to achieve a radiation protection culture.

- Experience in the management of professional risks must serve as a reference point for incorporating radiation risks into the overall management of risks in nuclear installations.

- An endeavor must be made to generate transparent information concerning the operations and risks that exist at nuclear installations both in the direction of the workers and of society in general, so that the former with their behavior can help to reduce risks and the latter can be convinced that this technology should be accepted. The image the company conveys to the outside world must be improve through the achievement of goals that are more and more demanding in matters regarding radiation protection.

- Both the behavior of the regulatory bodies and that of the organization itself, in the field of radiation protection, must progress towards being based more on performance targets with rational criteria concerning costs, benefits, economics, clients, etc.

- Programs and techniques must be developed that serve to motivate the workers and get them to participate in all the activities whose purpose it is to improve radiation protection, until such time as radiation questions form a natural part of everyday working life.

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IRPA9
1996 INTERNATIONAL CONGRESS ON RADIATION PROTECTION
APRIL 14-19, 1996
VIENNA, AUSTRIA

ALARA IN PRACTICE: A REGULATORY PERSPECTIVE

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"Radiation Protection at Nuclear Facilities"
Wednesday, April 17, 1996
Vienna, Austria

ABSTRACT:

The paper will start with a presentation of how ALARA programmes fit into the operation management of nuclear power plants. This will be illustrated with the practices adopted in Swedish power plants in recent years. A particular attention will be given on how regulatory requirements related to ALARA have been formulated and implemented in a systematic way by utilities.

The paper will also address all aspects related to inspection and controls of ALARA programmes. The experience gained so far in Sweden allows for some general recommendations about the way to establish an effective framework for the implementation of ALARA programmes and a positive co-operation between regulators and utilities.

INTRODUCTION

Occupational dose reduction is important not only for the health and safety of the workforce but also because the associated requirement for a good management system enhances safety, quality and reliability of the installation and thus the economy of the plant. Indeed, during the eighties and beginning of the nineties progress has been made and occupational doses have decreased in most countries but unfortunately this is not the situation in Sweden. We have learned that there is no time for complacency because as plants become older there is a general tendency of increased maintenance and repair requirements. Moreover, the ICRP Publication 60, which recommends more stringent dose limits, further draws the attention to the exposure of workers and consequently to ways of reducing such exposure.

BACKGROUND

In 1975 the Swedish Radiation Protection Institute started the work with a new regulation on limitation of releases of radioactive substance from nuclear power stations.
Because of the limit 5 manSv/ GW installed electric effect (global collective dose) for the releases from the power station, we had to say something about the occupational collective dose.

In the above regulation we can read the following:

"In the assessment as to whether the practice is justified, the contribution made by occupational exposures must also be taken into account. This also applies when assessing the scale of the radiation protection measures which should be taken ("protection optimisation"). In order to limit the future *per caput* dose it is found to be more suitable from an administrative point of view to limit the occupational exposure by means of special rules which are not associated with the releases of radioactive substances into the environment. The rules for occupational
exposures should nevertheless include a limitation of the collective dose commitment from this exposure per unit of the practice. Until further notice it is recommended that measures be taken with the purpose of keeping the contribution from occupational exposure within the power stations below 2 mansieverts per GW installed electric effect and year."

:, one of the most important reason to give the utilities a guidance value for the collective dose from onal exposures was our knowledge about the dose situation in other countries. During the early s there was a trend to higher doses. This was a situation we tried to avoid (figures 1 and 2).

1

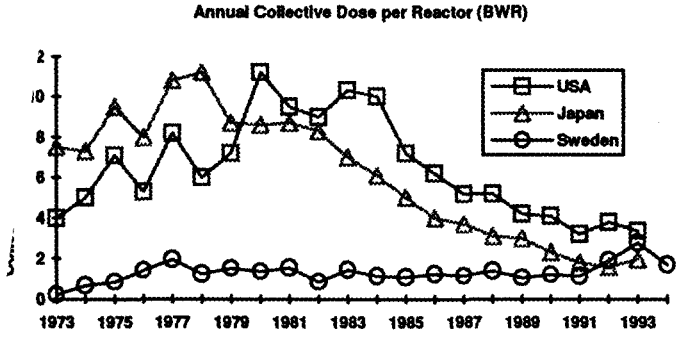
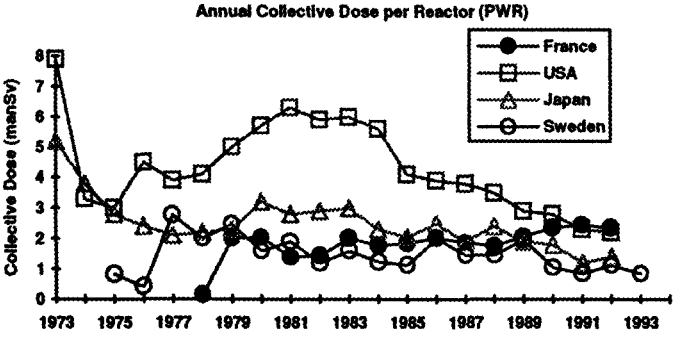
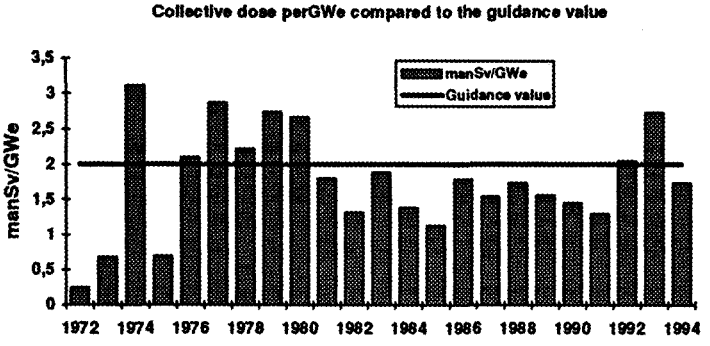


Figure 2



The decrease in doses we saw in the end of the eighties and in the beginning of the nineties (figure 3), didn't give any motivation for implementing new ideas with regard to lower the doses.

Figure 3

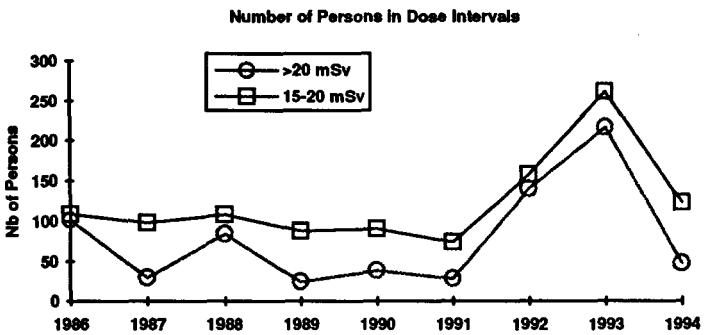


As can be seen from Figure 1, the collective doses in Swedish BWRs have been low, but started to increase in 1992 and continued to do so also in 1993. In 1993 the total collective dose (PWRs included) reached the level

of about 28 manSv, i.e. 2.8 manSv per installed GWe, which in fact exceeds the planning level of 2 manSv per GWe.

Also, the annual individual doses have increased during the same years but are still well below the dose limits (50 mSv for any one single year and 100 mSv as a total for five consecutive years). Moreover, the annual average for all the work force are below the ambition level of the SSI of 5 mSv per year. However, for some groups the average individual doses have exceeded that level. The number of persons with annual doses of more than 20 mSv was in 1993 about 200 but the number decreased to less than 50 for 1994 (figure 4).

Figure 4



REASONS FOR INCREASING DOSES

Due to significant amounts of work, the doses have increased considerably during 1992 and 1993. The increasing doses can partly be explained by the fact that some of the reactors are ageing thus requiring significant maintenance and repair works. Increasing safety requirements resulting in extending inspection programs are also contributing to this. In particular, a significant safety related event happened in 1992, when some insulation material was fed into the inlets of the safety injection systems causing risk of clogging. This event led to repair and modification works at all the BWRs of similar design leading to collective doses of about 7 manSv for the five reactors concerned.

ACTION TO TURN THE TREND

The changed situation with high collective doses has called for the establishment of more fundamental ALARA-programs (ALARA an acronym for As Low As Reasonably Achievable) especially for the BWRs. Using the research funds, the SSI started in April 1993 a significant development program in the field of dose reduction. The Swedish "reactor maker" ABB-Atom was asked to examine the reasons for the increasing dose levels, to assess the expected dose situation during the years to come as well as to give advice on actions to reduce occupational doses Ref. (2). The purpose with the program (DORIS=Dosreduction in Swedish BWRs), was to serve as a basis for the utility ALARA-programs.

The base of the investigation was a comprehensive analysis of exposure and radiation data from the ABB Atom BWRs. Extensive computer simulations were performed to find the factors responsible for this radiation build-up.

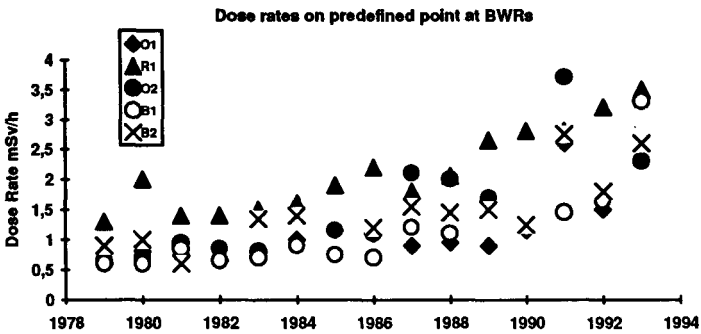
Dose rates

To get a measure of the evolution of the dose rates in the reactor systems, the nuclear industry is measuring continuously the dose rates at a number of places at the reactors. One such series of measurements is shown in Figure 4, from which it is evident that the radiation levels at that particular place on the recirculation loops are increasing with time.

The dose rates are in general higher the closer to the reactor and its primary systems you are. The five oldest BWR-reactors are here of particular interest in that their recirculation loops require significant inspection and

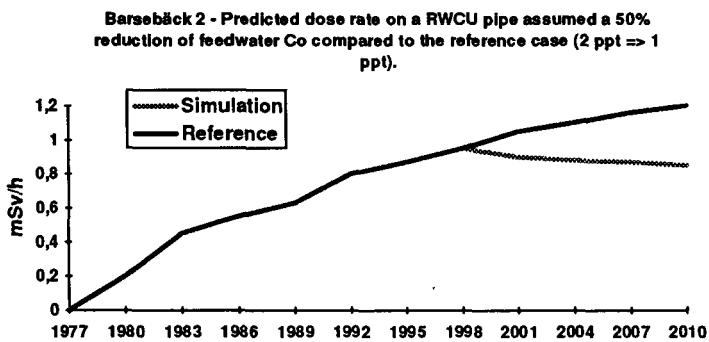
test activities causing important doses to the personnel. In PWR similar problems exist as far as their steam generators are concerned.

Figure 5



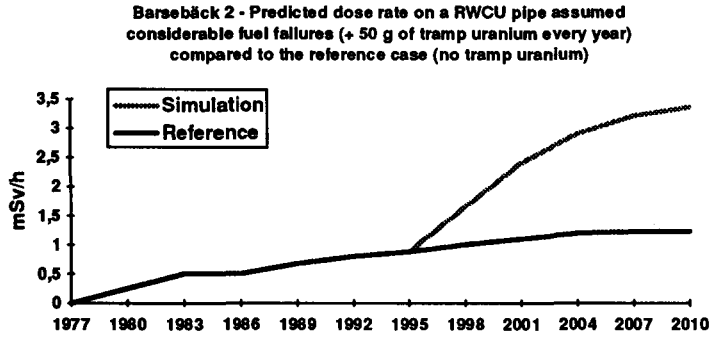
Erosion and corrosion of base material in the reactor systems mean that large amounts of corrosion products are fed into the reactor. An important part of them are deposited onto the fuel, activated and thereafter spread in the reactor systems. Activated corrosion products and in particular Cobalt-60 are the main source of the radiation fields in the nuclear power plants and thus to the resulting radiation doses. Cobalt is one element in stainless steel and amounts up to 60 per cent in the hard facing alloy Stellite, common in valves. The following figure shows the predicted dose rates assuming various levels of inflow of Cobalt to the reactor.

Figure 6



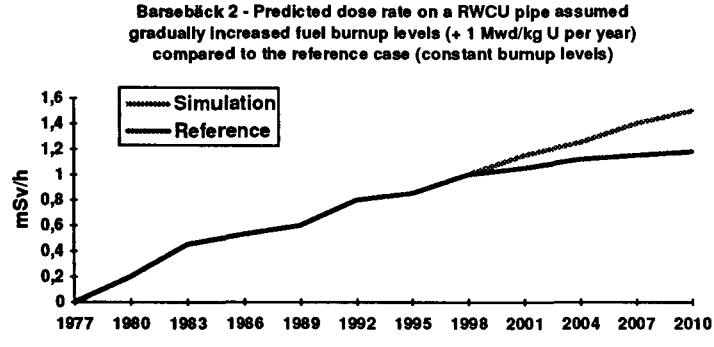
Also, fuel failures are causing an increased spread of cobalt-60 from the fuel to the reactor systems. This phenomenon is presently studied in more detail, but it is already now evident that increased attention to fuel failures are needed also from the occupational exposure point of view as in shown in the figure below.

Figure 7



As shown in the following figure, increased burnup level of more recent BWR fuel is also a factor responsible for the increasing radiation levels in Swedish BWRs. The effect is delayed, which means that the increasing radiation levels turn up after 5 years of operation or more with the new fuel with higher burnup level. The present burnup level is around 40 MWd/kg U.

Figure 8



With the results from project DORIS in fresh memory, we asked ABB, on our behalf, to do a feasibility study on ultrasonic decontamination of nuclear fuel.

The objectives of this study was:

- To determine the possibilities of decontaminating nuclear fuel using ultrasonics
- To outline how an ultrasonic decontamination could be performed and examine technical limitations due to fuel
  and fuel handling
- To determine the long-term aspects of radiation protection and compare the benefits in exposure reductions with the costs

The reasons for performing a fuel decontamination:

- Removal of Co and Co-60 from the cladding surface reduces radiation exposure
- Removal of loosely adherent crud may give a significant decrease of particle transients during reactor shut-down
- Removal of tramp uranium deposited on the core following fuel failures or minor core accidents mitigates the negative effects of these incidents

The study shows us as one of the results that, if two year old fuel elements are decontaminated each year there is a potential for dose reduction with up to 40-50 %.
For a typical BWR it corresponds to a reduction of the collective dose from approximately 2 to 1-1,2 manSv per year.

NEW REGULATIONS

In our new regulations (SSI FS 1994:2) on occupational exposure, new requirements were included. First of all we decided to introduce a new individual dose limit, 100 mSv in 5 consecutive year in addition to the annual individual dose limit which is 50 mSv.

We have also required an extended education and training program in radiation protection, addressed especially to foremen and team-leaders, working for the utilities as well as for contractors. We believe that this program will increase the understanding and motivation of the personnel to more heavily engagement in dose reduction.

Additionally, we believe in an ALARA, or work management approach, i e where the utilities systematically review their strategy towards radiation protection and develop goals in the area of occupational exposure.

We also require that each utility have to prepare an ALARA program. These programs shall contain objectives and dose targets for the short and longer terms, discussions on the basic considerations behind the choice of such objectives and targets, dose reduction plans (source and exposure time reductions to be considered) and ways to monitor, follow up and analysing experience. Finally, the plans shall contain programs for education and training of the workforce as well as the organisational aspects related to all the above.

WORK MANAGEMENT AND ALARA-PROGRAM

The SSI supervises through inspections and reporting the radiation protection situation at the Swedish nuclear power plants and monitored thereby early on the negative trend described above. One action taken by the SSI was to alert the plant managements on the situation and to explain the concerns of the SSI as regards this development. The SSI pointed out the importance of taking powerful measures to strengthen the radiation protection at the plants. To get doses to the ALARA level, it is, according to the SSI, necessary to go through and review *all* the components that make up the doses, one by one, and judge their dose reduction potential in relation to their respective costs Ref (3).

Management approach

The opinion of the SSI is that the management approach adopted towards radiation work can have a major influence on the degree of radiation exposure in the workplace. Experience has also shown that effective dose reduction needs firm management involvement and support as well as appropriate dosimetric systems and other tools.
In other words, the organisation, control and follow up of radiation works to ensure that doses are ALARA is a management issue and must not be left to the radiation protection department alone.

Also, the involvement of top management seems to be a very important measure to improve the attitude and awareness for effective radiation protection of all personnel. This is further enhanced if specific radiation protection goals are set up and all the personnel is informed about the importance management attaches to the achievement of those goals.

Policy defined by management is not enough, however, continuous support is needed; a support that has to penetrate and be made known at all levels in the organisation. A key to successfully pursuing ALARA seems to be **commitment**; a commitment that has to exist at all levels of management, not just at the top level. Many countries have realised this and several of them have defined ALARA programs and some more are in the progress of doing it.

In Sweden, the above mentioned regulatory requirement to prepare an ALARA program by each utility is part of such a management approach. These programs shall contain objectives and dose targets for the short and longer terms, discussions on the basic considerations behind the choice of such objectives and targets, dose

reduction plans (source and exposure time reductions to be considered) and ways to monitor, follow up and analysing experience. Finally, the plans shall contain programs for education and training of the workforce as well as the organisational aspects related to all the above.

Industry response

The Swedish nuclear industry has responded positively to the initiatives of the SSI. All the nuclear power plant managements have taken actions in a very constructive way, administratively as well as technically. They have clarified the responsibilities for radiation protection by delegated it to the line management. The industry has initiated a number of development projects aiming at dose reduction and is in the process of developing the ALARA programs referred to above, including explicit goals and targets for radiation protection for all the reactors. Finally, one has to emphasis that a number of technical actions have been taken or are in the process of being taken at each individual power plant, e.g. finding replacement material for Stellite, optimisation of water chemistry, decontamination of components and systems and development of improved strategies for fuel burnup and handling of fuel failures.

THE FUTURE

SSI thinks that is it necessary to go through and review *all* the components that make up the doses, one by one, and judge their dose reduction potential in relation to their respective costs. Such component comprise actions that will contribute to reducing the dose rates in the plants, and in particular to radically decrease the inflow of cobalt into the reactor core. Here the Stellite reduction efforts will be crucial. Other actions that most likely will be important are chemical decontamination, permanent shields, prevention against fuel failures and the limitation of fuel burnup by a skilled operation strategy.

When it comes to the other components that make up the collective doses, namely time and number of persons involved, planning, education and training as well as automations and robotisation will be important. However, a systematic approach will be needed including a review of

- working conditions (protections, working environment, etc.)
- worker characteristics (qualification, experience, information about the work, etc.)
- work organisation (scheduling, preparation, co-ordination, etc.).

The modification of one or several of the above factors will have a direct impact on the productivity of workers and then influence directly the exposed time and usually the costs of carrying out the work.

In addition to the above actions the SSI has emphasised to the nuclear industry the importance of having the management of the plants engaging themselves in radiation protection issues. Also, the SSI sees the importance of maintaining an open dialogue with all the work force including the outside workers that participate in the maintenance of the plants, particularly during the outage periods. Finally, the importance of education and training as well as setting up systems allowing analysis and feed back of experience has been pointed out by the SSI.

CONCLUSIONS

The nuclear industry has come to a point where greater emphasis on systematic dose reduction is needed. This is not only for the health and safety of the personnel, but also for maintaining the safety and economic viability of the plants and for public acceptance purposes. In Sweden the doses have increased significantly during a couple of years and great efforts are needed to turn the trend. To succeed in this, the plant management has to adopt a structured approach to radiation risks and tackle all the factors influencing exposure. The ALARA principle applied through, and by, all levels of management and in all important works seems to be a useful instrument in this respect. Management should deal with doses as they deal with money, i.e. establish systems containing targets, mechanisms to follow the results and to take corrective actions, when deviations so require. In this respect it should be pointed out that the dose targets should be challenging but possible to achieve, expressed in measurable terms and accepted by those who are responsible for the results at the company.

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INDIVIDUAL MONITORING FOR EXTERNAL RADIATION, SOME CONCEPTUAL AND PRACTICAL ASPECTS

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INTRODUCTION

Statistics show that, at least in the Western world, each member of the public is, on average, subjected to one X-ray examination per year. It is recognized that radiation doses due to the medical application of ionizing radiation are far from negligible but also that the benefits gained are significant, often life saving. Although reference values are presently being considered, dose limits do not exist for patients and doses are assessed only on an ad hoc basis.

For those, however, who professionally deal with sources of ionizing radiation the situation is quite different. For this category of workers a system of dose limitations has long ago been developed and includes the measurement of the actual exposure of the individual worker. In 1982 the International Commission on Radiological Protection (ICRP) state, in Publication 35 (1), that: "The primary purpose of individual monitoring is to obtain an estimate of the mean dose equivalent and of the effective dose equivalent in significantly exposed tissues. This information is useful in limiting radiation doses to individual workers and in demonstrating compliance with the full system of dose limitation recommended by the Commission and with authorized limits." Indeed, monitoring of workers constitutes an integral part of radiological protection. The main objectives of individual monitoring and some of the benefits that may accrue from it are:

- demonstration of compliance with legal requirements and regulations
- provision of information for the evaluation of the dose received by an individual in the event of an accidental exposure
- demonstration of good working practices and the adequacy of supervision and training which will motivate workers to reduce their exposures to the lowest possible level
- information on collective doses as a useful tool for risk-benefit analyses.

In 1991 ICRP (2) recommended a concise system of radiological protection, including dose limits for workers. This system formed the basis for the safety standards of international organizations such as the IAEA and the European Union from which national regulations are derived.

PRIMARY AND OPERATIONAL QUANTITIES IN INDIVIDUAL MONITORING

The ICRP system of dose limits (2) is based on the concept of effective dose, E. E is the sum of the weighted doses in all tissues and organs of the body as given by the expression

$$E = \sum_T w_T \cdot H_T \qquad (1)$$

where H_T is the equivalent dose in tissue or organ T and w_T is the weighting factor for tissue T. The equivalent dose in tissue T is given by

$$H_T = \sum_R w_R \cdot D_{T,R} \qquad (2)$$

where w_R is the radiation weighting factor and $D_{T,R}$ is the absorbed dose averaged over the tissue or organ T due to radiation R. Values for w_T and w_R are given in ICRP Report 60 (2).

The purpose of this *primary quantity*, E, (also called the *protection or limiting quantity*) is to provide a measure of detriment from stochastic effects for non-uniform irradiation of the body which could be equated to the detriment for uniform irradiation (3). Although it was recognized that the risk associated with a given exposure would vary – by up to a factor of about 10 – with the age and sex of the individual exposed, only one set of values for w_T (derived from averages for a population of workers) has been recommended by ICRP as being appropriate for the protection of any worker regardless of these sources of variability (2, 3). It is clear that effective dose, although in concept applicable to individuals, corresponds to an expectation of average detriment and therefore is, in practice, assumed to be applicable to an average population of workers.

In most national and international regulations annual dose limits are also given in terms of the primary quantity and are taken to apply to the individual concerned. Therefore, in principle, E (formerly H_E the effective dose equivalent (3)) is the quantity that should be determined in radiation protection dosimetry. However, this requires detailed knowledge of the equivalent dose in various organs and tissues in the body, and thus the quantity is difficult to assess and impossible to measure directly. For this reason ICRU, in 1985, introduced in Report 39 (4) the concept of *operational quantities* which was further elaborated in ICRU Reports 43 and 47 (5, 6). Presently, for individual monitoring the operational quantity is the *personal dose equivalent, $H_p(d)$*. This quantity is defined as the dose equivalent in soft tissue (ICRU 4-element composition) at an appropriate depth, d, below a specified point on the body (6) (usually taken to be the point at which the dosemeter is worn). For weakly penetrating and strongly penetrating radiations the recommended depths are 0.07 mm and 10 mm respectively (notations: $H_p(0.07)$ and $H_p(10)$).

The concept of personal dose equivalent has been created to provide a quantity considered to be metrologically sound and able to give a reasonable estimate of the protection quantity. In ICRU Report 43 (5) it was shown that, for uniform irradiation of the body, $H_p(10)$ could be considered a satisfactory representative of the effective dose equivalent, H_E (3), overestimating this (former) protection quantity by only some 20% for photon radiations of 50 keV and greater. In essence, this is also true for the effective dose, E, as introduced by ICRP in 1991 (2). For lower photon energies the overestimate, however, is significant and may be as high as a factor of 5 at 15 keV. (It is useful to note that practical dosemeters tend to underestimate $H_p(10)$ in this energy region and therefore may give a better estimate of E, see below).
The definition of $H_p(10)$ implies that, for a given orientation of an individual in a given radiation field, this quantity is multi-valued, it being dependent on the individual concerned, i.e. on the individual's size and shape (influencing, by scattering and attenuation, the dose equivalent at the depth d) and on the "specified point on the body" where the dosemeter is worn. These variations may, however, not be judged important in the context of the much larger variations in detriment resulting from sex and age.

It is worth noting that, as far as practical measurements are concerned, many current dosemeters do not strictly measure the dose equivalent in the adjacent tissues because they often use non tissue equivalent detectors covered with filters of various materials in combination with algorithms applied to match the calibration quantities under calibration conditions. Also, significant variations in dosemeter readings may occur due to variations in the separation of dosemeter and body and resulting from differences in the energy and angle characteristics of the radiation backscattered from the body (7). These variations, however, critically depend on the dosemeter design.

DOSIMETRIC REQUIREMENTS FOR PERSONAL DOSEMETERS

Personal dosemeters should be capable of providing reliable measurements of the appropriate quantities $H_p(0.07)$ and $H_p(10)$. ICRP (1) requires that in individual monitoring the maximum error in the measurement of a dose at the level of the annual dose limit should not exceed a factor of 1.5 at the 95% confidence level. This means that the measured dose should be within the interval of –33% and +50% of the conventional true value of the quantity of interest (8, 9, 10). Hence, if S is the standard deviation of the measurement,

$$1.96\ |S| \leq 0.5\ (\ 0.33 + 0.50\ ) \rightarrow S \leq 0.212 \qquad (3)$$

Here S is supposed to include all errors – both random and systematic – in the dosimetry system. If δ_r and δ_s are the resultant random and systematic standard deviations respectively then S can be obtained from:

$$S = (\delta_r^2 + \delta_s^2)^{1/2} \qquad (4)$$

Following the convention, δ_s can be calculated by assuming that systematic uncertainties follow a rectangular probability distribution (8, 9).
Among the systematic errors, the variation in the response of the dosemeter as a function of photon-energy and angle of incidence is the most important, the remaining systematic errors usually being relatively small.

The corresponding error and its relative standard deviation, ($\delta_{E,\phi}$), obviously depends on the design of the dosemeter, which includes the choice of the detector material(s), the filters used and the housing.
Hence (4) can be rewritten as:

$$S = \sqrt{\delta_r^2 + \delta_s^2 + \delta_{E,\phi}^2} \leq 0.212 \qquad (5)$$

where δ_r and δ_S are the relative standard deviations of the resultant random and remaining systematic errors

respectively.

By Equation (5) the performance criteria for personnel dosimetry systems are expressed in general terms, without specifically limiting the uncertainties of any of the parameters involved. This approach leaves some freedom with respect to the physical and technical characteristics of the dosimetry system used. If, for example, the radiation energy and angular dependence is small, relatively large random errors can be tolerated, and vice versa. This implies that if, for a particular dosimetry system, the relative standard deviations due to the overall random error (δ_r) and the remaining systematic error (δ_s) are known, the maximum allowable value, Δ, for the dosemeter design error ($\delta_{E,\phi}$) can be determined. For example, if $\delta_r = 0.06$ and $\delta_s = 0.04$, then $(\delta_{E,\phi})_{max} = \Delta = 0.20$ (10).

TYPE TESTING OF PERSONAL DOSEMETERS: THE PHANTOM PROBLEM

Type testing is the determination of the dosemeter response characteristics under laboratory conditions that come reasonably close to real life situations. E.g., when testing for $H_p(10)$ for photon radiation, a personal dosemeter is irradiated on a phantom under standardized laboratory irradiation conditions (8, 11) and its response as a function of radiation energy and angle of incidence, $R_{E,\phi}$, is compared to the dose at a depth of 10 mm in the phantom. $H_p(10)_{phantom}$ can be calculated from air kerma measurements (phantom absent) by multiplying K_a by $C(E,\phi)$, the conversion coefficient for energy E and angle of incidence ϕ (8, 12, 13). Ideally

$$R_{E,\phi} = H_p(10)_{phantom} = C(E,\phi)\, K_a \qquad (6)$$

Although the concept of operational quantities would suggest a straight forward recipe for type testing of personal dosemeters, this is not true. ICRU gave some hints in Report 39 (4): "The calibration of the dosemeters is done under simplified conventional conditions at the depth d in an appropriate phantom. For dosemeters to be worn on the trunk, a suitable phantom is the ICRU sphere" (30 cm in diameter and consisting of ICRU tissue). However, not only is it impossible to fabricate ICRU tissue, also there are several practical problems in the manufacture and use of spherical phantoms. Therefore, taking into account extensive discussion on this topic, ICRU gave further guidance in Report 47 (6): After indicating that "the calibration of dosemeters is generally performed under simplified conditions on an appropriate phantom", various phantoms in use are listed and, for the reason that "it is .... desirable to achieve uniformity in calibration procedures", the 30 x 30 x 15 cm PMMA phantom is recommended. The report also extends the definition of the operational quantities for individual monitoring to include the dose equivalent at a depth d in a phantom made of ICRU tissue, and states: "The quantity to be used for calibration is therefore $H_p(10)$ in a phantom having the composition of ICRU tissue, and the same size and shape as the calibration phantom". This "is the dose equivalent at the depth d below the point where the dosemeter is to be calibrated".
As was done earlier for the ICRU sphere, coefficients were calculated to convert air kerma to $H_p(10)$ at various depths (including 10 mm) in the ICRU slab and phantoms of different shapes and materials for various angles of incidence of the radiation (12, 13, 14). Detailed type testing procedures have also been published (8, 15).

As mentioned above, PMMA has been suggested and even recommended (6) as a satisfactory surrogate for ICRU tissue material. Although PMMA is cheap and easily available, it has the disadvantage that it produces dosimetric complications resulting from differences in absorption and backscatter as compared to tissue. It is important to be aware of these complications, especially because the interpretation of type test results in terms of performance criteria (8, 15) is a crucial part of the official approval of a dosimetry system for individual monitoring. In view of this the concept of using PMMA phantoms in conjunction with PMMA related dosimetric reference data for the purpose of type testing (and/or for the design) of personal dosemeter is a very dangerous one, for various reasons: Dosemeters designed to correctly measure the dose in the ICRU tissue slab phantom may, when type tested on a PMMA phantom, show a relative response as a function of photon energy and angle of radiation incidence different from $H_p(10)$ in PMMA and, as a result, may not comply with the officially required performance criteria and hence not be approved. Alternatively, dosemeters with relatively poor response characteristics in terms of $H_p(10)$ in tissue, might show fortuitous agreement with the results of tests made on a PMMA phantom.

One might indeed wonder whether a dosemeter which is designed to correctly measure the dose in a PMMA phantom (in order to make it pass the test) can be expected to, as intended, correctly measure the dose in soft

tissue. The fact that, between 25 and 70 keV, $H_p(10)$ in PMMA is about 20% higher than $H_p(10)$ in tissue (7) may find some compensation by the increased fluence (some 10 to 15%) at the PMMA phantom surface. This compensation, however, only applies to dosemeters that are sensitive to and able to correctly measure the dose from backscattered radiation. Clearly a dosemeter, which is entirely insensitive to backscatter (hence does not respond to the presence of a phantom or an individual at all), designed to correctly measure $H_p(10)$ in PMMA and successfully tested on a PMMA phantom will, when worn by a person, be in error by some 20% in the photon energy range between 25 and 70 keV (important in diagnostic radiology). Factors meant to correct for the larger contribution of backscatter radiation, produced by PMMA (15, 16), are not generally applicable because such corrections strongly depend on the design of the dosemeter.

From the above it may be concluded that, from both a conceptual and a practical point of view, the PMMA slab phantom can be considered unsuitable (7). This seems to be gradually becoming recognized. ISO is in the process of adopting a better substitute for the ICRU tissue slab, i.e. a water filled PMMA phantom of the same dimensions (30 cm x 30 cm x 15 cm) the front wall of which is thin, 2.5 mm, while the remaining walls are 10 mm thick (17). Backscatter properties of this phantom are very close to (i.e. less then 3% different from) those of the ICRU tissue slab. $H_p(10)$ in the latter will still be the reference quantity. This will certainly help to dissolve the confusion which still remains in the minds of many who run an individual dosimetry service. It will, however, not solve the still remaining problem of type testing dosemeters for angular responses at angles larger than 60 degrees (7).

TYPE TESTING FOR PHOTON ENERGY AND ANGULAR RESPONSE

In the case of type testing of personal dosemeters for photon-energy and angular dependence it is recommended (8) to determine the energy response for both $H_p(0.07)$ and $H_p(10)$ at four angles of incidence: 0° (normal incidence), 20°, 40° and 60°. Irradiations should be made on a suitable phantom (preferably the 30 x 30 x 15 cm water filled PMMA ISO slab (17)) using the narrow spectrum reference radiations as specified by ISO (8, 11). A combined energy and angular response can be calculated and plotted by, for each energy E, averaging the response over all angles of radiation incidence. The resulting average response, $\overline{R}_E$, then becomes:

$$\overline{R}_E = 0.25 \left(R_{E,0} + R_{E,20} + R_{E,40} + R_{E,60} \right) \tag{7}$$

$R_{E,\phi}$, the relative response at energy E and angle ϕ, is obtained from:

$$R_{E,\phi} = \frac{\left(H_{E,\phi}\right)_m}{\left(H_{E,\phi}\right)_t} \tag{8}$$

where $\left(H_{E,\phi}\right)_m$ = the measured dose and $\left(H_{E,\phi}\right)_t$ = the conventional true value. $\left(H_{E,\phi}\right)_t$ should, when type testing for $H_p(0.07)$ and $H_p(10)$, be taken to be the dose equivalents at depths of 7 mg.cm^{-2} and 1000 mg.cm^{-2} respectively in the ICRU tissue equivalent slab phantom (*not* in the ISO phantom, which only serves the purpose of creating realistic back scatter conditions!). In Equation (7) the responses at each angle are weighted equally, which is assumed to be acceptable in practical situations where irradiation conditions are likely to be two dimensional rotationally isotropic (18).

The average uncertainty related to the angular response at energy E is represented by $| \overline{R}_E - 1 |$.

If, as indicated above, the maximum allowable value of $(\delta_{E,\phi})$, i.e. $(\delta_{E,\phi})_{max} = \Delta$, due to variations in the combined energy and angular response is calculated according to Equation (5) then the performance requirement for $\overline{R}_E$ becomes:

$$| \overline{R}_E - 1 | \leq 1.96 \Delta \tag{9}$$

for all energies between 15 keV and 1.25 MeV.

A typical example of the response for $H_p(10)$ for various photon energies and angles of incidence of a TLD (LiF) based personal dosemeter (10) is given in Figure 1.

SHOULD AN INDIVIDUAL DOSEMETER BE SENSITIVE TO BACKSCATTER?

At first sight effective dose, as defined by ICRP in Publication 60 (2), would seem – and is usually considered – to be an individual oriented quantity. Not true: Even if it were possible to obtain the equivalent doses in the organs of the individual concerned, the next step would be to weight these values using tissue weighting factors as given by ICRP. These factors, however, are – as mentioned earlier – population values averaged over sex and age. E therefore cannot be assessed for a given individual. Conceptually, E is more related to a mathematical model (such as the anthropomorphic models ADAM and EVA as used in Monte Carlo (MCNP) calculations), with population averaged weighting factors. Both from a fundamental and a practical point of view it is clearly illogical to have a multi-valued individual related quantity, $H_p(10)$, as an estimator of a single-valued, sex and age dependent quantity (7).

If one were to attempt to estimate an individual specific value of effective dose from a measurement of personal dose equivalent, it would at least be required that variations in E, caused by differences in H_T resulting from differences in shape and/or size of the individual, are reflected in similar changes in $H_p(10)$ and (for the purpose of measurement) in the reading of the dosemeter worn on the surface of the body. However, it is generally to be expected that $H_p(10)$, as well as the radiation field at the surface of the body and therefore – possibly – the reading of the dosemeter, will increase with increased size of the individual, owing to increased backscatter, whereas the equivalent dose to organs is expected to decrease owing to greater attenuation of the radiation within the body. Furthermore, E includes in its formulation the term $0.05\ H_T$ for the female breast, which can neither be reflected in $H_p(10)$ for the male nor in the reading of a dosemeter (assuming that the latter is not different for female and male workers). Consequently, the answer to the question: "Should a personal dosemeter be sensitive to backscatter? " is: No. This is not meant to imply that personal dosemeters should *not at all* be sensitive to backscatter, which would be difficult to realize in practice. It just indicates that there is no point in expending a great deal of effort to design personal dosemeters capable of responding to or measuring radiation backscattered from the wearers body.

It would, in fact, seem more straight forward to design dosemeters to assess $H_p(10)$ in the ISO water filled slab phantom, irrespective whether they will or will not be responding to backscatter. It can be assumed that such dosemeters, when worn at an appropriate position on a person, would measure $H_p(10)$ with sufficient accuracy to provide an estimate of E as good as $H_p(10)$ does (or better). If they were also performance tested on this type of phantom, difficulties in interpreting the results would also disappear. Moreover, when justified by the magnitude of the exposure and provided information on the energy and angular distribution of the radiation concerned is available, corrections can be made to the reading of the dosemeter to improve the assessment of $H_p(10)$. Further corrections, based on the relationship between $H_p(10)$ and E for the actual radiation condition, could then be applied to obtain a better assessment of E. Final effort could then still, in principle, be made to correct the assessed value of detriment knowing details of the individual exposed.

It has been suggested (7) that a more direct estimate of E from a dosemeter reading should be considered at some future stage. Arguments against this approach, e.g. that this is only valid if the field is uniform, are also applicable to the utility of $H_p(10)$. Where the field is non-uniform. e.g. resulting from the wearing of lead aprons, ICRP states (2) that more extensive measurements may need to be made. Here more than one dosemeter is frequently worn and the estimate is made of E, not of $H_p(10)$ which can have no meaning for a multi-badged person.

Following this suggestion, dosemeters might be designed to have an air kerma response matching that of E, rather than of $H_p(10)$. This means that the relative response of a dosemeter as a function of energy and angle of incidence, $R_{E,\phi}$, should follow as closely as possible the relation

$$R_{E,\phi} = C(E,\phi)\, K_a \tag{10}$$

where

$$C(E,\phi) = E\,(E,\phi)_{Adam(or\ Eva)} / K_a. \tag{11}$$

The fact that many (if not most) TLD based dosemeters, used today, show a severe underresponse relative to $H_p(10)$ at low (photon) energies and therefore compensate for the fact that $H_p(10)$ gives a significant overestimate of E in this energy region, seems to support this idea.
Type testing of dosemeters could be performed on a standard phantom (e.g. a water-filled thin walled PMMA elliptical cylinder may be preferred) the backscatter characteristics of which are adequately similar to the mathematical models (ADAM, EVA.). The field quantity used to determine the response characteristics

would be unchanged, i.e. air kerma for photons. Difficulties arising because of inappropriate wearing positions are no different from those currently for $H_p(10)$.

DOSE RECORD KEEPING

Often the concepts and physical aspects of personal dosimetry gain more attention than the recording, reporting and analysis of the data obtained. It should be borne in mind, however, that proper recording of radiation doses is an equally essential part of the process of individual monitoring and shares in the same objectives.

Specific objectives of dose record keeping are (19):

- to inform the workers and the management on occupationally received doses to, e.g. optimize operations and to demonstrate the effectiveness of ALARA.
- to prevent overexposure of individuals, particularly important for outside workers.
- to provide data for analysis of dose distributions and for evaluation of trends, on the basis of which radiation protection procedures and monitoring programs can be developed and, if necessary, improved.
- to provide data for legal and medical purposes
- to provide data for epidemiological studies.

From the above it will be clear that a dose record keeping system serves a broader goal than just satisfying the local, regional or national legal authorities. It should rather be a Dose Registration and Information System (DRIS) (8,19), to be used as a multi-purpose tool in radiation protection. For the larger Individual Monitoring Services dose record keeping systems are often integrated in an overall automated management system.

It should be emphasized that a DRIS– if properly set up – contains information that can be used as a tool for assessing the performance of a personnel dosimetry service by statistical analysis of measuring results, especially those at low doses (see below).

Within the European Communities, co-operation and exchange of personnel between countries is increasing as a result of an on-going integration of the member states. This will enforce the need for international data communication by means of computer networks. The latter is even explicitly mentioned in the CEC Directive 90/641, which deals with the "operational protection of outside workers exposed to the risk of ionizing radiation during their activities in controlled areas" (20). These developments emphasize the necessity for international harmonization of dose record keeping systems and categorization of the data they contain. Such harmonization is of vital importance to solve the problems related to significant differences in the way international bodies – CEC, UNSCEAR, NEA, WHO – approach statistical analyses of occupational exposures.

Because of the availability of relative user friendly data base management systems, setting up a dose record keeping system may seem relatively easy. It should, however, be emphasized that numerous difficulties can be encountered in both programming and in developing adequate procedures. Special attention should be given to size, structure, accessibility, dissemination of information, organization and quality assurance procedures. This is especially important if a DRIS covers a wide range of customers (e.g. hospitals, nuclear power plants, military installations etc.) and even more so if the system operates as a National Dose Registration and Information System (NDRIS). For the latter it is crucial to keep the dose information up-to-date which, however, may be difficult if workers move from one employer to another or if they switch from one monitoring service to another.

Because it is of paramount importance that the effectiveness of the system, the quality of the recorded data and the confidentiality are guaranteed, the persons responsible for the system need to be well trained and competent.

STATISTICAL ANALYSIS OF ROUTINE MONITORING DATA

As stated above, statistical analysis of the dose data stored in a DRIS is a useful tool for quality control, especially for the performance at low dose levels. Regular analysis of the dose distributions can show the stability of both the dosemeters and the measuring equipment (21). Depending on the population monitored, often the majority of the workers will not receive any occupational dose at all. This means that the frequency distribution of the occupational dose (i.e. the measured dose minus the contribution due to natural background) will show its maximum at 0.00 mSv, i.e. the modal dose is 0.00 mSv. Due to variations in the natural background from place to place and due to intrinsic variations in the dosimetry system, the dose distribution will have a gauss like shape, which will be slightly skewed to a somewhat higher level, due to real occupational doses. An example is given in Figure 2, in which data taken from the TNO individual monitoring service are plotted. The width of the dose distribution should be in agreement with independently obtained information on variations in the natural background and noise of the dosimetry system. Consequently variations in the position of the modal dose or in the width of the dose distribution are indicators for possible

problems with the dosimetry systems at very low dose levels. Using more advanced techniques the various contributions to the width of the distribution can be traced, giving an indication where to look for the source of problems, which allows for fine tuning of the system.

QUALITY ASSESSMENT IN INDIVIDUAL MONITORING: THE ULTIMATE PROOF

Quality Assurance (QA) and Quality Control (QC) are more and more becoming integral parts of our societies. Originally QA and QC mainly applied to the manufacturing of industrial products, to-day also public services are required to provide high quality products for a reasonable price. This can be guaranteed only by developing and implementing a QA/QC program that is tailor made and touches the entire organizational and technical structure of the service in question. Some of the basic elements of such a program, applied to Individual Monitoring Services (IMS), have been published (8). These should be considered complementary to more general requirements as published by ISO (22).

A QA/QC program involves every single step in the process of providing dosemeters to the customer, reading and evaluating the dosemeters returned and reporting the results. It is, however, very useful to in addition test the overall performance (dosimetric, organizational, administrative and financial) of the IMS as well as the postal service. This is done at TNO since 1987, by creating a "QA-subscription" for three "dummy" customers to whom dosemeters are assigned: 10 on a bi-weekly, 6 on a four-weekly and 6 on a quarterly basis respectively (21, 23). These dosemeters are randomly chosen, indistinguishable from those for the regular customers, and mailed to the private address of a member of the staff. Subsequently all dosemeters are taken back to the laboratory. Of each set 2 dosemeters are stored in a lead pig, 2 are placed in the IMS processing area and 2 are irradiated to 2.00 mSv ^{60}Co gamma radiation. The 4 remaining dosemeters from the bi-weekly set are, in duplicate, irradiated to 0.20 and 12.00 mSv respectively. At the end of each issuing period the related dosemeters are taken back home and returned to the laboratory by regular mail. Evaluation of the dosemeters and reporting of the results follow the standard (highly automated) procedures. Copies of the reports are sent to the responsible staff members.

Statistical analysis of the results of the QA-subscriptions gives both the accuracy and the precision of the dosimetry system under routine conditions. Figure 3 shows, as an example, the results of the dosemeters of the biweekly QA-subscription which were irradiated to 2.00 mSv. In the figure the statistical parameters for each year are given. Here avg denotes the average, std the standard deviation in the individual values, Hm/Ht the quotient of the measured dose and the conventional true dose and std/Ht the relative standard deviation in the measured values. The annual average of Hm/Ht ranges from 0.99 to 1.03. The relative standard deviation in the individual dose assessments ranges from 7% in early years to 4% in the more recent years.

Applying regression analysis on the standard deviation for all five dose levels used, results in a value for the relative standard deviation over a broad dose range. In 1994 this relative standard deviation was 4.1% (correlation coefficient 0.999) (23).

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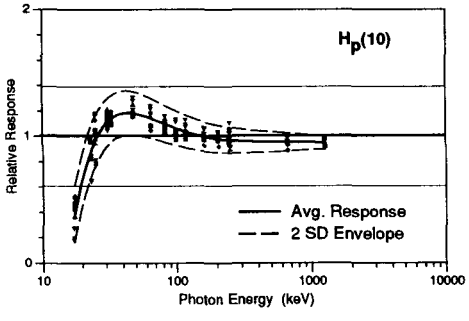

Figure 1. Combined photon energy and angular response data for $H_p(10)$ of a LiF based personal dosemeter.

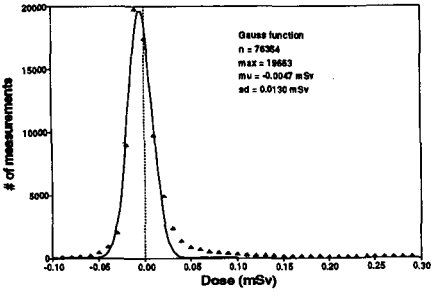

Figure 2. Distribution of net occupational doses, taken from the TNO IMS

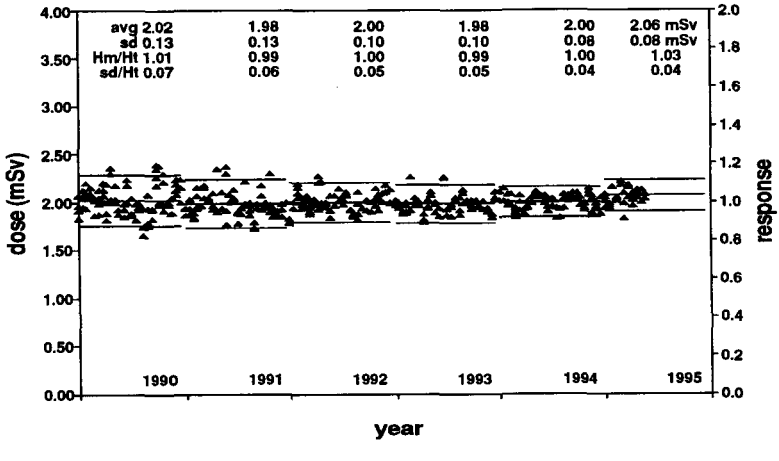

Figure 3. Results of the TNO IMS QA subscription dosemeters, irradiated to 2.00 mSv. The solid lines represent the average and the confidence limits.

ENVIRONMENTAL RADIATION MONITORING IN THE CONTEXT OF REGULATIONS ON DOSE LIMITS TO THE PUBLIC

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INTRODUCTION

Environmental radiation monitoring is performed for the purpose of assessing the dose to the general public from a nuclear or radiological facility, usually to demonstrate compliance with regulations that limit the allowable dose to the public from manmade sources. Measurements are often made at the property line of the facility or further off-site. Monitoring may be performed by the operator of the facility, and in some cases by government regulatory authorities. Assessment of the total effective dose to the public requires modeling of all source pathways. Wherever possible, input to such models should include measurements of appropriate parameters. For example, measurements at plant vent monitors and meteorological data may be required to make projections of the dose to populations at various locations. In addition, monitoring of air, soil, water, and biota may be required to determine the internal dose, and direct measurements may be required to determine the external dose. This paper focuses only on measurements of the external component of the total dose. Considered is the application of current and proposed public dose limits in the context of the natural radiation background, current measurement technology, appropriate quantities, and the development of standards for the determination of external dose.

ICRP PHILOSOPHY ON DOSE LIMITS FOR THE PUBLIC

In most countries, regulations on radiation dose to the public reflect the basic philosophy of the International Commission on Radiological Protection (ICRP). Wherein, it is assumed that stochastic effects occur with a probability (rather than a severity) that is a function of dose, with no threshold. In determining an acceptable dose limit for the public, ICRP report 26 (1) describes an approach that allows for a risk for fatal cancers to be comparable to risks that are accepted for other aspects of everyday life, such as public transportation. The Commission found that this risk would correspond to a whole body annual dose of 1 mSv (100 mrem). However, they recommended that it would be sufficient to set a whole body dose equivalent limit of 5 mSv y^{-1}, applied to the mean dose received by a "critical group", that is a set of individuals most highly exposed.

ICRP report 60 describes an alternative approach, which is to consider variations in the dose from natural background radiation. Consistent with both approaches, the Commission currently recommends a limit to the general public of 1 mSv y^{-1} (effective dose) (2). The ICRP further specifies that, under special circumstances, a higher effective dose may be allowed in 1 year as long as the average over 5 years does not exceed 1 mSv per year. These limits refer to the whole body dose (mean effective dose to a critical group) from all sources and all pathways, except that from natural background radiation, medical applications, and radon.

NATIONAL REQUIREMENTS AND IMPLICATIONS FOR EXTERNAL DOSE LIMITS

Current and proposed regulations from a few countries are summarized in Table 1. While most national regulations are designed to be consistent with ICRP-26 or ICRP-60, the specific requirements may vary from country to country. For example, the limits may be defined in terms of a maximally exposed individual, rather than the mean dose to a critical group, or there may be additional separate limits for gaseous or liquid effluents. Perhaps the aspect most open to local interpretation is how to allow for the potential exposure from more than one source. ICRP 26 states that consideration must be given to the possibility that some individuals may belong to more than one critical group. ICRP 60 also notes the potential need for source-specific constraints, but does not make quantitative recommendations on this subject. In contrast, the U. S. National Council on Radiation Protection and Measurements (NCRP) recommends a single source limit of 0.25 mSv to the maximally exposed individual, unless an assessment is performed to ensure that the dose from all manmade sources does not exceed 1 mSv (3).

The approach used in the Netherlands is to assume that any individual has the potential to be exposed to 10 sources (4), so that the dose limit from a single source is one tenth of the recommended annual dose. In their current regulations, the Netherlands goes further than the ICRP in setting the annual limit at 400 μSv, with a 40 μSv limit at the fence line of any single facility where the public has access. However, the regulations do allow for higher doses if the site is not in an occupied area, and there is a proposal to increase the annual limit to 1 mSv, with the corresponding limit per source of 100 μSv.

In the U. S. there are three government agencies that have oversight over nuclear and radiological facilities. The Environmental Protection Agency (EPA) is responsible for establishing generally applicable standards for the protection of the public from radioactive material. The Nuclear Regulatory Commission (NRC) is responsible for

Table 1. Summary of National Regulations on Dose to the General Public.

Country (document referenced)	Annual Limits on Total Effective Dose (mSv)	Additional Specifications
United States	5	0.25 mSv[d] U fuel cycle, 0.10 mSv airborne
EPA 26FR9057; 40CFR190&61 59FR66414	1[a,d]	effluents
NRC 10CFR20.1301	1	0.50 mSv[d] external component
59FR43200[a]		0.15 mSv[a] decommissioned sites
DOE 10CFR834[a]	1[a]	0.30 mSv[a,d] for single site
Netherlands	0.4	0.04 mSv for single facility
ORS[b]	1[a]	0.10 mSv[a]
Poland	1	____
National Atomic Energy Agency March 31, 1988		
Denmark	1	____
National Board of Health Order No. 821 Dec. 7, 1990		
Germany[c]	0.3[d]	____
England	5	____
Ionizing Radiation Regulations, 1985, Schedule 1		
Spain	5	____
Royal Decree 53/1992		
France	5	Limits on intake of specific radionuclides
Journal Office de la Republique Francaise, No. 88-521 (Article 4) May 6, 1988		

[a]Proposed
[b]Omgaan met risico's van straling, Tweede Kamer 1989 Kamer 1989-1990, 21 483 nr 2; SDU The Hague 1990
[c]Allgemeine Verwaltungsvorschrift δ 45 Strahlensschutzverordung, Ermittlung der Strahlenexposition durch die Ableitung radioaktiver Stoffe aus kerntechnishen Anlagen und Einrichtungen » of 21 February, 1990.
[d]Dose to maximally exposed individual

licensing commercial nuclear power reactors; fuel cycle facilities; medical, academic, and industrial uses of nuclear materials; and the disposal of nuclear materials and waste. The Department of Energy's (DOE) responsibility currently includes its uranium separation facilities, nuclear production and research reactors, nuclear weapons assembly and disassembly facilities, tritium recovery facilities, nuclear materials storage vaults, and high energy particle accelerators. Both the NRC and the DOE set specific restrictions on facilities under their jurisdiction to ensure that they meet the EPA's general requirements for protection of the public. The DOE is moving towards codifying its requirements, previously specified as internal DOE orders, which means that they will be enforceable through civil penalties and allow for greater public review, as is the case for NRC and EPA requirements.

The current EPA regulations (5) set the annual whole body dose limit at 5 mSv, but proposed new guidance (6) would change this to 1 mSv y^{-1} and would recommend that additional source-specific limits be established to "take into account the present and future potential for doses from other sources". For example, an earlier EPA regulation (7) set such a source-specific limit for the "uranium fuel cycle" at 0.25 Sv annual whole body dose, and another (8) limits the dose from airborne emissions from DOE facilities to 0.10 mSv y^{-1} to the general public.

The proposed DOE regulations (9) would set the annual whole body dose limit at 1 mSv, and specify that the potential of exposure from other sources need be taken into consideration if the dose to the public from the DOE site exceeds 0.30 mSv. The current NRC regulations (10) limit the dose to the public to 1 mSv y^{-1}. They also state that compliance may be demonstrated by showing that the external dose for an individual continuously present in an unrestricted area would be <0.5 mSv in a year if other requirements for liquid and gaseous effluents are met.

In the U. S., a related topic of increasing importance is the assessment of dose to the public from facilities that have been decommissioned in order to verify that the site can be released for unrestricted use. Currently, about 20 commercial nuclear power reactors and 130 DOE sites are in the process of being decommissioned. As some of the larger facilities reach the end of their useful lives in coming years, the clean up issues are expected to become more complex with an estimated 2000 sites eventually requiring decommissioning. The NRC is in the process of revising its regulations to include explicit criteria for decommissioning. Their proposed criteria specify that radioactivity from decommissioned sites be as low as reasonably achievable below the level that would result in a 0.15 mSv y^{-1} dose from all pathways to the average individual in a critical group (11). The intention of the NRC is to provide a margin of safety below the annual 1 mSv limit to allow for potential exposure from other sources and the fact that there are no controls once a site is released for unrestricted use.

Most of these examples of current and proposed regulations on public dose do not explicitly state a separate limit on the external component, so what is acceptable in a particular situation will depend on the magnitude of the internal component. (An exception being the NRC limit of 0.5 mSv y^{-1}). In any case, the measurement of an

external dose on the order of 1 mSv y^{-1} (about 114 nSv h^{-1}) or less from manmade sources must include consideration of the natural background radiation field which is on the same order.

REVIEW OF EXTERNAL DOSE FROM NATURAL BACKGROUND RADIATION

Spatial Variations

The background radiation field is due to terrestrial and cosmic sources. Photons and beta rays from primordial radionuclides in the soil, mainly ^{40}K and the ^{238}U and ^{232}Th decay chains, make up the significant terrestrial component. Due to the low penetrating power of beta rays, they are not as significant as gamma rays for dose from natural background. Because of weapons tests in the 1950s and 60s, a small contribution from ^{137}Cs may now be considered part of the radiation background. The terrestrial component varies with geography, depending on the relative concentrations of these radionuclides. The external dose equivalent from terrestrial sources can typically range from 0.1 mSv y^{-1} to 1.4 mSv or more per year depending on the local geography. Some localized areas have been identified with higher levels. The worldwide average annual effective dose is estimated to be 0.46 mSv from terrestrial sources (12).

At ground level, the cosmic component mainly consists of high energy muons, photons, and electrons. The cosmic dose at any given location depends on the altitude, about 0.3 mSv y^{-1} at sea level and about twice that at elevations of 1.6 km. There are also smaller variations on the order of a few percent with latitude because of effects of the earth's magnetic field; the dose is about 10% lower near the equator than near the poles.

Temporal Variations

Measurement of a potential manmade dose component is complicated because the natural background radiation dose levels are not constant in time. Figure 1 illustrates some of the well-known natural variations in background radiation that are associated with diurnal and precipitation effects. (The original measurements were of exposure. For this report the exposure data was converted to absorbed dose in air (1 R/8.7 mGy) and the UNSCEAR recommended factor for environmental radiation was used to convert to effective dose (0.7 Sv Gy^{-1}).) The diurnal effects are related to temperature changes and the accompanying atmospheric turbulence. Radon gas exhaled from soil during the night stays near the surface of the earth while the air there is relatively cold, with the gamma emitting progeny causing an increase in the radiation background level during this time. As the air grows warmer during the day, vertical diffusion reduces the radon concentration and the ground level radiation decreases. Diurnal effects can be on the order of a few percent to as much as 10% where the radon exhalation rate is very high.

Precipitation also plays a major role in natural variations in background radiation. For example, rain or snow can scavenge airborne radon progeny causing an increase in radiation levels by up to a factor of two (or in rare cases three) for several hours. Subsequently wet ground or snow deposited on the soil surface attenuates the terrestrial component causing radiation levels to drop below the previous baseline after the precipitation stops. Other possible natural variations are related to seasonal influences on the exhalation rate of radon from the ground, such as frozen soil allowing less of the gas to escape.

The cosmic-ray component does not vary as much as the terrestrial component on a day to day basis, but the 11-year solar cycle can result in variations on the order of 10% from the average value. On occasion, solar flares have been observed to produce measurable increases up to a factor of 3 in the ionizing component at sea level.

QUANTITIES TO BE MEASURED

For environmental monitoring, the use of the operational quantities "ambient dose equivalent" and "directional dose equivalent" is recommended in ICRP 60 as reasonable approximations to the effective dose and the equivalent dose in the skin (see ICRU 51 for precise definitions). For field measurements it would seem logical to use measurable physical quantities rather than dose equivalents, thereby avoiding complications that arise with ongoing developments in radiation biology. However, since the purpose of such measurements is to verify compliance with dose limits, they ultimately have to be related to the quantities specified in regulations. To convert between physical and dose equivalent quantities requires knowledge of the nature and energy of incident radiation. Since the radiation field may not be known, the ideal detector would have a flat response with energy for the quantity of interest.

A typical photon energy distribution from terrestrial sources based on calculations for reasonable soil and source distribution parameters (13) is shown in Figure 2. For comparison, the photon fluence, air kerma rate, and ambient dose equivalent quantities are shown. It is important to note that though most of the fluence (~70%) is due to gammas of energy <500 keV, these low energy photon components do not contribute as much to the air kerma or ambient dose equivalent as the higher energy photons. Most of the total ambient dose equivalent (~70%) is from

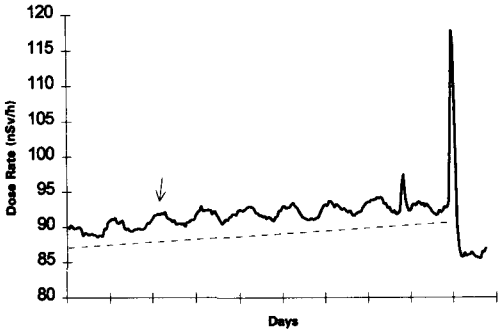

Figure 1. Short-term variations in natural background external dose rate due to temperature and precipitation effects, as measured with a PIC. Arrow indicates changes related to daily temperature effects on radon. Dashed line indicates gradual increase in dose rate due to soil drying out after rainfall. Large peak corresponds to rain scavenging of atmospheric radon progeny.

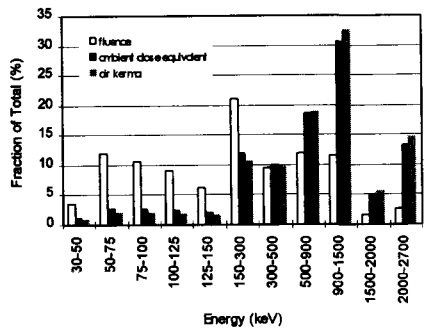

Figure 2. Approximate energy distribution of typical background terrestrial gamma rays at height of 1 m. Derived from data in Beck (13) using ICRU 47 (1992) factors to convert from fluence to ambient dose equivalent. Average factor for each energy range was used.

photons with energies >500 keV. Comparison of the overall energy distribution in terms of the physical quantity air kerma and the operational quantity ambient dose equivalent shows there is very little difference. This is because above about 30 keV the energy response of air and the ICRU sphere is essentially the same. Photons below about 30 keV are weakly penetrating and do not make a significant contribution in environmental radiation monitoring.

INSTRUMENTS FOR DIRECT MEASUREMENTS OF EXTERNAL DOSE

Pressurized Ionization Chambers, Geiger Mueller Tubes

Figure 1 shows data obtained with a pressurized ionization chamber (PIC). Similar data can be obtained with Geiger Mueller (GM) tubes. Both types of instruments are good for providing a detailed time record of the dose rate. Current models are sensitive to changes on the order of 0.6 nSv h^{-1} and typically allow measurements every minute or less and can store over 1 month of data unattended. It is important that they be appropriately calibrated for the environmental spectrum encountered, with consideration given to the relative sensitivity to cosmic and terrestrial components.

Spectrometers

Sodium iodide and germanium detectors (which have superior energy resolution) record the energy of incident gamma radiation and can therefore be used to identify sources. With appropriate calibration, they can also be used to quantify the concentrations of radionuclides to provide *in situ* soil analysis and to distinguish the terrestrial component of the dose (14). This technique has been used for the study of global fallout, and for assessment after the Chernobyl accident. Because of a spectrometer's high cost and more extensive operating requirements (higher voltage biased supply, amplifier, liquid nitrogen in the case of Ge detectors), they are usually not left at a site unattended for long periods for routine environmental monitoring. But for site characterization and final status surveys for decommissioning or where dose limits are very low, the technique of *in situ* gamma spectrometry can be particularly useful, as explained below.

To assist in the interpretation of environmental radiation measurements, it is often useful to combine results from more than one instrument. For example, the increase in radiation levels due to precipitation, as illustrated in Figure 2, could look like an accidental release. Denmark's "early warning system" (15) established after the accident at Chernobyl uses a sodium iodide detector to assess fluctuations of 10% or more in PIC measurements of the ambient background level in order to separate natural events from possible manmade releases. Likewise, precipitation data from a weather station can be applied for the same purpose. As another example, the terrestrial component of the dose rate determined with a spectrometer can be compared to the reading of a PIC or GM detector as a quality assurance check on both instruments, or to correct for the PIC or GM detectors' differing sensitivities to the cosmic and terrestrial components.

Passive Environmental Dosimeters

Passive detectors provide an integrated measure of the total dose. While this information is limited in comparison to that provided by PICs, GM detectors and spectrometers, it is often adequate for environmental monitoring

purposes, as will be illustrated below. While film and electrets are occasionally applied, the most widely used integrating dosimeters for environmental monitoring are thermoluminescent dosimeters (TLDs). Though the laboratory instruments used to process a TLD can be costly, the TLDs themselves are inexpensive, reusable, and rugged, and can therefore be deployed in locations that are remote or subject to potential loss from such factors as vandalism or environmental insults.

To monitor the external dose to the public, nuclear installations may use a few PICs or GM detectors at the fence line, and many more TLDs at the fence line and beyond to provide coverage off site. There are presently over 17,000 sites on earth being monitored with TLDs (16). In the hours immediately following the accident in the U. S. at the Three Mile Island nuclear power plant in 1979, TLDs were the only environmental monitors in the field and they provided significant data about dose since plant vent monitors had reached their upper range limits (17). In the U. S. (partly as a result of this) the NRC established its own network of TLD sites around all power plants, in addition to those used by the private utilities. The NRC's network consists of 13 to 48 monitoring stations around 75 facilities, and typically covers standard windrose sectors circling the site boundary, as well as 3 km from the site, with additional stations 8 km or more from the site, including major population areas (18).

Recently developed TL materials provide increased sensitivity, and have been shown to be capable of measuring daily background doses, i.e., on the order of 2 μSv (19). Some TLD materials such as LiF, $Li_2B_4O_7$, and BeO are nearly tissue equivalent. The differing spectral sensitivity of other materials and the use of additional filters has been exploited to perform crude spectrometry (20). This can be especially useful to monitor for such things as a 6 MeV gamma ray from ^{16}N produced in nuclear power plant, or low energy photons from ^{133}Xe releases.

DISTINGUISHING DOSE ABOVE BACKGROUND LEVELS

While all of the detectors described above are sensitive enough to measure absolute dose levels on the order of the current and proposed limits for public exposure, distinguishing a potential manmade component from the naturally varying background is a more complex problem. The relevant question to address is: Can the available instruments provide data on external dose that would be useful to measure compliance with public dose limits?

To address this question it is useful to look at data from the long-term monitoring of natural background radiation. The Environmental Measurements Laboratory (EML) has been monitoring radiation exposure (and other conditions) at a rural location with no local sources of pollution and undisturbed soil (21). Figure 3 shows PIC and TLD data from 14 years of simultaneous monitoring. The PIC data were obtained with EML designed ionization chambers. The TLD data are from dosimeters consisting of 15 ^{7}LiF:Mg,Ti phosphors ("chips") deployed within about 1 m of the PIC during each calender month (22). While the TLDs measure the total dose, a mean dose rate is obtained by dividing by the field deployment time. This can then be compared with the average dose rate measured by the PICs during the same time period. For the PIC data, each point represents the average of thousands of readings, and the associated statistical uncertainty is small (<<1%). For TLD measurements, the uncertainty tends to be much higher (~5-10%) due variations between chips and possible changes in reader conditions, as well as the expected poorer statistics resulting from averaging over a small number of chips per dosimeter. This is the probable reason for the somewhat larger variations observed in the TLD results.

Figure 3 shows good agreement between the TLD and PIC data, and it illustrates that while the monthly average background dose rate can vary by as much as almost 40%, the yearly average tends to be more stable, varying only a few percent from year to year. The largest monthly variations are low dose rates corresponding to winter months with greater snow cover. The overall PIC average is 87 nSv h^{-1}, and is marked with a solid line in the figure. The dashed lines show the hourly dose rate levels that would correspond to an additional dose, above the PIC average, equal to that of some of the annual public dose limits shown in Table 1.

The dashed lines show that an additional external dose corresponding to an annual dose of 1 mSv (i.e., 114 nSv h^{-1}) above the average background dose would be clearly beyond the range of natural measurement variations, and, therefore, easily detectable. The same is true of the current NRC annual external dose limit of 0.50 mSv. The single source limits being used by EPA (0.25 mSv) and proposed by DOE (0.30 mSv, not shown in Figure 3), are both well above the maximum data variation also. However, the 0.10 mSv limit being considered in the Netherlands and applied to airborne emissions in the U. S. is approaching the measurement variation, as is the 0.15 mSv limit currently proposed by the NRC for decommissioned facilities. As these limits should include internal dose components, the type of TLD and PIC measurements shown here may not be sufficient and additional information would likely be required.

In the case of facilities undergoing decommissioning, it is usually known what contaminant radionuclides could be present. The NRC is in the process of determining specific soil concentrations for various nuclides that would result in a total effective dose below the proposed 0.15 mSv limit ("release criteria") (23). These concentrations are calculated using a dose modeling code (24) that considers various source transport and exposure pathways and

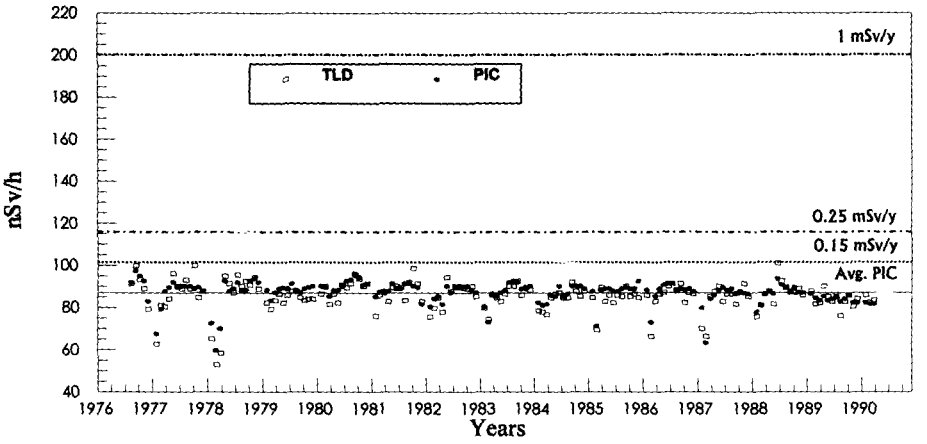

Figure 3. Fourteen years of monthly TLD and PIC environmental monitoring data at a rural site in the U. S. with no local sources of pollution.

land-use scenarios. To demonstrate compliance, it may be sufficient to show that residual soil contamination does not exceed the release criteria. It is interesting to note that while an additional external dose on the order of 0.15 mSv y^{-1} (17 nSv h^{-1}) may be difficult to detect in monthly averages, the corresponding allowable concentration of a specific nuclide can be measurable by the *in situ* gamma spectrometry technique. For example, the release criteria estimated in the current working draft for ^{238}U in soil is about 0.3 Bq g^{-1} (7.8 pCi g^{-1}), or more than 10 times the natural background concentration of uranium (23). For another example, the release criterion for ^{60}Co is about 0.1 Bq g^{-1} (3 pCi g^{-1}).

However, it must be emphasized that the modeling code is being refined and the release criteria have not been finally determined. The calculated value depends on the input parameters used for such things as soil permeability, and the water situation for the modeled site. Such factors are more important for nuclides with a significant internal dose pathway, and the final values for the release criteria could be much different. For these reasons, it would be useful to make measurements of the external dose to verify the assumptions made in the modeling code. It is also worth noting that because the proposed limit applies to the mean dose to a critical group rather than the maximally exposed individual, depending on how the occupancy and shielding factors are determined, an external dose at a field site greater than 0.15 mSv may be allowed. Also, any external dose from a decommissioned site should be static, and a determination would be made based on comparison with a reference site that is not contaminated, in which cases TLD or averaged PIC measurements may yet be adequate for some cases.

Many environmental monitoring programs presently use a quarterly monitoring cycle, which should result in a lower range of variations than those seen in Figure 3. Furthermore, if regulations are stated in terms of an annual dose limit, dose rate measurements near the limiting value would have to be present for more than one quarter to result in a dose in excess of the limit. A dose large enough to exceed annual limits in one quarter would be well beyond the range of data variations for most of the limits now being considered. However, the Netherlands limit of 0.04 mSv would likely be within the natural variations and not detectable in the monthly averages. Real time PIC monitoring may be required in this case. Depending on how other regulations specify what fraction is allowed for internal and external components, the measurability could be difficult as well. It is clear that any monitoring program would have to establish natural levels for a specific site. Where possible, this would ideally be achieved through a few years of monitoring before the facility becomes operational. Where this is not possible, it may be necessary to establish background levels at some distance from a site. Alternatively, gamma spectrometry could be applied to determine if there is any facility contribution.

INTERNATIONAL INTERCOMPARISONS OF ENVIRONMENTAL MONITORING INSTRUMENTS

The European community has sponsored several international intercomparisons of environmental monitoring instruments beginning in 1984. The most recent intercomparison involved eight countries and was held at the PTB in Germany and Riso National Laboratory in Denmark in 1994 (25). In an earlier study, PIC, GM, proportional counter and TLDs were used to measure the routine discharges of a nuclear power plant over a 2-month period (26).

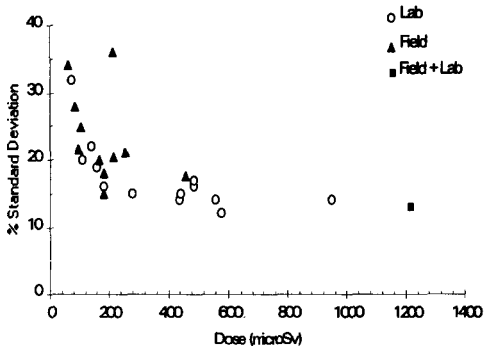

Figure 4. Relationship of dispersion of participants' results and the delivered dose from the International Intercomparisons of Environmental Dosimeters. Most of the participants used TLDs.

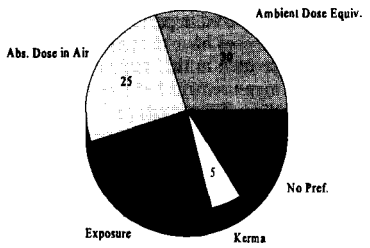

Figure 5. Results of survey of preferred quantity to use for environmental monitoring from 10th International Intercomparison of Environmental Dosimeters (28). Number of participants preferring each quantity is shown, out of 100.

While the active instruments provided an indication of the time of the releases and tracked the power level of the reactor, only the TLDs provided a good measure of the total dose without corrections that required independent knowledge of the energy spectrum of the radiation field (27). This is a result of the LiF TLDs' appropriate energy response for such monitoring. Because of the variations in their energy response, the PIC, GM and proportional counter overestimated the dose by 10%, 40%, and 50%, respectively.

This illustrates that passive integrating dosimeters can be particularly useful for assessing the external dose to the public to check compliance with annual dose limits. While they do not have the time resolution to identify transient dose increases, this information is not required for assessing compliance with annual dose limits. Also, because they are completely passive there is no associated "down" time that is inherent in any type of active instrument which may occasionally not be serviced on schedule because of such things as weather conditions. Active detectors like PICs and GM detectors are most useful to monitor in real time or for short-term studies or for model validation. TLDs are useful for widespread coverage off-site.

The U. S. Department of Energy (EML) has sponsored the "International Intercomparison of Environmental Dosimeters" since 1972. This program is open to any type of integrating dosimeter, and typically includes over 100 participants from 25-30 countries. It provides a measure of the state of the art in passive environmental dosimetry. For example, in the 10th International Intercomparison of Environmental Dosimeters (28) in 1993 it was found that 90% of the participants were within 30% of the delivered dose. Figure 4 shows the dispersion of participants' results plotted against delivered dose from all the intercomparisons. For lower doses, there is greater dispersion, but still most participants were within the performance criteria recommended by the American National Standards Institute (ANSI)(29). The 10th intercomparison included a survey where participants were asked to select their preference from a list of quantities. The results are shown in Figure 5. Thirty percent of the participants selected ambient dose equivalent, 54% selected one of the physical quantities of exposure, absorbed dose in air, or air kerma, and 16% indicated no preference.

NATIONAL AND INTERNATIONAL ENVIRONMENTAL DOSIMETRY STANDARDS

Despite their widespread application in measuring dose to the public, there are presently no testing or accreditation programs for environmental dosimetry providers. In contrast, several national programs exist for the testing of personnel dosimetry services. There may not be a need for such testing unless there are cases where public dose assessments are challenged. In such cases it may be difficult to defend doses derived completely from calculations. In anticipation of such a need the ANSI has developed a draft standard on performance testing of environmental dosimeters (30) that is analogous to the one currently used for the occupational dosimetry National Voluntary Laboratory Accreditation Program (NVLAP) in the U. S. (31). In addition to this, the current standard for environmental thermoluminescence dosimetry, ANSI-N545 is being revised as ANSI-N13.37. The Netherlands is in the process of establishing requirements for environmental dosimeters in its standard NVM5648 (32). There

; a more general international standard already in place that covers this subject as well, IEC 1066. Particularly for nvironmental monitoring where the dose limits are much lower than for occupational settings, one of the most elevant issues to be covered in performance standards is requirements for the lower limit of detection for dosimetry ystems. Many researches have analyzed the methods for defining this quantity, which is not straight forward (33).

Regulations are written in the interest of public health and they do not necessarily take into consideration the abilities of current measurement technologies. National and international standards, on the other hand, may focus on physical properties of detectors and are not necessarily linked to regulations. It would be most useful if both regulations and performance standards could converge on practical methods to meet the needs in dose assessment.

CONCLUSIONS

Many countries have regulations that limit the radiation dose to members of the general public. Most are consistent with the ICRP and are moving towards a 1 mSv annual dose limit, with some specifying additional lower limits to allow for exposure from multiple sources. Such limits currently being considered are in the range of 0.10 mSv to 0.30 mSv. Limits apply to all pathways, and, therefore, will require dose modeling. Measurements of the external dose could help to verify the assumptions made in dose modeling codes. Based on long-term measurements of the natural background radiation, assessing an additional external dose on the order of 0.15 mSv or more would be possible using currently available technology. However, interpretations of any environmental monitoring data require knowledge of the natural background radiation field and its variations. Results from international intercomparisons demonstrate that passive dosimeters, such as TLDs, are appropriate devices for measuring the external dose off-site. National standards being developed for such devices should address the needs of regulations on dose limits. Active instruments such as PICs and GM detectors are useful for real-time monitoring or spot measurements, and provide more information than may be needed to demonstrate compliance with public dose limits. Limits of 0.15 mSv or less may require more extensive measurements with spectrometers and calculations of derived limits for soil concentrations, similar to those being presently developed in the U. S. for decommissioned sites.

ACKNOWLEDGMENTS-- The author acknowledges the contributions of Pepin Carolan, Marcel Reginatto, Kevin Miller, Wayne Lowder and Harold Beck to the content of this paper. Thanks to Matthew Williamson and Andre Bouville for information on regulations, Jenny May for assistance with graphics, Nancy Chieco for editorial contributions, and Levi Reyes for producing the final manuscript.

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AREA MONITORING

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INTRODUCTION

The exposure of individuals to ionizing radiation is normally subject to legal controls which specify limits, and restriction, of doses that both occupational workers and members of the public are exposed to.

The dose limits recommended by the International Commission on Radiological Protection (ICRP) (1), and adopted by many countries are specified in terms of body dose equivalents, sometimes referred to as limiting quantities. However these quantities are not measurable since they are defined as average doses in organs and tissues of the human body. Thus for the same physical radiation field the dose delivered to an individual will not have a fixed value but will depend upon each individual's sex, age and physical construction as well as upon their orientation in the field.

Ionizing radiation, also, cannot be detected by a person's physical senses of sight, smell, hearing and touch. Thus to control the levels of radiation to which individuals are exposed, special operational quantities are used and quantitative measurements have to be made using personal dosemeters or area monitoring equipment. This report describes the operational quantities that are being increasingly used worldwide for area monitoring. It provides information on the international standards that are available on the performance requirements for monitoring equipment designed to measure these operational quantities. It also reviews the recent international standards that have been published on the calibration and type testing of area monitors and gives an example of legislative requirements on area monitoring equipment.

OPERATIONAL QUANTITIES

The International Atomic Energy Agency (IAEA) has incorporated into its Basic Safety Standards (2) the ICRP recommendations on a revised system of dose limitation (1). The dose limitation system is based on the equivalent doses in various organs or tissues, H_t, of an individual and the weighted sum of the equivalent doses in some tissues and organs, the effective dose, E. These limiting quantities are essentially unmeasurable so they are estimated through the use of operational quantities that can be measured in the radiation exposure environment.

The operational quantities are defined by the International Commission on Radiation Units and Measurements (ICRU) under receptor-present conditions. For area monitoring the quantities ambient dose equivalent, H*(10), and the directional dose equivalent, H'(0.07) are defined for the ICRU tissue equivalent sphere (3). For personal monitoring the personal dose equivalent, Hp(d) is defined for the human body (4).

For the purpose of demonstrating compliance with the dose limit, ICRP and the IAEA recommend that the following criteria be met:

$$\frac{H_p(d)}{DL} + \sum_j \frac{I_{j,ing}}{I_{j,ing,L}} + \sum_j \frac{I_{j,inh}}{I_{j,inh,L}} \leq 1$$

where DL is the relevant limit for the year of the effective dose, $H_p(d)$ is the personal dose equivalent from external penetrating radiation during the year; $I_{j,ing}$ and $I_{j,inh}$ respectively are the intake via ingestion or inhalation of radionuclide j during the same period, and $I_{j,ing,L}$ and $I_{j,inh,L}$ respectively are the annual limits on intake (ALI) via ingestion or via inhalation of radionuclide j.

Radiations are also characterized as either "weakly" or "strongly" penetrating depending on which dose equivalent is closer to its limiting value. For strongly penetrating radiation, namely from photons with energies above about 15keV, neutrons and for electrons of energy above about 5Mev, the effective dose is

appropriate and for weakly penetrating radiation, namely from photons with energies below about 15keV and electrons with energies below about 5Mev, either the dose equivalent in the lens of the eye or that in the skin is relevant.

A summary of the operational quantities distinguished according to the radiation penetration is given in Table 1; the lower part of the table gives the measurements used from which estimations are made of inhalation and ingestion doses.

Table 1. Summary of the operational quantities

External Radiation	Limiting Quantity	Operational Quantity for	
		Area Monitoring	Individual Monitoring
Strongly Penetrating Radiation	Effective Dose	$H^*(10)$	$H_p(10)$
Weakly Penetrating Radiation	Skin Dose	$H'(0.07,\Omega)$	$H_p(0.07)$
	Dose to the Lens of the Eye	$H'(3,\Omega)$	$H_p(3)$
Contamination		Area Monitoring Measurement	
Airborne	$I_{j,inh,1}$(ALI)	Airborne Contamination Units (Bq m^{-3})	
Surface	$I_{j,ing,1}$(ALI)	Surface Contamination Units (Bq m^{-2})	

EXTERNAL DOSE AREA MONITORING

For external strongly penetrating radiation the effective dose is estimated by measuring the personal dose equivalent $H_p(10)$. However this is measured using a personal dosemeter worn on the trunk of an individual and for most dosemeters the dose can only be evaluated retrospectively after the period of wearing, typically one month. Members of the public are not issued or wear personal dosemeters so their doses have to be assessed from measurements made with area monitors.

For photon radiation from 10 keV to 1.5 MeV at normal incidence to the body the ambient dose equivalent is within 10% of the personal dose equivalent and for other geometries the ambient dose equivalent is always the greater.

For all irradiation geometries for neutrons, except AP and PA, and for all energies up to 10 MeV, area monitoring of the quantity ambient dose equivalent, $H^*(10)$, provides a conservative estimate of the effective dose equivalent (5), for photon radiation see Figure 1. Such measurements can therefore safely be used to assess and optimize the doses of individuals working within controlled and supervised areas.

To control the doses from weakly penetrating radiation, area monitors with thin entrance windows should be used to measure the quantity directional dose equivalent, $H'(0.07)$. In practice, the exposure of the skin is highest for normal radiation incidence. The area dosemeter should therefore be rotated and the maximum reading taken for estimating the skin dose.

AREA MONITORING OF CONTAMINATION

Measurement of the internal doses arising from the inhalation and ingestion of radioactive materials can only be estimated following their retention inside the body. Area monitoring of the surface contamination and airborne contamination do, however, provide information which can be used for preventative radiation protection purposes, such as the delineation of controlled areas, restriction of access to such areas and the

issuing of protective clothing, respirators and breathing air apparatus. Measurements made at the exit to controlled areas also ensure that radiation workers do not spread contamination from themselves or their equipment to non controlled areas.

Monitors used to measure surface contamination are calibrated in terms of the calibration source's certificated emission rate per unit area, S_A. The monitors calibration factor $CF_E$is given by:

$$CF_E = \frac{S_A}{M-M_b}$$

where M is the monitor's count rate (units s^{-1}) when exposed to the calibration source at a specified distance and M_b is the monitor background count rate. The calibration distance, from the surface of the source to the detector window is small and is usually chosen to be equal to, or very close to, the distance that will be used for subsequent measurements of contaminated surfaces. When the instrument is used to assess the amount of radioactive material on a surface it measures the surface emission rate from the surface and so the measurements have to be interpreted in terms of activity per unit area, Bq cm^{-2}. The surface activity per unit area, As, is estimated by multiplying the observed background subtracted reading, $M_{(e-bg)}$ by the calibration factor, CF_E and by a conversion factor, P, which is the ratio of the surface activity per unit area to the surface emission rate i.e. $As = M_{(e-bg)}. CF_E. P$
and P= (a), (b), (c), (d), (e) where:

(a) The number of particles of a given type produced per decay of radionuclide.

(b) Where a parent/daughter radionuclide is being measured the activity refers to the combined number of particles produced by both the parent and daughter(s) when in equilibrium with each other.

(c) Not all the particles are emitted from the surface into the upper 2π solid angle where the detector is making the measurement. Up to a maximum of half will be emitted downwards into the material on which, or in which, the activity being measured is located. Some particles emitted downwards will be backscattered and emerge into the upper 2π solid angle.

(d) Self-absorption will also reduce the number of particles emitted into the upper 2π solid angle. The magnitude of this factor will depend upon the type of radiation, its energy and the thickness and distribution of the radioactive material on or within the surface being measured.

(e) The uniformity and distribution of the radioactive material on the surface. Whilst the calibration source is constructed to provide a uniform distribution (activity per unit area) over the area of the detector, surfaces being measured are very rarely contaminated uniformly.

Factors (a) and (b) are uniquely characterized by the decay scheme for the particular radionuclide and may therefor be quantified. Factors (c), (d) and (e) are dependent upon the characteristics of the contaminated surface and have to be estimated values. When direct measurements are made of a surface contaminated with a different radionuclide than that used for the calibration monitor, the factor (a) is replaced by another factor (a*).

(a*) A correction factor has to be applied to allow for

1) The difference between the number of particles of a given type produced per decay from the calibration radionuclide and from the radionuclide being measured on the surface; and

2) Difference of the monitor's energy response when the particle energies are different for the calibration source and the surface radionuclide. By careful selection of the calibration radionuclide to have a particle energy close to the particle energy of the radionuclide being measured on the surface this factor can be assumed to be unity.

For example where an ^{129}I source is used to calibrate a monitor that will be used to measure surfaces contaminated with ^{125}I the weighted mean energies are respectively 28.8 keV and 26.04 keV so no correction

r factor 2) is required. However, the total probability per decay for ^{129}I is 0.858 and is 1.598 for ^{125}I, so (a*) ould equal 0.858/1.598 = 0.54.

Where a direct measurement is made of a surface contaminated with a mixture of radio isotopes requently the composition will not be known and the relative amounts will probably be less well known. In he interest of safety a conservative estimate may be made by the use of the highest value of the factor (a*) within the considered mixture.

The product (c), (d), (e) may be assumed to have the following values:

For beta emitting radionuclides with a maximum energy >0.4MeV an approximate value of 2 may be used as a general rule. For beta emitting radionuclides with a maximum energy >0.15 MeV and <0.4 MeV, as well as for alpha emitting radionuclides, a value of 4 may be used. For photon emitting radionuclides an approximate value of 2 may be used. However, it should be noted for the measurement of contamination present in very thin layers, the activity would be overestimated if significant backscatter was present. Conversely, for thicker layers of surface contamination, the activity would be underestimated if there was significant attenuation within the contaminated surface. For the lower maximum energy radionuclides considered in this procedure the effect of air path attenuation could be significant if the surface contamination instrument is used to measure surface contamination at distances different from that used for calibration.

AIRBORNE CONTAMINATION MONITORING

For alpha emitting radionuclides and low energy photon radionuclides the limitations on minimum detectable activities by bioassay measurements, namely in vivo and in vitro analysis, means that such measurements may not be practical and that area monitoring may have to be utilized. Such monitoring will provide warnings that preventative action may be required or that intakes may occur and bioassay measurements may be necessary. In some countries it is permitted to use air monitoring results to assess internal doses.

The committed effective dose per unit intake for workers and members of the public via inhalation and ingestion (Sv Bq^{-1}), for different nuclides, are given in the IAEA Basic Safety Standards (2).

The intake can be estimated from the measured air concentration, multiplying this by the time that the worker breathed at this concentration and dividing by the number of working hours in a year and by a factor which allows for protection if a respirator is worn.

Alternatively to estimating the effective dose the derived intakes may be compared with the respective annual limit on intake (ALI).

PERFORMANCE CHARACTERISTICS OF AREA MONITORS

The International Electrotechnical Commission (IEC) have published standards on the performance and testing of monitoring equipment. Examples of some of these IEC standards are given in Table 2.

Table 2. Examples of IEC standards on radiation monitoring equipment

Publication	Equipment
	Photon and Beta Monitoring Equipment
1018	High range beta and photon dose and dose rate portable instruments for emergency radiation protection purposes.
532	Installed dose rate meters, warning assemblies and monitors for X or gamma radiations of energy between 50 keV and 7 MeV
846	Beta, X and gamma radiation dose equivalent and dose equivalent ratemeters for use in radiation protection
1017-1 1017-2	Portable, transportable or installed X or gamma radiation ratemeters for environmental monitoring - Part 1: Ratemeters; Part 2: Integrating assemblies
	Personal Dosimetry
1283	Direct reading personal dose equivalent and/or dose equivalent rate monitors for the measurement of personal dose equivalents $H_p(10)$ and $H_p(0.07)$ for X, gamma and beta radiations
1066	Thermoluminescence dosimetry systems for personal and environmental monitoring
	Neutron Monitoring Equipment
1005	Portable neutron ambient dose equivalent ratemeters for use in radiation protection
1323	Direct reading personal dose equivalent and/or dose equivalent rate monitors for neutron radiation
	Monitoring of Individual Radioactive Contamination
325	Alpha, beta and alpha-beta contamination metes and monitors
504	Hand and/or foot contamination monitors and warning assemblies
1098	Installed personal surface contamination monitoring assemblies for alpha and beta emitters
	Monitoring of Airborne Radioactive Contamination
579	Radioactive aerosol contamination meters and monitors
710	Radiation protection equipment for the measuring and monitoring of airborne tritium
1171	Monitoring equipment - Atmospheric radioactive iodines in the environment

The IEC standards specify the requirements and tests that shall be undertaken to determine the radiation, electrical, mechanical and environmental performance of radiation protection instrumentation. As an example of the requirements of the IEC standards Table 3 gives details of the radiation specifications for

the equipment to measure ambient dose equivalent, Standard 846.

Table 3. Radiation characteristics of ambient dose equivalent (rate) meters

Characteristic under test of influence quantity	Range of value(s) of influence quantity	Limits of variation
Relative intrinsic error - X and gamma radiation	Effective range of measurement	± 15% [1]
X and gamma radiation energy	10 keV to 30 keV 30 keV to 0.2 MeV 0.2 MeV to 1.5 MeV 1.5 MeV to 10 MeV	To be stated by the manufacturer ± 35% [2] ±15% [2] ±35% [2]
Beta radiation energy	Beta radiation up to E_{max} of 4 MeV	±50%
Angle of incidence - X and gamma radiation	0° to 90° from reference direction 90° to 180° from reference direction	±25% To be stated by the manufacturer
Overload	100 times the range maximum for range maxima up to and including 0.1 Svh^{-1} 10 times the range maximum, or 10 Svh^{-1}, whichever is the greater, for range maxima more than 0.1 Svh^{-1}	Indication to be off-scale on the high side or assembly to indicate overload (for 5 min)
Effects of neutron radiation		Response to be stated by the manufacturer

[1] This error is additional to the uncertainty in the determination of the conventionally true dose equivalent (rate).

[2] Limit of variation from indication under reference conditions.

Many countries have legislative requirements on the calibration and use of radiation monitoring equipment. For example, in the UK the Ionising Radiations Regulations 1985 requires that every employer who undertakes work with ionizing radiation shall ensure that levels of ionizing radiation are adequately monitored for each controlled area or supervised area that he has designated. He is required to provide equipment which is suitable for carrying out such monitoring and this equipment has to have had its performance established by tests before it is taken into use for the first time, be properly maintained and be thoroughly examined and tested at least once every 14 months. The employer also has to make suitable records of the results of the monitoring and of his testing, calibration and maintenance of the monitoring equipment. The legislation, as well as good metrological practice, require that the measurements of the radiation characteristics of an instrument made as part of the type, acceptance and routine tests, as well as the calibration, should all be traceable to an appropriate national standard.

The IEC standards give a very good basis upon which to judge its performance and whether it is suitable for carrying out the monitoring task. The IEC testing is based upon the International Standards Organization Standards on reference radiations for calibrating and type testing of radiation monitoring

equipment, one example of which is the photon reference standard (7). These ISO standards give the conversion co-efficients that shall be used to convert the physical quantitives used to characterize the dosimetric properties of the reference calibration fields, e.g. Air Kerma or Fluence, to the quantities used by the area monitoring equipment, e.g. ambient and directional dose equivalent.

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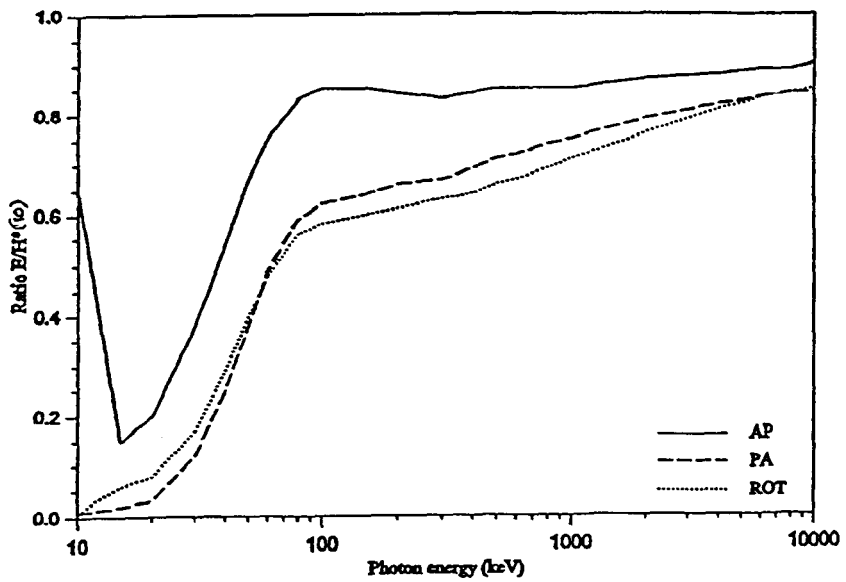

Figure 1. The ratio of $E/H^*(10)$, for the AP, PA and ROT irradiation geometries, as a function of photon energy.

UV AND SKIN: THE BIOLOGICAL EFFECTS OF UVA AND UVB

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INTRODUCTION

Almost all of the photobiological reactions that occur in the skin are induced by radiations between 290 and 380 nm. These radiations correspond to the wavelengths from the solar emission which are received at the earth surface after absorption by the terrestrial atmosphere. However, from artificial sources, used at work or in domestic environment, the complete spectrum of UV may present a potential hazard.[1]

After considering the optical properties of the human skin, acute effects of UV on the skin will be considered, followed by the potential phototoxic and photoallergic consequences of UV exposures and finally, the adverse cutaneous reaction to sunlight.[2] Protective measures will be considered.

THE OPTICAL PROPERTIES OF THE SKIN

To establish a photobiological reaction, three components are necessary: the biological system, the radiation and a radiation absorber in the biological system. All the components of the tegument are potentially able to react with the incident UV and, as a consequence, a protective filter for the constituents located below.

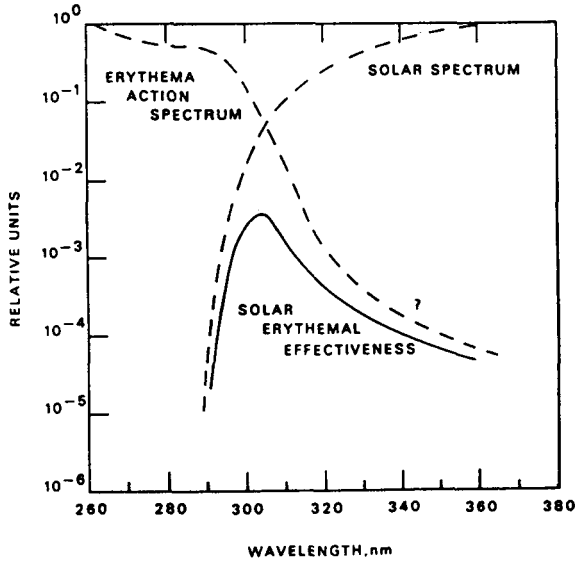

Figure 1. The human solar erythemal effectiveness is obtained by the convolution of the human skin erythema action spectrum and the solar spectrum. It clearly presents a maximum at 307 nm which is the most efficient wavelength to induce sunburn. For artificial sources the erythemal effectiveness shape of the curve may be completely different. The same holds for chemicals with different absorption spectrum.

Skin reflects a large amount of incident visible and near infrared radiation and to a lesser degree, some UVA. Scattering and absorption play a greater role in the attenuation of radiation than reflection. Melanins and melanosomes, nucleic acids, proteins, lipids, urocanic acid, cholesterol, and histidine do limit the penetration of UV.

UVC are absorbed by the stratum corneum and the upper layers of the stratum Malpighi. UVC has only indirect impact on the living layer of the epidermis (melanocytes and keratinocytes), but are able to generate the cytokin production responsible for erythema and to alter the immune function of Langerhans cells.

UVB has indirect impact on the full thickness of the epidermis, 10 to 15% of the longer UVB are reaching the papilla (the upper part of the dermis).

UVA are absorbed at 50% by the epidermis, the rest is able to penetrate the dermis up to 2 mm depth. Because of the importance of the shorter wavelength UV, in the production of the photobiological reaction, the acute effects will be mostly produced at sea level by radiation around 307 nm.

After repeated insults from UVR, and particularly UVB, the thickness of the epidermis is increasing with direct consequences on the UVB absorption, making the basal layer of the epidermis (where keratinocytes, able to divide, are located) out of direct reaction with UVB radiations. Soluble mediators produced by maturing keratinocytes are still efficient.

BIOMOLECULAR EFFECTS OF UVA AND B

The complexity of these reactions are out of the scope of this document. However, it is enough to say that DNA is one of the most important chromophores for both direct and indirect (activated oxygen species mediated) effects of UV. Once a DNA molecule has been altered by either of these processes, the damages have to be repaired by more or less complex pathway. The repair processes may be error-free or error-prone, depending on the severity of the DNA damages. Schematically, UVB are essentially inducing DNA dimers which are, most of the time, error-free repaired. UVA induce DNA strand breaks much less frequently than UVB, but the strand breaks are error-prone lesions. As a consequence, after long term exposures, the genetic code is altered equally by UVA and UVB. UVA-1 (340-400 nm) , considered for long time as inefficient in producing damages, are responsible for only strand breaks, while UVA-2 and UVB are inducing mixture of dimers and strand breaks (the 50/50 ratio being obtained around 320 nm).[3-4]

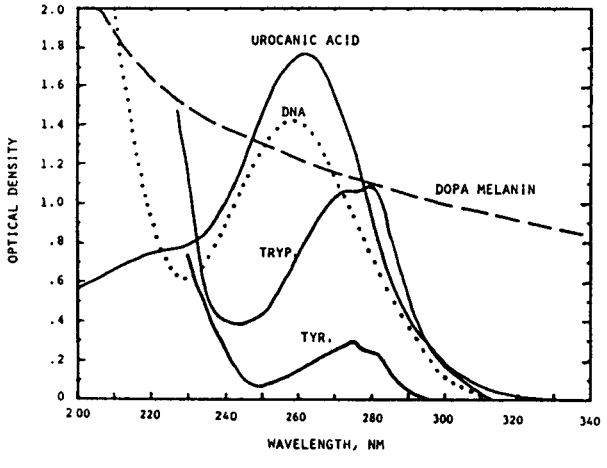

Figure 2. Absorption spectrum for different chromophores of the epidermis. The peaks of absorption for urocanic acid and DNA lie in the UVC and the small tails in the UVB are responsible for most of the deleterious biological effects, when human skin is exposed to sunlight.

The second most action of UV is on lipids which undergo peroxidation through indirect mechanisms. Other chromophores are able to absorb and react with UVB and UVA leading to activated molecules responsible for the production of several cytokins, biological inducers of mitosis, vascular changes, and systemic distant effects. The urocanic acid (maximum absorption at 265 nm) is still absorbing up to 315 nm

and is thought to play some role as an "endogenous sunscreen" of the epidermis. The energy of the photons are transferred to the chemical *cis-trans* isomerization of the urocanic acid.[5]

ACUTE EFFECT OF SUNLIGHT ON THE SKIN

- **Sunburn reaction and minimal erythemal dose.**

The sunburn reaction is the most common adverse effect produced by sunlight. The cutaneous changes induced by the erythemogenic radiation depend on the amount of radiation, the degree and the quality of melanins and the thickness of the stratum corneum. The erythema (reddening of the skin) is a visual aspect of the sunburn response. It is delayed 2-4 hours after the irradiation and peak at 14-20 hours, persisting normally for 72 hours. A severe sunburn is usually followed by an increase of the epidermal thickness and desquamation of dead epidermal cells. A minimal sunburn is light red and not painful. An extremely severe sunburn is followed, 48 hours after, by blisters. The UVA sunburn is more violaceous with prominent vasodilatation.

Based upon their personal[6] and previous works[7], McKinlay and Diffey proposed an action spectrum for a minimal sunburn. The action spectrum was adopted by CIE and, later, by several international agencies. This action spectrum is very closed in the UVB range to the absorption of UV by DNA and several other biological events like bacteria mutagenesis, cell toxicity, delayed pigmentation. It has also a good correlation in the UVB range with the ACGIH action spectrum[8], adopted by IRPA and ICNIRP for UV-induced biological hazards including eye hazards. It should be noted that in the UVA, the erythema action spectrum is not representative of pigmentation or mutagenesis.

The minimal erythemal dose (MED) is a useful tool to define the biological effects of a UV dose. It is generally expressed in mJ.cm^{-2} or J.m^{-2}. Unfortunately, international agreement has not been reached on the value of the MED. However, it is commonly set à 21 mJ.cm^{-2} when normalized at 297 nm which corresponds to the dose necessary to induce a minimal redness on the back of a very sensitive skin. This value has been adopted by most of the agencies to calculate for example, the UV risk for indoor/outdoor workers or the potential consequences of changes in the ozone layer thickness.

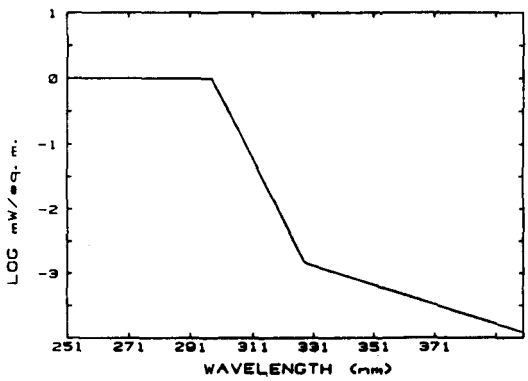

Figure 3. CIE human skin erythema action spectrum adopted after McKinlay and Diffey proposals.

For epidemiological inquiries related to skin cancers, melanocompromized subjects have an average MED of 25 ± 5 mJ.cm^{-2}, and melanocompetent subjects, an average MED of 45 ± 15 mJ.cm^{-2}. These values are characteristic of Caucasian subjects, mongoloid and metis have values closed to 90 mJ.cm^{-2}, and Negro skin, closed to 180 mJ.cm^{-2}.

- **Skin pigmentation**

Skin pigmentation is considered as an adaptation process, a direct consequence of UV radiations for a sufficient dose. It takes at least 24 hours to notice an increased pigmentation and it culminates at 8 days for a single irradiation. This **delayed pigmentation** is a consequence of an increased production of melanin pigment and its transfer to surrounding keratinocytes. In the UVB range, the induced division of the keratinocytes increases the thickness of the epidermis and so, the total content in melanin. This photoadaptative complex mechanism may be enhanced by further consecutive irradiations to an equilibrium state reached by almost doubling the total epidermal thickness, including the stratum corneum and a large load of melanin which increases the basic MED by a factor 4 to 10. In the UVA range, the multiplication of keratinocytes is

minimum while the increased pigmentation is provided directly by stimulation of melanogenesis. In this case, the basic MED is only increased by a factor 2. Phaeo and eumelanins coexist in melanocytes and keratinocytes in different ratios rendering the protection by melanins very much dependent on the quantity of phaemelanins which have been found more phototoxic than photoprotective when irradiated by either UVA or UVB. Melanocompromized subjects[9] have a large load of phaeomelanins which increases after irradiation and it is believed nowadays that the phenomenon is responsible for greater actinic damages and their consequences.

If the erythema action spectrum and the delayed pigment action spectrum for melanocompromized and melanocompetent subjects are nearly same in the UVB range, and in the shorter part of UVA, the action spectrum for pigmentation of melanocompetent subjects is diverging from the action spectrum for pigmentation of melanocompromized subjects. The same pigmentation is achieved for 5 times less dose of UVA in melanocompetent than in melanocompromized which remain closed to the UVA erythema dose[7].

The **immediate pigment darkening** reaction, a skin color observed immediately after UVA irradiation, is a totally different phenomenon which is believed to be the consequence of rapid oxidative reaction of colorless precursors of eumelanins. This transient phenomenon is nearly fully reversible with few hours and does not have any protective effect[10]. Only melanocompetent subjects are able to exhibit clearly this phenomenon which is readily perceptible after a 10 J.cm^{-2} UVA irradiation. The action spectrum has been obtained in the UVA range and is nearly flat from 330 to 380 nm, extending into the visible blue band.

- **Immunological effects**

The epidermis being an interface between environment and the body, it is naturally presenting a physical defense against external chemical or xenobiotics, but also an immunological organ able to control through immunological processes, the penetration of xenobiotic agents or the emergence of abnormal cells within the epidermis. The Langerhans cells located underneath the stratum corneum is the major afferent pathway toward a central lymphnode system, immunologically competent cells circulating by and through the capillary vessels, constituting the afferent pathway (response). For UV doses as low as half MED, the Langerhans cell functions are deeply altered within 24 hours and it takes almost 3 weeks to restore these functions[11]. Cytokins produced by other epidermal constituents, including the UV absorbing urocanic acid in its *trans* form, are able to modify either the afferent or efferent pathways, inducing a state of tolerance vis à vis any foreign substance. Several action spectra for this UV-induced tolerance have been proposed but are not yet acknowledged by the international scientific community.

In the modern vision of skin cancer initiation, promotion, and diffusion, the immunological consequences of UVR play an important role in favorishing the emergence of abnormal clones of cells[12].

- **Vitamin D_3 synthesis**

This important beneficial effect of solar radiations on the skin has been extensively studied: the UVB irradiation of 7-dehydrocholesterol by UVB produces previtamin D_3 which later will be converted into vitamin D_3 by thermal energy. The hyperproduction of previtamin D_3, by excess of UVB, is prevented by the conversion of previtamin D_3 by the production of lumisterol and tachysterol when previtamin D_3 is further irradiated. Upon continued irradiation, a photoequilibrium state is reached. As a consequence of the prevention of rickets, the production of vitamin D has probably been responsible for the establishment of skin variegated color population at the surface of the earth. Without supplementation of the diet in vitamin D_3, dark skin population children will be affected by rickets in countries with low level of UVB radiations (Nordic countries)

ADVERSE CUTANEOUS REACTIONS TO SUNLIGHT

Most of the cutaneous effects of sunlight are injurious. Beside the constitutional effects that we have already exposed, true pathogenic effects have been observed[13]. Two types of photomediated reactions may occur: phototoxic or photoallergic. Some molecules, either naturally produced, or absorbed through the skin or by the diet, present in the skin (epidermis/dermis) may absorb UVR and most of the time in the presence of oxygen, become toxic for the tissues. Clinically, **phototoxic reactions** are characterized by erythema and edema occurring within a few minutes to several hours after exposure. These chromophores may absorb at the maximum in any wavelength of the UV spectrum. For example, furocoumarins, following UVA radiation, are in their triplet state which is able to react with DNA to form adducts. Chlorpromazine reacts to form similar adducts with RNA. After the excitation, the photosensitizing molecule will return to the ground state and will be structurally unchanged, ready for more phototoxic reactions. Most of the photosensitizing molecules present this type of reaction in the UVA range.

Photoallergy can be defined as an acquired capacity of the skin to react in the presence of the photosensitizer by the development of circulating antibody or cell mediated immune response. These reactions are generally uncommon, not depending on the concentration of the photosensitizer and extend beyond the

exposed areas. Usually, less radiation exposure is required to induce a photoallergic response than a phototoxic response, the later being dose-dependent.

Among the adverse cutaneous reactions, **photo-induced diseases or photo-aggravated diseases have been** described. The spectrum of the diseases is very large from viral activation like herpes, to unknown agents responsible for polymorphic light eruptions which are afflicting 10% of the women population of clear skin complexion. The course of systemic diseases, metabolic disorders (porphyry) and complex immunological disorders (lupus), may be severely altered.

PROTECTIVE MEASURES

It is desirable to provide skin protection against acute and hypothesized chronic exposure hazards. In order to reduce the ambient solar aggression or the leisure/vacation overexposures, several measures should be taken[1]. Education should play a great role. The education programs should start very early at school and be regularly repeated for adults, through publicity campaigns, using all media. Staying in the shade during the most solar aggressive hours, wearing adapted clothing, hat with minimal 7 cm brim, applying adapted protective sunscreens, are elementary measures. The concept of a protection factor is useful to characterize items such as sunscreens, clothing. Generally, the protection factor is defined as the ratio of effective dose (unprotected) to effective dose (protected). It should take in account the spectral irradiance and the relative spectral effectiveness, the spectral transmission of the protective item. To communicate for the protective measures, the global UV-Index established in 1996, is a very effective tool.

CONCLUSIONS

International guide lines on exposure to ultraviolet radiations have been based on the same basic criteria of ACGIH (1993) and IRPA / ICNIRP (1991). The basic exposure limit (EL) for general public and occupational exposure to UV incident on the skin is 30 $J.m^{-2}$ effective when the spectral irradiance on skin surface is mathematically weighted by the hazard relative spectral effectiveness factor from 180 nm to 400 nm. Several tables express the relative spectral effectiveness of each wavelength and the maximum values of radiant UV exposure incident upon the unprotected skin with an 8-hour period.

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ULTRAVIOLET RADIATION: THE EYE

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INTRODUCTION

Under most conditions, the eye is well adapted to protect itself against ultraviolet radiation encountered in the outdoor environment as a result of the exposure geometry of the sun. Only when snow is on the ground does one experience acute effects of UV sunlight exposure (i.e., "snow blindness," or photokeratitis).[1-6]

With regard to artificial sources, there are many occasions where one views bright light sources such as tungsten-halogen lamps, arc lamps and welding arcs. Such viewing is normally only momentary because of the aversion response to bright light and due to discomfort glare. However, such an aversion does not take place for germicidal lamps and other UV lamps which do not contain a strong visible component in their spectrum. The adverse effects from viewing such sources has been studied for decades and during the last two decades guidelines for limiting exposure to protect the eye have been developed. The guidelines were fostered to a large extent by the growing use of lasers and the quickly recognized hazard posed by viewing laser sources.[3,7-12]

BIOLOGICAL EFFECTS

For each adverse effect upon the eye in the UV spectral region, the dominant injury mechanism is initially a photochemical reaction in the exposed tissue. The relative spectral risk for each type of injury is determined by a photobiological *action spectrum*. As with any photochemical reaction, the action spectrum describes the relative effectiveness of different wavelengths in causing a photobiological effect. The action spectra for UV photokeratitis (or "welder's flash") for humans as well as animals,[13-14] for photoretinitis[15-17] and acute cataractogenesis[18-19] have been published. Epidemiological studies of UV effects upon the eye support laboratory findings that acute effects are largely the result of UV-B radiation exposure.[3] The product of the dose-rate and the exposure duration always must result in the same exposure dose (in J/cm^2) to produce a threshold injury. This characteristic of photochemical injury mechanisms is termed the *Bunsen-Roscoe Law*, or the rule of reciprocity (of exposure duration and irradiance), and helps to distinguish these effects from thermal burns.

There are at least four separate types of hazards to the eye from UVR exposure, and protective measures must be chosen with an understanding of each:

1. *Photokeratoconjunctivitis* (acute inflammation of the cornea and conjunctiva as in "welders' flash") has been defined for wavelengths from 200 nm to 400 nm and cataractogenesis (lens cataract) has been demonstrated principally in the wavelength range from 290 to 320 nm, and perhaps occurs at greater wavelengths. The clear relationship between UV-B exposure and the onset of symptoms a few hours later conclusively prove that UVR causes this acute ocular effect. The signs and symptoms seldom last for more than a day or two.[13-14]

NOTE: The opinions or assertions herein are those of the author and should not be construed as reflecting official positions of the Department of the Army or Department of Defense.

2. *Pterygium and droplet keratopathies*, as age related pathologies of the cornea, which are common in island environments rich in UVR exposure have frequently been linked to chronic, life-long UVR exposure.[20] There are clearly other potential factors.

3. *Cataractogenesis*. There has long been a suggestion that UVR causes cataracts. The earlier age of onset of cataract in equatorial zones has led to a number of theories to explain this latitudinal dependence. Although some studies conclude (and it has long been argued) that exposure of the human eye to UVR plays an etiologic role in the development of some corneal pathologies,[17-20] this role in cataractogenesis has been questioned by others.[20-27] Even more under debate are theories that suggest that UVR and light may affect retinal disease.[22,26-27] However, epidemiologic studies which carefully consider individual exposure and ocular dosimetry show a clear relationship between life-long UV-B exposure and cataract.[23]

4. *Photoretinitis*. The principal retinal hazard resulting from viewing bright visible light sources is photoretinitis, e.g., *solar retinitis* with an accompanying scotoma ("blind spot"), which can result when one stares at the sun for several minutes. Solar retinitis was once referred to as "eclipse blindness" with an associated "retinal burn." Only in recent years has it become clear that photoretinitis results from a *photochemical* injury mechanism following exposure of the retina to shorter wavelengths in the visible spectrum, i.e., from violet and blue light. Prior to conclusive animal experiments at that time (1976),[3] photoretinitis was thought to result from a thermal injury mechanism. However, it has been shown conclusively that an intense exposure to short-wavelength light (hereafter referred to as "blue light") can cause retinal injury. The action spectrum has been defined principally from 400 nm to 550 nm in the visible; but, wavelengths from 310 nm to 400 nm in the UV can also produce effects in aphakic eyes (i.e., eyes with lens surgically removed), and possibly even somewhat in normal eyes. Normally, UV-A radiant energy is <u>not</u> implicated in this type of retinal injury unless the normal (UV-absorbing) crystallin lens of the eye has been surgically removed, as during cataract surgery. Accelerated ageing of the macula (age-related macular [central retinal] degeneration) may also be related to chronic retinal exposure to UV-A and short-wavelength visible light.[15]

Blue-light retinal injury (photoretinitis) can result from viewing either an extremely bright light source (e.g., the sun) for a short time, or a bright light of lesser brightness (e.g, a tungsten-halogen filament) for longer exposure periods.[3] Figure 1 summarizes the aforementioned biological effects.

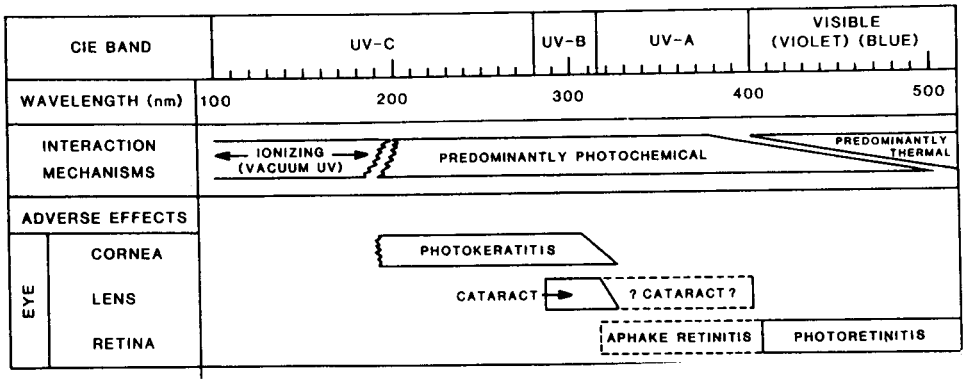

Figure 1. Adverse Biological Effects Attributed to Ultraviolet Radiation and Short-Wavelength Light.

GEOMETRY OF EXPOSURE

Exposures of the anterior structures of the human eye to UVR depend upon the relative position of the light source and the degree of lid closure. In bright sunlight, the orbital ridge above the eye and the upper lid considerably shade the cornea from excessive UVR exposure, thus limiting the chance for UVR photokeratitis in bright sunlight, except when viewing highly reflective snow.[3-6] Geometrical factors dominate the determination of ultraviolet radiation (UVR) exposure of the eye. The degree of lid opening limits ocular exposure to rays entering at angles near the horizon. Clouds redistribute overhead UVR to the horizon sky. Mountains, trees and building shield the eye from direct sky exposure. Most ground surfaces reflect little UVR. The result is that highest UVR exposure occurs during light overcast where the horizon is visible and ground surface reflection is high. By contrast, exposure in a high mountain valley with green foliage results in a much lower ocular dose. Other findings of these studies show that retinal exposure to light and UVR in daylight occurs largely in the superior retina.[6]

Despite a substantial literature on the adverse effects of ultraviolet radiation (UVR) and intense visible light upon ocular structures--particularly upon the retina--controversy still surrounds the question of whether natural and man-made light sources are really hazardous under normal viewing conditions. Acute studies of UVR and light injury of the cornea, lens and retina of experimental animals have generally employed excessive light levels. Although scientific evidence accumulates to indicate that chronic exposure conditions may accelerate ageing processes in the retina, the quantitative question of: What is safe?" remains to be conclusively answered. In recent years the potentially adverse effect of UVR exposure of the eye have been the subject of numerous scientific reviews and meetings. Present scientific knowledge in this area impacts upon the determination of optical product safety, ophthalmic instrument safety, and the design of sunglasses, and ophthalmic lenses. The one apparent controversy with respect to thresholds of UV photokeratitis and the ACGIH EL for UV-B can be seen at 310 nm in Figure 2. The reported threshold may be higher, since the bandwidth of the xenon-arc monochromator exposure system was at least 5 nm, and the effective wavelength would be less than 310 nm along such a steep slope.

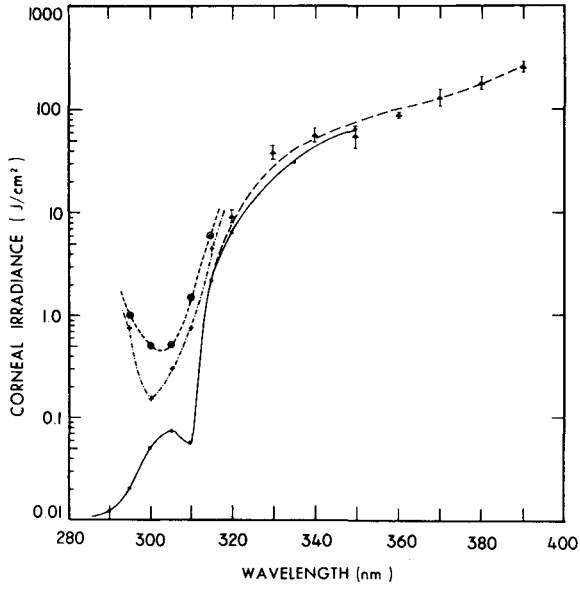

Figure 2. The thresholds (inverted action spectra) for photokeratitis[13-14] and cataractogenesis[18] are shown as a function of wavelength. Note the logarithmic ordinate.

Ultraviolet radiant energy at wavelengths shorter than 280 nm (referred to as UV-C) is totally absorbed in the cornea and is totally imperceptible. Wavelengths in the UV-B (280 - 315 nm) are absorbed largely in the cornea, but also reach the crystallin lens, and at wavelengths greater than 300 nm, there is a very small fraction of energy (of the order of 1 % or less) that may reach the retina.[3,15-17] Higher UV-A transmittance of the lens is most notable during youth.[3,20] Wavelengths in the UV-A (315 - 400 nm) spectral band are strongly absorbed in the lens; much of this energy is re-emitted as fluorescence in the visible spectrum.[28] In the absence of visible light, UV-A radiant energy may be perceived by fluorescence of the lens and some visual sensitivity to wavelengths below 400 nm which reach the retina. This general perception of a UV-A source can be described as a fuzzy, somewhat discomforting-to-view, blue-grayish light.[28] UV-A emitting lamps (frequently termed "black lights") are used extensively in industry for non-destructive testing by fluorescence techniques, and in entertainment and novelty applications to illuminate fluorescent signs, posters, in discos for special effects, in mineralogy to classify certain minerals, etc. In these applications, the eye is normally exposed to UV-A irradiances less than 1 mW/cm^2.[3] Although the UV-A at ground level from sunlight is 1 - 5 mW/cm^2, the actual UV ocular exposure to ground reflections and skylight is normally about 1 mW/cm^2 or less.[21]

THE EYE'S EXPOSURE OUTDOORS

Our exposure to UVR and light outdoors constantly changes during the day. We are largely unaware of the degree of these changes--especially the change in exposure rate of UV-B (the shortest wavelengths in terrestrial sunlight). This spectral change is not apparent in the visible part of the sun's spectrum until nearly sunset; and an attempt to take a color photograph either very early or very late in the day will result in a picture that is yellowish or orange in hue. We are fortunate that the Rayleigh scattering of sunlight by air molecules favors UVR and blue light (hence the blue sky). For longer pathlengths through the atmosphere when the sun is low in the sky, much more UVR and sunlight is scattered and the sun which is white at noonday becomes yellow and then orange as less UVR and blue light are present in the direct rays. When the sun is overhead and "white," it would take only 90 seconds to stare at the sun and receive a blue-light retinal burn. A few hours later, it would take several minutes, and it is virtually impossible at sunset. Thus, the geometry of exposure as well as the spectrum (hue) plays a major role in determining the hazards of viewing the sun. Fortunately, we seldom look directly overhead when the sun is very hazardous to view, and the sun is not very hazardous to view when the sun is sufficiently low in the sky to fall within our normal field-of-view. Furthermore, when the sun is greater than about 10° above the horizon, we squint, thus shielding the retina from direct exposure. Figure 3 illustrates the limited angular exposure.

Although the cornea is more sensitive to UVR injury than the skin, we seldom experience a corneal burn when out in sunlight. Again, the geometry of exposure helps. When the sun is overhead and UVR exposure is most severe, the brow ridge and upper lid shield the cornea, and if the eye is turned away from the sun, the more intense scattered UVR from overhead strikes the cornea at a grazing incidence where most is reflected and little is absorbed. Only when the incident UV rays are parallel to the pupillary axis are most rays absorbed (in fact, 98 % are absorbed). When one looks down at the snow the UVR is reflected directly into the eye, hence the eskimos' traditional eye protector, the slit in whalebone, provided geometrical rather than spectral protection against the UVR exposure. The strong dependence of reflections with angle of incidence is termed Fresnel's Law of Reflection. This Law not only explains the survivability of the cornea in an overhead bath of UVR, but also the glare experienced over water. When the sun is overhead, a body of water reflects the UVR upward, but only approximately 2% is reflected. When the sun is low in the sky, much of the incident light is reflected, but now the UVR and blue light have been filtered out of the direct rays by the atmosphere and are therefore harmless. Nevertheless, discomfort glare originates from the strong water reflections which because of polarization by reflection and scattering can be selectively filtered by polarizing lenses. Such lenses may not necessarily filter out the UVR.

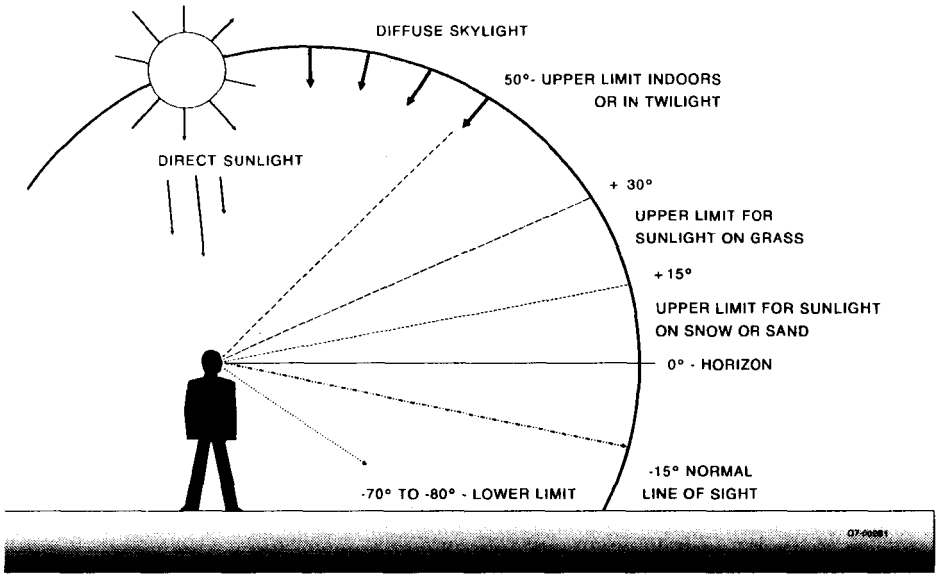

Figure 3. The limited solid angular exposure to UVR in the outdoor environment is limited by the geometry of the brow-ridge, but also strongly affected by lid position (squinting), which depends on the brightness of the visible light environment.[4-6]

UVR EXPOSURE FROM ARTIFICIAL SOURCES

The envelopes of most commercial lamps used for illumination filter out UV-B or are installed in a filtering lighting fixture. However, in some industrial processes, workers may actually be exposed to levels greater than one would experience in the outdoor environment. In this figure, the relative risk of fixating upon different types of bare lamps after the spectral distribution has been weighted by biological action spectra for uv photokeratitis and the retinal "blue-light" hazard. Since the ambient optical radiation levels in our natural environment are barely tolerated, it is hard to imagine that an exposure to an artificial light source would be tolerated at an exposure level exceeding the outdoor levels. Aside from pulsed lasers, where the individual may have no natural protective mechanisms such as the blink reflex of the eye, I have not encountered an instance where an individual repeatedly exposed himself or herself to levels where acute injury took place.

PROTECTIVE MEASURES

Where feasible, it would appear to be desirable to provide eye protection against an hypothesized chronic-exposure hazard simply to be on the safe side. If eye protection is worn, peripheral protection is paramount because of the *Coroneo Effect*.[6] However, when protective measures are extremely costly, or they possibly introduce another hazard, one has a serious dilemma. Such a dilemma has apparently arisen with regard to ultraviolet absorbing intraocular lenses.[17]

CONCLUSIONS

The present ICNIRP EL's and ACGIH threshold limit values (TLV's) for UVR are based upon an understanding of both acute and chronic effects upon the eye. Although exposures to optical radiation in the physiological range have generally been accepted as safe, greater knowledge today suggests that the absence of acute effects does not mean that some damage has not occurred at the

molecular level. Delayed effects appear to be possible from UVR exposure to the eye as well as the skin. Furthermore, growing evidence suggests that retinal degradation may also occur from chronic exposure to short wavelength light, and possibly from the trace amount of UV-A that reaches the retina as well. The present exposure limits probably reduce the risk from all of these delayed effects quite significantly, but one cannot argue that the risk is zero. As with almost all physical and chemical agents, one cannot speak of "zero risk" -- it is scientifically specious.

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CLIMATOLOGY OF UVB AND OZONE VARIATIONS & THE GLOBAL SOLAR UV-INDEX

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INTRODUCTION

Human overexposure to solar ultraviolet radiation (UVR) can result in acute and chronic adverse health effects on both the skin and the eye. Skin cancer (both non-melanoma and malignant melanoma) and cataract impose a huge social and cost burden on many societies throughout the world. Such human health problems can be avoided if the individual reduces their UVR exposure. Unfortunately enlightenment may not help persons who have experienced high episodic exposures during childhood as this appears to be an important causal factor in melanoma. In some countries public educational campaigns have been underway for decades in other countries they are just beginning; the global solar uv-index provides a globally consistent means of reporting or predicting UVR as part of public education on UVR exposure. There are now indications that some of these programs have been effective in halting the climb in melanoma incidence.

The UVR, and in particular UVB, reaching the earth's surface varies with both latitude and time (both of the day and year). The transmission of the extraterrestrial radiation through the atmosphere is determined by ozone, clouds, aerosols and to a lesser extent, trace gases. In recent decades there has been considerable concern that long-term changes in ozone and perhaps clouds and aerosols may result in changes in the UVB at the earth's surface.

Atmospheric attenuation of solar UVR

The extraterrestrial solar spectrum is rich in UVC (200-280 nm), UVB (280-315 nm) and UVA (315-400 nm). However passage through the atmosphere removes the UVC and most of the UVB component. This reduction is due mainly to absorption by ozone and molecular scattering. The amount of transmitted UVB is sensitive to even small changes in the ozone column (see Figure 1).

The spectral irradiance can be calculated for different total ozone amounts using an ARL-modified version of the Björn model (1). The decrease in the ozone column from 300 to 270 DU results in the UVB increasing by 95%, 20% and <5% for the wavelengths of 295, 305 and 315 nm, respectively. Overall this 10% decrease in ozone results in about a 10% increase in total UVB (280-315 nm).

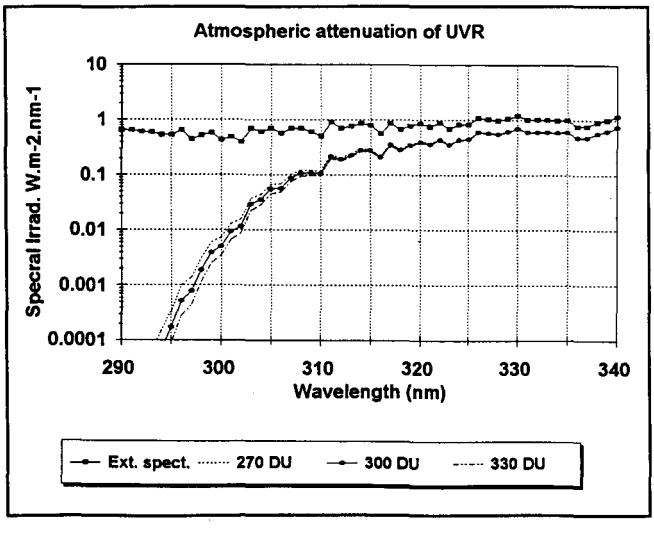


Figure 1
Spectral irradiance at the earth's surface for clear skies, a SZA of 5° and for three different ozone columns. Also shown is the extraterrestrial spectral irradiance.

Solar zenith angle (SZA)

The SZA describes the angle between the local vertical direction and the direction of the solar disk. The smaller the SZA, the less atmosphere the solar radiation must pass through and hence the greater the solar UVR (see Figure 2). The SZA varies with time of day and year and with latitude. Only within the tropics is a SZA of 0° attainable (as the sun passes overhead).

For a given ozone column (here 270 DU) the total UVB for SZA=35° is 64% of its value at SZA=5°. At 70° the corresponding value has dropped to 6.4%. For total UVR the effect is less with values of 77% (35°) and 23% (70°) obtained for the two examples.

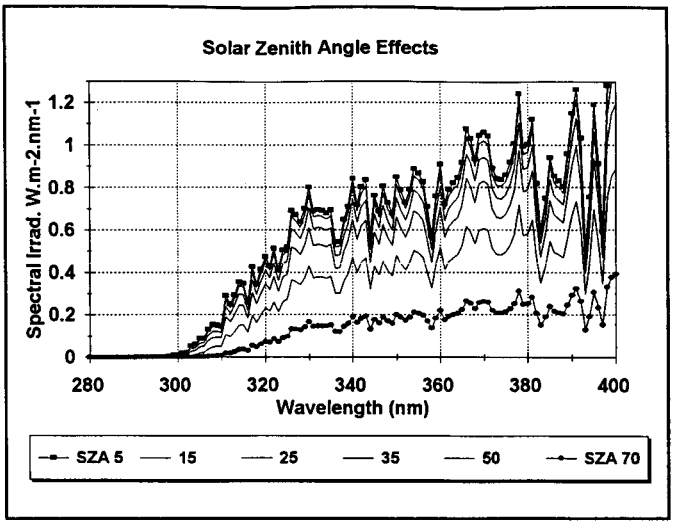


Figure 2.
The effect of changing SZA on the spectral irradiance at the earth's surface.

Total ozone

Stratospheric ozone is produced by the photolysis of oxygen molecules by short-wavelength solar UVR (wavelengths <242 nm). Ozone is formed largely in the equatorial regions and it is transported polewards. The amount of ozone is determined by the dynamic balance between the production and several catalytic destruction cycles and by transportation processes. Ozone depletion concerns in recent decades result from a man-made chlorine destruction cycle.

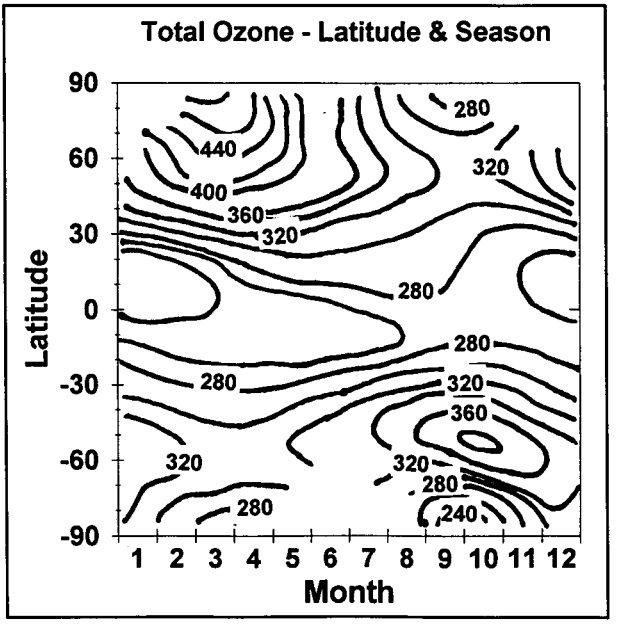


Figure 3.
Indicative latitudinal and monthly variation of total ozone (DU). Values are averages for the years 1979-89 from the TOMS satellite data (2).

Ozone depletion

Both satellite and Dobson ground-based measurements show decreases of total ozone in winter in the Northern Hemisphere. There is evidence of significant decreases in spring and summer in both the Northern and Southern Hemispheres at middle and high latitudes. There is no evidence for any significant trends in the tropics. Throughout the 1980s and 1990s there have been deep and extensive ozone holes in the southern spring.

Table 1.
Total ozone trends in units of percentage change per decade (3).

Season	TOMS Data: 1979-91 45°S	Equator	45°N
Dec-Mar	-5.2	+0.3	-5.6
May-Aug	-6.2	+0.1	-2.9
Sep-Nov	-4.4	+0.3	-1.7

The analysis shown in Table 1 indicates that little or no depletion has occurred in the equatorial region and at 45°N the greatest depletion occurs during winter when solar UVB is at its lowest. At 45°S significant depletion occurs all year with summer only slightly less than winter.

Effective UVR: computed and measured

The biological effectiveness of ultraviolet radiation is very wavelength dependent with UVB being considerably more biologically effective than UVA radiation. Within the UVB, between 300 and 315 nm, the relative spectral effectiveness decreases by three orders of magnitude which means that spectral measurements within this region must be very precise or large errors in the calculated effective radiation will result. The erythemal response curve of the International Commission on Illumination (4 and Table 2) is used to weight the spectral irradiance distribution.

Table 2
CIE erythemal effectiveness function.

Effectiveness function (S_λ)	Wavelength region (nm)
1.0	$250 \leq \lambda \leq 298$
$10^{0.094(298-\lambda)}$	$298 \leq \lambda \leq 328$
$10^{0.015(139-\lambda)}$	$328 \leq \lambda \leq 400$

The amount of UVR required to induce sunburn or erythema in normal human skin is usually designated as the minimum erythemal dose (MED). Using the CIE erythemal effectiveness function, the MED for skin type I, the most sensitive type (skin that always burns and never tans), is approximately 200 J m^{-2} (5).

The monthly average daily erythemal UVR for Melbourne (37.8°S) is shown in Figure 4 for six of the years between 1986 and 1993. The general shape is typical of a mid-latitude site and is determined by the SZA variation over the year. The annual total MEDs are also given. The average annual total is 4160±270 (SD). Year-to-year differences are due largely to variations in cloud cover and total ozone.

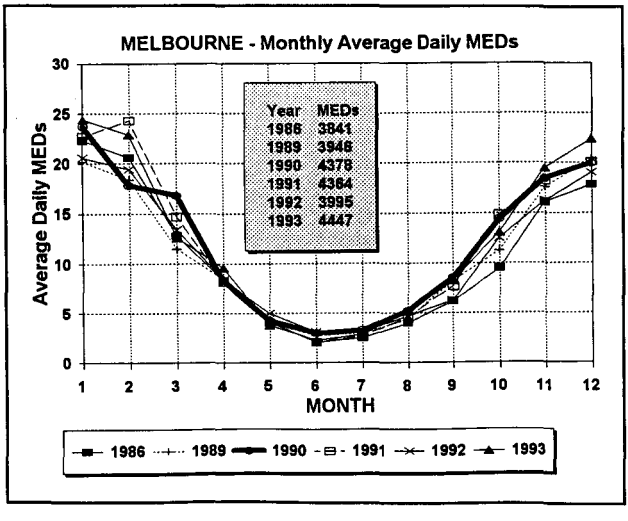


Figure 4. Variation of the total mean erythemal UVR (in MED units) for Melbourne in the years indicated.

The global distribution of erythemal irradiance can be calculated (3) and this is shown in Figure 5. The patterns are as expected with very high and relatively constant values throughout the equatorial and tropical regions. In the mid-latitudes the values are high in summer but are very low in winter. The high levels in Antarctica as a result of the ozone hole are not represented in this plot.

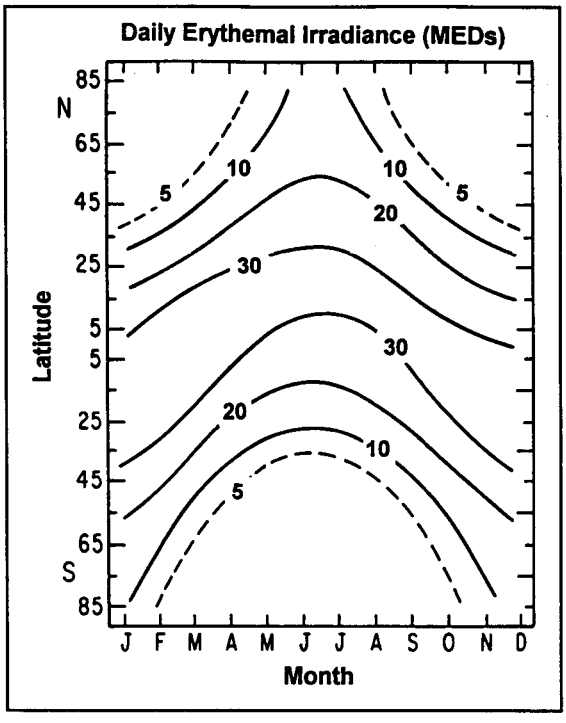


Figure 5. Latitudinal and monthly distribution of integrated daily erythemal irradiance in units of MEDs. Computed for clear skies and based on TOMS zonally averaged total ozone for 1980 (3).

Measurements and calculations have been made of erythemal or effective UVR for a number of locations in both the Northern and Southern Hemisphere and some of these are shown in Figure 6. The results are consistent with the data presented in Figure 5. The results for the southern hemisphere (6) are measurements and hence include the effect of cloud cover which can be quite significant; for example, during the tropical 'wet-season'. The high latitude results (9) are measurements from Finland. The more recent results of Seckmeyer et al.(10) agree largely with Figure 5 but also illustrate the impact of the Antarctic ozone hole on the southern hemisphere results.

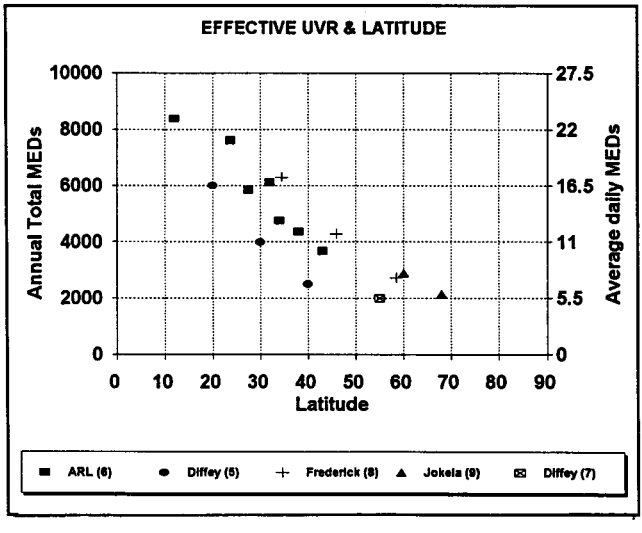


Figure 6. Measured effective UVR during the years 1989-1991. The results for the NH and SH are not differentiated on the plot.

Erythemal UVR and clouds

The transmission of UVR through clouds is a function of the optical depth. Solar UVR transmissions can range from a few percent up to 80 or 90% for cirrus clouds. In the presence of clouds measured UVR at the earth's

surface can range from practically 0% to 120% of the clear sky value. The enhanced solar UVR (11) is the result of scattering from the sides of cumulus clouds - obviously these levels will only be obtained when the solar disk is not blocked by cloud.

There is still much research to be done in this area. Difficulty arises when there is the need to include cloud cover in computer-generated UVR irradiances. A number of different parameterisations have been used (2). Data for Australia is reasonably well-represented by: $Irrad_{cloud}/Irrad_{clear} = 1 - 0.06C$; where C is the cloud cover in octas.

Cloud droplets which are considerably larger than aerosols are assumed to have scattering cross-sections that will be largely independent of wavelength throughout the UV region and therefore clouds are not expected to have any wavelength dependence transmission factors in the UV.

Four years of daily total erythemal data (in MEDs) is shown in Figure 7 along with the computed clear sky data. The large variation is due mainly to cloud although the enhanced summer values are the result of short-term total ozone reductions.

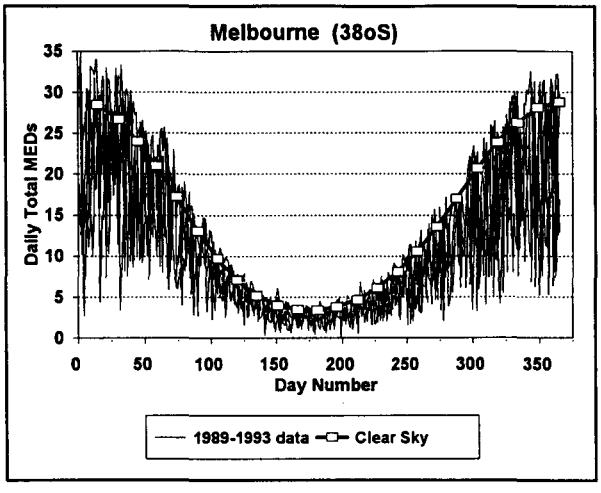


Figure 7. Fours years of erythemal data for Melbourne (37.8°S). Also shown is the computed clear sky data.

Erythemal UVR and SZA

The erythemal UVR is obviously quite dependent on the SZA. With increasing SZA there is a disproportionately greater drop in the UVB than the UVA and because of the greater biological responsivity in the UVB the weighted irradiance will drop at an even greater rate (Figure 8).

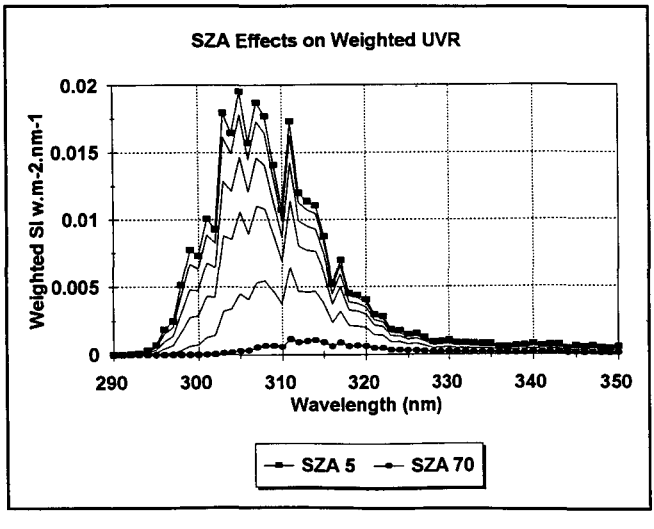


Figure 8. The effect of SZA (5°, 15°, 25°, 35°, 50° and 70°) on the effective spectral irradiance. Computations have been performed for an ozone column of 270 DU. The unweighted spectral irradiances are given in Figure 2.

It can be seen that the peak in the weighted spectral irradiance shifts to longer wavelengths with increasing SZAs.

Table 3 provides the integrated effective irradiances (280-400 nm) for different SZAs. Similar results are also provided for total ozone columns of 300 and 330 DU. The numbers in parentheses give the time (minutes) to reach 1 MED at that constant irradiance. Values range from 10.9 to 190 minutes or 5.50 to 0.3 MEDs/hour.

Table 3

Effect of SZA on the integrated effective irradiance ($W.m^{-2}.nm^{-1}$) for different ozone column amounts. The number in parentheses is the time in minutes to achieve 1 MED.

SZA	270 DU	300 DU	330 DU
5	.307 (10.9)	.269 (12.4)	.239 (13.9)
15	.283 (11.8)	.249 (13.4)	.221 (15.1)
25	.240 (13.7)	.211 (15.8)	.188 (17.7)
35	.185 (18.0)	.163 (20.5)	.145 (23.0)
50	.100 (33.4)	.088 (37.8)	.079 (42.2)
70	.022 (154)	.019 (171)	.018 (187)

Trends in ozone and calculated trend in erythemal UVR

The averaged trend in ozone and the corresponding trend in erythemal UVR is shown in Figure 9 for the period 1979-89 (2).

The asymmetrical nature of the plot is due to the large springtime losses in Antarctica. Unfortunately the erythemal data which has been computed for clear skies has not been verified at this point in time because of the absence of accurate measurement over a sufficient period of time.

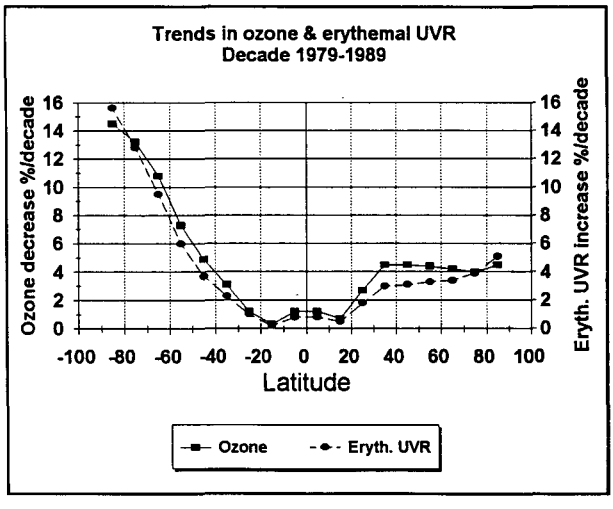


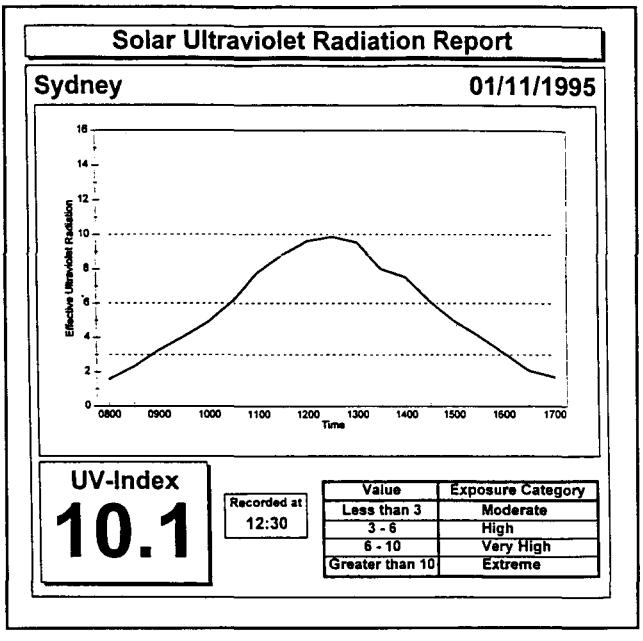
Figure 9. Trend in ozone (zonally averaged) for the decade 1979-89. The computed change in erythemal UVR is also given (2).

Educational campaigns

Public and professional education about skin cancer commenced in Australia in about 1960 with programmes in both Queensland and Victoria (12). Since 1981 the Anti-Cancer Council of Victoria has run a "Slip!, Slop!, Slap!" and later "Sunsmart" national campaigns (13). The two aspects to skin cancer control - prevention and early detection - are both embraced by what is now a national program coordinated by the Australian Cancer Society.

The Australian Radiation Laboratory, using data from its national measurement network (14), provides to the media a daily chart of UVB for the capital cities in each state. These are broadcast in the weather section on the nightly television news report. This 12-15 second time slot provides a unique opportunity to reach a large audience and to help educate the public about UVR and the link or otherwise to ozone, cloud cover and daily temperature. This has generated considerable public interest. Figure 10 shows a typical report that is provided to the media at 1700 hours, local time. The template the media uses is colour graded with red and orange representing the highest UV categories.

Figure 10. Typical daily UVR chart provided to the media.



Global Solar UV-Index

The index, a joint recommendation of WHO, WMO, UNEP, ICNIRP and IARC (15) is a measure of the solar UV levels relevant to health effects and is useful for warning people about the degree of hazard that existed or could exist on a particular day. The index is calculated by multiplying the daily maximum CIE-weighted UVR, averaged over 30 minutes, by 40. For the first time there will be a global system in place which will make comparisons straightforward and will also assist tourists. Figure 10 also provides the index for the day and this allows the public to compare with the previous day's prediction.

Many schools and community groups are taking a greater interest in minimising UVR exposure and a variety of initiatives have emerged from the enforced wearing of hats in school grounds, modification of the hours that schoolchildren spend outdoors to the planting of more shade trees in open areas. The provision of shade structures in homes, school grounds and in public areas including parks is strongly advocated as a sensible radiation protection strategy. Appropriate materials are available (16,17) although care should be taken in design as users can often still be exposed to scattered solar UVR.

Evaluation of programs

Evaluation of a prevention program measures initially a change in knowledge and attitudes, then a change in behaviour and eventually a change in the incidence and mortality rates of skin cancer (13).

A survey of sunscreen use conducted on Brisbane beaches (18) found about 70% of females and males applied sunscreen. Half of the sunscreens provided the maximum protection (SPF 15+) and almost 90% used a waterproof formulation. However the sunscreen was not applied over the entire body with over half neglecting ears and lower limbs.

Following the introduction of the Sunsmart programme, Hill (19,20) conducted surveys in three summers in the period 1988-1990 to determine trends in exposure to sunlight by monitoring the prevalence of sunburn and sun-related attitudes and behaviours in 4,400 adults. In 1988 up to 18% of some groups reported sunburn in a given weekend. Over the three years the average proportion sunburnt dropped from 11% to 7%. They also found that substantial attitudinal changes also occurred with hat wearing and sunscreen use increased to 29% and 21% respectively. There was no clear trend in the proportion of the body covered by clothing - the mean proportion of body surface area covered being 70%. However, there was significant evidence that beliefs were more realistic and conducive to sun-protective measures.

With regard to skin cancers it is still too early to fully evaluate the effectiveness of the educational campaigns but there are promising indications. It has been reported that excised cutaneous malignant melanomas are becoming thinner which indicates that they are being found earlier (in more than 90 per cent of cases there is no clinical evidence of secondary spread of the cancer) which in turn results in a higher 10 year survival rate (21). In Australia it has been found that melanoma mortality has peaked with figures showing a small but significant fall in rates for women and a plateau in the male death rate. For 1990-91 the age-standardised

mortality rates in Australia were 4.9 and 2.5 per 100,000 for males and females respectively (22). It is expected that a consistent fall will be found over the next two years (23).

Conclusions

Much of the world's population reside in locations where it is easy to obtain large doses of solar UVB for at least part of the year. The individual's genetic susceptibility and personal behaviour will determine whether or not exposure will ultimately result in adverse health effects. Continuing atmospheric ozone depletion places even more pressure on the individual to adopt sensible outdoor behaviour.

Educational programs aimed at both the workforce and the public have succeeded in creating an awareness of the dangers of overexposure to ultraviolet radiation. The global solar UV-index is a useful addition to this campaign. The fact that changes in knowledge and attitudes have been accompanied by behaviour changes indicate that educational programmes are having an effect in some countries. Consumers have the products available which will provide good personal protection and the information to enable a proper choice. The adoption of sensible behaviour and personal protection can help to achieve :

- avoidance of sunburn which is accepted to be a major risk factor for melanoma and
- a reduction of an individual's accumulated UVR dose during work and leisure hours which has a role in non-melanoma skin cancer.

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CHRONIC EFFECTS OF UV ON HUMAN SKIN

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INTRODUCTION

UV radiation causes a number of chronic degenerative changes in the skin, mainly in Caucasian populations as a result of its action on keratinocytes, melanocytes and components of the dermal stroma including fibrous tissues (collagen, elastin) fibroblasts and blood vessels. Among the degenerative changes, skin cancers are the most dreadful perspectives, but photoaging is of the most important general concern for the population.

MECHANISMS OF UV CARCINOGENESIS

There is abundant evidence that UV causes damages by direct photochemical effects and oxidative effects[1]. There is also evidence that activated oncogenes and mutated tumor-suppressor genes are present in some skin cancers[2-3]. From a review of the animal studies, it has been found that the effectiveness is to peak in the UVB range, that UVA is also carcinogenic at a much lower level of effectiveness[4]. It is believed that UVC is not effective for human carcinogenesis and that visible light will not play a significant role. Also from animal studies, the immune system may be deeply altered[5]. The suppression of systemic responses may favor the malignant cell proliferation and the metastatic processes. In man, beside anecdotal reports, no firm assessment can be made. Figure 1 represents the most comprehensive scheme to illustrate the complexity of carcinogenesis

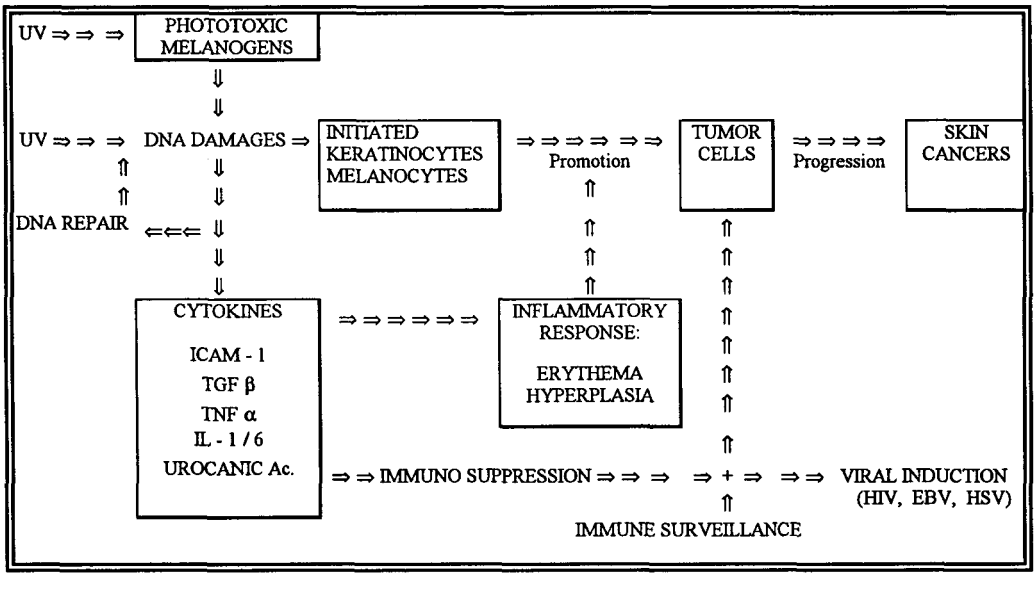

Figure 1. Scheme adapted from Ref. 5, modified to include melanin and melanocytic system.

Following irradiation, one can observe the production of p53 protein[2], the expression of a tumor suppressor gene which blocks the cell division in G1, allowing the DNA damages to be repaired[6]. The p53 protein is rapidly induced and has normally a very short half-life. If the damage is unreparable, the p53 pushes the cell into programmed cell death, called apoptosis (sunburn cells)[3]. Loss of normal function of the product of the p53 gene is usually caused by missense point mutations. The mutations in the p53 gene are very common alterations[8] and 5 specific loci of the gene have been found target for UV-induced lesions. The mutated p53 accumulates in the nucleus and is correlated with more aggressive tumors, metastasis and lower 5-year survival rate[6]. The discovery of the role of the p53 gene and other genes involved in the control of cell division is of a major importance to the understanding of the photocarcinogenesis process[7] as illustrated on the following figures 2 and 3.

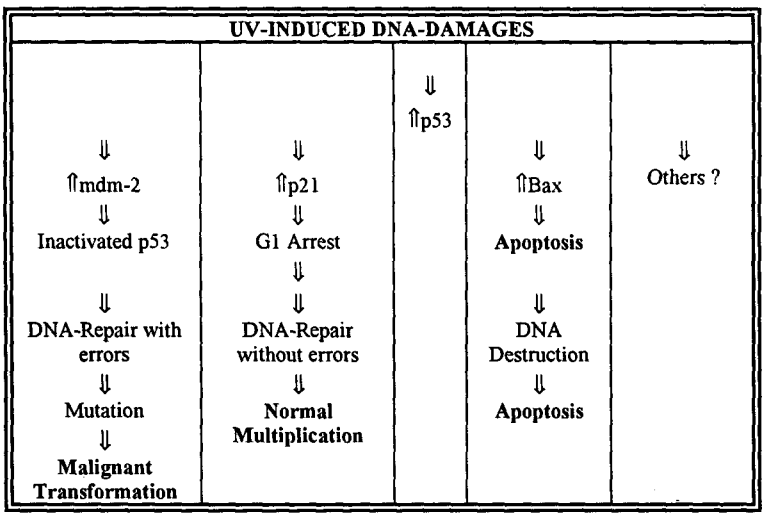

Figure 2. Molecular events following UV irradiation.

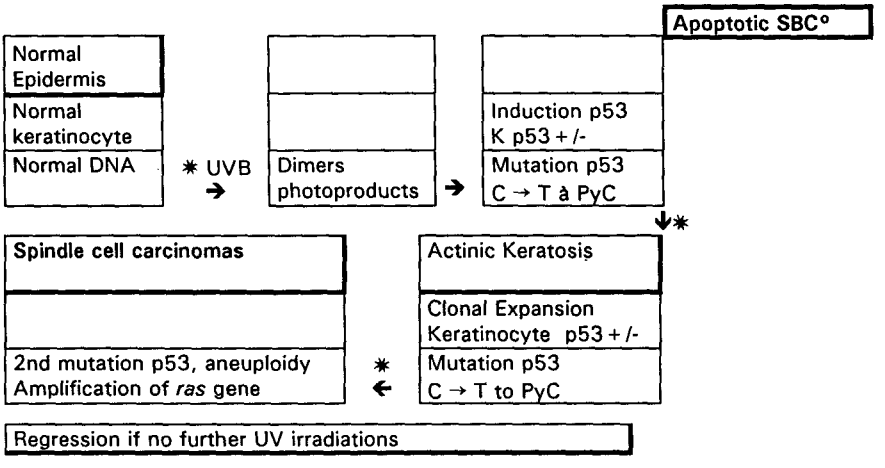

Figure 3: Synthetic aspects of steps initiating photocarcinogenesis

Benign lesions induced by solar exposures

Freckles and **solar lentigo** are pigmented macules occurring on the sun-exposed skin of Caucasians. Their prevalence is increased in highly sun-sensitive skin. Freckles occur commonly in children and are considered as marker for high sensitivity in most of the melanocompromized subjects. Solar lentigo increase with age, and are characterized by an increased number of melanocytes and increased concentration of melanin in the basal layer. An increased risk of melanoma has been observed in relation to freckling in childhood and an increased risk for non melanocytic skin cancer has been found in relation with freckling and prevalence of solar lentigo[9].
Melanocytic nevi are benign proliferations of melanocytes associated, in white populations, with phenotypic indicators of constitutional high sensitivity to the sun. They occurs mainly on body sites, maximally or intermittently exposed to the sun. There number and size are inconsistently associated with multiple sunburns in early life. They are associated with an increased risk for cutaneous melanoma. Precursors of melanins may induce DNA damages by photosensitization reaction[10] and might be responsible for melanocytic proliferation and melanoma.
Solar keratosis are benign proliferations of epidermal keratinocytes. They are very common on permanently exposed body sites in elderly people, in Caucasian populations living in areas of high ambient solar irradiance. They have been considered as phenotypic indicators of cutaneous sun-sensitivity and estimates of total sun-exposure. They are strongly associated with a risk of non-melanocytic skin cancers and are considered as precancerous lesions. They are also regularly associated with signs of dermal sun-induced degeneration (heliodermatosis).

Non-melanocytic skin cancers
There are 2 major types of non-melanocytic skin cancers: basal cell carcinoma (BCC) and squamous cell carcinoma (SCC). They represent one half of all cancers in the white population. Ethnic background is an important determinant for the risk on non-melanocytic skin cancer in Caucasians. In people of southern European origin, born in Australia, relatively to other people born in Australia, the risk of BCC was found 0.56 for those with one southern European grand-parents, 0.17 for two, and 0.1 for three or four[11]. Among recent case control and cohort studies, the relative risk for red or light hair-color population is around 2 for BCC and 2.4 for SCC. The incidence rate of non-melanocytic skin cancers does appear to increase with proximity of the equator with similar gradients for men, women at all ages. The same pattern was observed in Australia and in USA. Epidemiological studies on migrants reveal that the incidence and the mortality is lower in migrants coming from areas of lower sun-exposure when they arrive in the high solar irradiance countries, after 10 years of age. This has shown for BCC and malignant melanoma (studies performed in Australia, Israel, and California).
Beside the skin sensitivity, the total solar exposure has been found of out-most importance. Attempts have been made to estimate the dose-response relationship between UV and SCC and BCC in retrospective cohort study in general[12]. An alternative approach is estimating a dose-response relationship in a whole population. This work has been made in the context of estimating the increase in skin cancer expected from some increment in ground level UV caused by depletion of atmospheric ozone. The results have commonly been expressed in term of biological amplification factor (BAF)[13].

Authors	Region	Sex	BAFs for BCC	BAFs for SCC
Non melanocytic skin cancer			**Incidence**	**Incidence**
Scotto et al 1983	USA	M	1.3 - 2.6	2.1 - 4.1
		F	1.1 - 2.1	2.2 - 4.3
de Gruijl & Van der Leun 1991	USA	M & F	1.4	2.5
Moan et al 1989	Norway	M	1.5 - 2.0	1.2 - 1.5
		F	1.6 - 2.1	1.6 -1.8
Melanoma			**Mortality**	
Scotto & Fears 1987	USA	M	0.4	
		F	0.5	
Pitcher & Longstreth 1991	USA	M	0.4	
		F	0.3	
Moan & Dahlback 1992	Norway	M	0.9	
		F	3.2	
	Sweden	M	1.9	
		F	2.3	
	Finland	M	1.3	
		F	2.2	

Table I. Recent estimates of BAF for non-melanocytic skin cancers and cutaneous melanoma based on geographical correlation between average annual ambient UV and skin cancer incidence of mortality.

ACTION SPECTRUM FOR PHOTOCARCINOGENESIS

Recently, two large collections of data on carcinogenicity of UV in albino hairless mice, following exposure to multiple sources emitting in the UVA, B, and C ranges, have been combined to produce the best estimate for action spectrum for skin carcinogenesis in animal strains[4]. The figure 4 reproduces the action spectrum and its adaptation to the human skin situation taking in account the absorption by the multiple layers[14]. The definition of the action spectrum is excellent in the UVB range following closely the erythema action spectrum. A large uncertainty is still observed in the UVA range which can be considered as a plateau from 340 to 380 nm. This plateau is closer from the melanogenesis action spectrum in sensitive skin than from pigmentation in darker skin[15].

In a fish model in which UV radiations are able to induce melanomas, the action spectrum is confounded with the erythema action spectrum from 290 to 315 nm, but in the UVA, the tumor develops for 10 times less dose than for the mouse non-melanocytic tumors[16]. This observation raises the possibility of a specific deleterious role of melanin in the genesis of melanocytic tumors, a point which should be linked with some specific melanin-DNA induced damages in vitro[10, 17, 18].

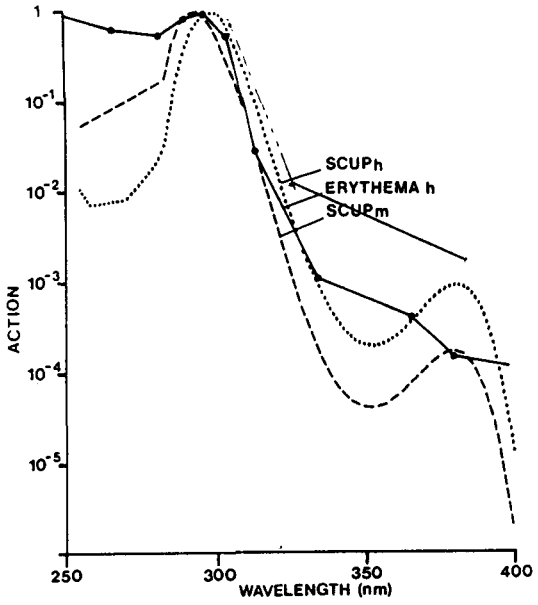

Figure 4. Estimated action spectrum ($SCUP_m$) for induction by UV of SCC in the skin of albino hairless mice (Skh-Hr1). The $SCUP_h$ is for human. On the top, the fish melanoma action spectrum is reproduced.

SKIN AGING AND ELASTOSIS

Chronic excessive UVR leads to the clinical picture recognized as photoaging. The clinical aspect is typically present on permanently exposed body sites, associating dryness, wrinkling and telangiectasia. Irregular pigmentation and solar lentigo are also observed. Histologically, this is caused by thinning of the skin, reduction in the papillary dermis thickness and replacement of the normal collagen-elastic material by elastotic bundles which lacks the elastic properties of the normal dermis. These changes are always associated with BCC and SCC and with a special form of malignant melanoma lentiginous type. The solar aging is dose-related and also related with the skin complexion. The disorders have been reproduced in hairless mice and attempts have been made to produce an action spectrum, using a specific clinical aspect of the mouse skin named sagging. Sayre and Kligman[19] have produced an action spectrum very close to the human erythema

action spectrum. However, Bissett[20] and Wulf[21] proposed a completely different action spectrum where UVA is 100 times more important. A final consensual action spectrum has not been yet obtained.

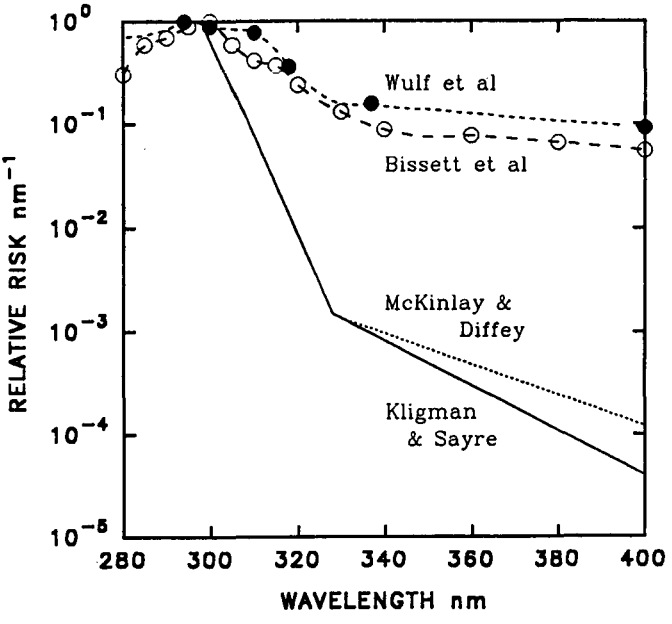

Figure 5. Mouse aging action spectra from reference 19

CONCLUSIONS

Chronic exposures and acute accidents of the skin to UV has been recognized as an important risk for skin cancers in human. Attempts have been made with mathematical models to correlate the ambient UV dose and occupational irradiations with the risk of skin cancers. Development of accurate global measurements of solar irradiance and personal dosimetry is expected in the future in order to reduce the exposure of the general population, to precise the measures to be taken for indoor and outdoor workers.

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RECENT DEVELOPMENT ON RADIOLOGICAL PROTECTION AND RADIOACTIVE WASTE DISPOSAL

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Throughout the world there are numerous specialised conferences organised on radioactive waste. Why debate this subject at IRPA9 among radiation protection specialists ?

In the nineteen sixties and seventies, the aim was claimed to be the finding of a satisfactory technical option. It was the time when the advantages and drawbacks of each option were weighed carefully : shallow land and underground disposal, sea bed, and sea dumping, sending it into space, transmutation... The range of options progressively decreased as public awareness increased particularly regarding any form of disposal in the sea. Therefore, those in charge of waste management attempted to gain international consensus from experts to demonstrate that disposal was technically feasible and that long term safety could be assessed.

The most recent developments on the social and ethical dimensions are opening new perspectives on how properly take into account the interest of future generations.

Within this context, attention is particularly focused on the difficulty of applying to long-term waste management the principles of radiation protection as defined by the International Commission on Radiological Protection (1).

From whence does this difficulty arise ?

Schematically, the radiation protection system which is applied to « practices » in general requires that the source of radiation be under active control. Protection options (physical, institutional and operational) put into place just as much at the design stage as during operation of the corresponding installations are such that exposure levels can be kept below individual dose limits (limitation principle), and, for each source, below dose constraints corresponding to well managed operations (optimisation principle). Practices can even be prevented from being adopted if the disadvantages that could result from it outweigh the advantages (justification principle).

The protection system for practices must take into account both normal as well as potential exposure which are not certain to occur and could be the result of accidental departures from the planned operating conditions following internal or external events.

By analogy with the normal exposures, ICRP recommends for potential exposure compliance with risks limits and risk constraints. The two terms of the risk which must be considered in this case are the probability of an event which could lead to exposure, and the consequences of this exposure.

It is important to stress that, should the potential exposure really occur i.e. the source be no longer under control, it could be necessary to intervene (intervention situation).

For what concerns long term solid waste disposal, there are some specific aspects to be taken into account in applying the radioprotection system. These aspects makes it more complex to demonstrate that the source is under control :

- it is difficult to guarantee institutional monitoring beyond a certain number of years, thus active control of the source ceases. Yet monitoring is a key part of the system because it means application of the principles can be verified and makes it possible to intervene should control be lost.

- it is difficult to assess the exposure situations which correspond to normal or potential development of disposal situations because of the large degree of uncertainty involved, the complexity of the degradation process, the variety of possible events and the difficulties involved in assessing the environmental characteristics.

In 1985, the ICRP dedicated a specialised publication to the disposal of solid waste, ICRP46 (2). This states that during the long time periods of disposal two kind of scenarios of radionuclides release from the site has to be envisaged :

a) normal scenarios involving the gradual degradation of the barriers leading to the slow migration of radioactive elements : these are scenarios which are very likely to come about ;

b) probabilistic scenarios which correspond to processes or events which have constant or time varying probabilistic occurrence.

ICRP46 represented significant conceptual progress because it was the first publication by the Commission which extended the protection system to cover probabilistic situations and thus introduced the idea of potential exposure.

The introduction of this concept was both the result of ideas developed among ICRP experts as well as NEA and the IAEA experts from a broad range of situations, and at the same time was the beginning of the revision of the radiation protection system in light of this concept. As noted above, this concept is dealt with in the 1990 general recommendations of the ICRP (ICRP60) and is the subject of a specialised publication, the ICRP64 : Potential Exposure. A Framework. (3)

However, to the taste of some, the ICRP46, like many pioneering works, presents a too schemative view of the problem of waste. According to these critics, the emphasis is too heavily placed on the quantitative criteria which have to be respected, furthermore the aggregation of the terms of risk does not help the analytical approach to safety assessment.

One of the essential aspects not sufficiently pointed out in ICRP46 in applying the radiation protection system to solid long lived waste lies in the importance of the design phase, and the way in which all the various bits of information convince us that the long-term safety of disposal will be assured *The licensing issue of most concern is not what formal criteria should be, but rather how one can demonstrate compliance with a given set of criteria. (4)*

It is indeed essential to allow the authorities to make decisions based on precise regulatory requirements. These requirements should not be expressed in terms of probability or exposure levels which are likely to occur in the very long term. Everyone knows that complex models with numerous parameters can be implemented so as to reach a desired conclusion. Rather, the regulatory requirement need to be a technical one which should be chosen in such a way that they are easily checked.

As regards the development of the concept of potential exposure, particularly taking into account the debates which have brought together specialists in radiation protection, waste and safety in the context of the NEA and the IAEA, (5) (6) the emphasis is placed on :

- the distinction between individual and collective risk, the latter being more than merely the sum of individual risks,

- the drawbacks inherent to an over-aggregation of the various components of the risk, and to the consequent need to respect a criterion for design of the waste repository based on limitation of individual risks.

Recent work within INSAG has tended to show that if the exposure were actually to occur and give rise to levels of exposure higher than the 10 mSv - 100 mSv range, considerations in addition to that of simple individual risk would need to be taken into account. Indeed, the authorities would be expected to intervene, changing the scale of the problem. (7)

This reasoning, directly coming from the reflexion on the safety of installations, needs to be adapted to waste disposal as far as the potential over exposure of some individual does not result from the loss of active control

but from a voluntary or unvoluntary intrusion. Moreover, the accidental dispersion of activity in time and space, even at low level, could be considered.

In conclusion, we feel it is important to underline that, essentially, protection of the radioactive waste disposal systems is clearly defined by the principles of the ICRP. However, the numerous current debates must lead to the establishment of concrete ways to apply these principles, particularly for the benefit of the decision makers. This matter has taken on even greater importance since development has begun within the framework of the IAEA, of a convention on radioactive waste which must correspond to an international consensus, yet national regulations are still not in line with each other and correspond to more or less narrow interpretation of the ICRP's intentions.

At the light of this general considerations, and along the line of the conceptual framework established by the ICRP46, the ideas now being developed by waste specialists tend to explore or go deeper into the specific aspects of the problem and integrate with ideas on the following themes :

- to refine the considerations which relate to the ethical aspects of disposal of long-lived waste. This dimension must indeed be taken into consideration because we are transferring a risk to future generations who have not directly benefited from the activities which generated it, and who have not been able to have a say in the decisions which are being made today, or will be made later, on the disposal options. (8)

- to clarify the safety approach by identifying the different stages of this approach in such a way that the decision makers and the public may have « reasonable confidence » in the long term safety of the disposal system. One of the ways to clarify this matter is to divide the future into periods of time. Expressing an increasing degree of uncertainty the further away they are. To propose that the safety criteria be replaced with different « indicators » enabling the impact of disposal to be assessed. (4) (9) (10)

- to ensure that the management methods for the different categories of waste to be assessed are in line with each other high active and long-lived waste, low and medium active short-lived waste and waste from mining and milling of uranium and thorium bearing ores. The latter present particular difficulties due to their vast quantity, the specific radionuclides involved and to their extremely long life span. (11)

I thought it would be interesting to examine these different topics during the IRPA-9 seminar on waste management.

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Ethical Aspects of Long-Lived Waste Disposal

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Introduction

Independent of the long debate on the use of nuclear power, waste management specialists have a clear, unassailable set of environmental goals aimed at protecting the public and workers from any unjustifiable exposure to radiation. It is recognised that releases to the environment must be minimised, operational doses from waste handling kept low, and storage facilities constructed and operated with very high levels of safety. A philosophy of how to make best use of the available resources is embedded into the established principles of the ICRP, requiring justification of practices, limitation of doses and optimisation.

Absolving these responsibilities within the scope of most current waste management practices requires a philosophical or ethical basis, primarily addressing the issue of **intragenerational equity**. Are we being fair and equitable to our present society? Are the levels of protection required in the radiation field compatible with those for other hazards? Could one promote public health and safety better by allocating resources differently? These questions are difficult to answer in practice. However, they can, in principle, be resolved by making the costs and benefits of enhanced radiation protection transparent to today's society which can then, explicitly or implicitly, make the appropriate choices. This approach is possible as long as the same individuals who benefit from any enhanced protection measures also are those who pay for these additional benefits.

The situation is different when we consider the particular case of disposal of long-lived radioactive waste. Properly designed and sited repositories will present only low levels of risk - but these risks are predicted to peak only after many thousands of years. It is obvious, therefore, that this disposal involves the present and immediately following generations investing resources into the protection of far-future individuals. Attention has focused upon this **intergeneration** issue in recent years, leading to intensified debate on all ethical aspects of waste disposal. In this paper, I will try to provide a short overview of recent relevant work, to indicate the ethical principles agreed upon and to highlight the currently most controversial issues.

Does waste disposal really present unique ethical issues?

As mentioned, the specific point leading to emphasis on ethical aspects of waste disposal is that any negative impacts are likely to affect far-future generations who will not directly benefit from the activities producing waste and who will not have shared in the relevant decision making. There are, in fact, other activities today for which the same dilemma arises. Global warming due to CO_2 is the most topical subject, but there are numerous older examples for which the issue of fairness to future generations has

not been recognised explicitly enough. A clear case is the exploitation of natural resources in the earth's crust. The fact that our current voracious consumption of fossil fuels will exhaust in centuries valuable resources which have been built up over millions of years, leaving future generations a fundamentally altered planet, deserves more emphasis in ethical debates. Why is the debate on the long-term aspects of radioactive waste disposal so much more intensive than for other cases where much more dramatic permanent changes in the environment may occur?

One reason, paradoxically, is that the radiotoxicity of nuclear wastes is not permanent but decays with time. This draws attention to the sometimes very long timescales and increases public anxiety; the timescales tend to be compared to human lifetimes and not, for example, to the infinitely long-lasting toxicity of some other wastes. A further reason for so much time and effort being devoted to debate on radioactive waste issues is that, relative to other comparable areas, high funding levels and advanced technological solutions are available for solving nuclear waste disposal problems. The nuclear power industry produces concentrated volumes of long-lived wastes which are small when measured against the energy produced; waste management is a minor (although growing) part of nuclear power costs. Thus, resources have been made available for developing advanced solutions which are hardly practicable for more voluminous wastes (e.g. heavy metal mine tailings) or for more diffuse wastes (e.g. CO_2).

A further reason for particular controversy surrounding the ethics of waste disposal is, of course, that some nuclear power opponents are constantly seeking chinks in the armour of this technology and are therefore interested in hindering any "solution to the waste problem" and in generally raising public anxiety. This is no difficult task since the well documented "nuclear dread" to be found in our westernised societies can be exploited here.

There are also more objective explanations for the topicality of the discussion on ethics and on inter- and intragenerational equity in waste disposal. Repository projects are becoming more mature and both the technical community and the public are conscious of the importance of responsibly taking care of hazardous wastes. This implies that specific guidelines for judging the safety and the acceptability of such projects must be formulated. Accordingly, there has been debate at national and international levels on the principles and criteria which must be applied and this debate has included intensive discussions on the ethical questions involved.

Ethical issues are debated in many national programmes

At a national level there have been numerous meetings and position papers on ethical issues. In Sweden, for example, the advisory council, KASAM, organised a Symposium on the subject in 1987; emphasis was placed in the discussions on the overriding importance of keeping future options open - a topic to which we will return later. In Canada, a workshop was held to give ethical input to the national strategy for disposal

of spent fuel. In Switzerland, as a preliminary to revision of the government regulations governing long-term disposal of radioactive wastes, a seminar was held at which ethical issues were presented by experts from outside the nuclear community. The USA has an extensive literature on the general question of achieving equity between successive generations and this discussion has been taken up by those concerned with radioactive waste management.

The ideas developed in these national programmes and many others have fed into international efforts aimed at achieving consensus on the ethical aspects of waste disposal. As a result, the IAEA produced in 1995, following a long period of iterative comments, an important document entitled "The Principles of Radioactive Waste Management". This paper with its strong emphasis on ethical issues is described in more detail below. A further, equally important document is the "Collective Opinion on the Environmental and Ethical Basis of Geological Disposal" produced by the NEA/IAEA/ EEC in 1995. This consensus view, drafted following a 2-day, wide-ranging workshop on Environmental Aspects of Long-Lived Radioactive Waste Disposal, is also commented upon in more detail below.

Ethical issues highlighted by the IAEA

In the above mentioned IAEA paper on waste management, the following 3 principles are most directly related to issues of ethics:

Principle 3: ***Protection beyond national borders***
Radioactive waste shall be managed in such a way as to assure that possible effects on human health and the environment beyond national borders will also be taken into account.

Principle 4: ***Protection of future generations***
Radioactive waste shall be managed in a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today.

Principle 5: ***Burdens on future generations***
Radioactive waste shall be managed in a way that will not impose burdens on future generations.

Each of those principles is obviously based on sound ethical beliefs and may appear at first sight to be immune to criticism. Each, however was the subject of intensive debate and a source of substantial disagreements. Principle 3, for example, was originally framed so as to prohibit explicitly any state from causing potential radiation exposures in neighbouring countries **higher** than those in the country of origin. Some countries, however, - especially those with no, or limited nuclear programmes - did not believe it ethical to expose their populations even to impacts **equal to or less** than those in the source country.

These states advocated that, ethically, a source country should have to accept any potential exposure limits set by the receiving country. With a no-threshold assumption for dose to risk conversion, however, this proposal could allow any state, by imposing a zero limit, to render disposal activities in neighbouring countries impossible - even if only trivial releases at far future times could be predicted. The final formulation leaves scope for bilateral discussions; it could set a good ethical example for other technologies where localised emissions can have regional or global impacts.

Principle 4 provoked a long debate on acceptable levels of exposures for future generations. Should these exposures be, as stated, "not greater" than those allowed today? Should they be decidedly **lower** than today's levels since future generations will not have the benefits of today's nuclear energy? Should they be allowed to be **higher** because far-future detriments may be (and are in practice) discounted.

The comforting logic of Principle 5 is shaken by the conviction of a large body of persons who believe that imposing burdens on future generations is, in fact, **more ethical** than unnecessarily restricting their freedom of choice. This body of opinion includes the vocal lobby calling for (indefinite) surface storage of radioactive wastes as an alternative to implementing deep repositories. Included amongst the supporters of this view are, however, not only declared opponents of nuclear power (such as those propagating the "guardianship concept") but also more neutral bodies such as the KASAM organisation mentioned above or the Dutch Government, which has recently ruled that all toxic wastes must be retrievably stored.

The Collective Opinion of the NEA/IAEA/EEC

The OCED/NEA has issued over the years selected "collective opinions" intended to record the views of its senior committee of experts on key waste management issues. Somewhat paradoxical - and directly indicative of the defensive battle being fought by the nuclear community - is the sequence of these opinions. A first paper gave the consensus view that radioactive waste disposal **could be carried out safely**. A second recorded and justified the consensus view that **adequate methods were available for assessing repository safety**. The most recent should in fact have preceded the others in that the documented consensus is that the concept of geological waste disposal **rests on a firm ethical basis.**

The collective opinion on ethical aspects was carefully prepared following the previously mentioned workshop on the topic. For the workshop, a background document was prepared listing in an open manner the numerous issues to be tackled and posing the direct question as to whether disposal concepts fit into the framework of sustainable development and ethical responsibility which is accepted today.

In the background text to the Collective Opinion, attention is focused upon "the achievement of intergenerational equity by choosing technologies and strategies which minimise the resource and risk burdens passed to future generations" and it is recognised "that each generation leaves a heritage to posterity involving a mix of burdens and benefits and that today's decisions may foreclose options or open new horizons for the future".

A set of guiding ethical principles is developed in the NEA document; these are broadly similar to the above mentioned principles of the IAEA. Two issues, however, are more strongly emphasised. One is that "a waste management strategy should not be based on a presumption of a stable societal structure for the indefinite future, nor of technological advance". This principle leads to rejection of indefinite storage strategies in favour of geologic disposal concepts offering permanent protection. The second issue discussed more extensively in the Collective Opinion is the wish to ensure that one does "not unduly restrict the freedom of choice of future generations". It is judged that an incremental process, involving development of deep repositories in a stepwise fashion over decades, meets this requirement - even when disposal facilities have no deliberate provisions for waste retrieval following repository closure.

In its summary opinion on the ethical aspects of waste disposal, the Radioactive Waste Management Committee considers that responsibilities to future generations are best discharged by the strategy of geologic disposal and believes that both inter- and intra-generational issues are thereby taken into account. Ethical responsibilities to current generations require also that we should keep in perspective resource deployment in all areas where there is potential for reduction of risks to humans and that the implementation of geological disposal should proceed stepwise with ample opportunity for proper public participation in the decision process.

Consensus in the waste management community - necessary but not sufficient

It is apparent from the above overview that the waste management community has in recent years devoted much time and effort to examination of relevant ethical issues. A broad consensus, reflected in the international documents described, has been arrived at **within the community**. This consensus is that waste disposal concepts and plans do indeed satisfy the requirements of ethical behaviour. In fact, amongst waste management specialists, there is a widespread belief that, because of the specific measures taken to ensure intra- and intergenerational equity, radioactive waste disposal can actually play a pioneering rôle, pointing the way ahead for other environmental activities.

It is, of course, reassuring for those experts in the waste business to learn that numerous outside, neutral specialists in the topic of ethics agree with this positive judgement. It must not, however, be overlooked that agreement on all issues is by no means unanimous. Even within the waste community there are important divergent views. A strong body of opinion would argue that intragenerational equity is certainly not given

in our society which devotes too much of its resources to reducing already trivial levels of radiation to the public rather than focusing on the many obvious major health concerns. There is also a strong argument that potential doses in the far-future should be weighted less than real doses today. Put extremely, would we sacrifice 10 lives today to potentially save 20, or even 100, lives some generations ahead?

Perhaps more important than these continuing internal debates is, however, the fact that the highly ethical picture which the waste management community has of its activities has not been communicated to, or accepted by, the general public. Anti-nuclear groups, sometimes using real examples of poor waste management and thoughtless environmental contamination, are allowed to push the idea that waste disposal is a cheap and nasty, "out-of-sight, out-of-mind" solution. For those involved in waste management, it is comforting to realise that they can go about their business with a clear conscience. But this is not enough. If the waste community sincerely believes in the high ethical standing of its activities, then they should be out there taking the moral "high ground" in this debate. We have a responsibility to future generations which begins right now. We must make today's public aware of our ethical concerns for these future generations and we must continue to work towards implementing the disposal concepts intended to meet those responsibilities.

AN INTEGRATED SAFETY APPROACH AND CONSIDERATION OF SOME BASIC CRITERIA FOR THE DISPOSAL OF RADIOACTIVE WASTE

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Abstract

According to Swedish regulations the nuclear power plant owner is responsible for the safe management and disposal of radioactive waste. Criteria for protection of health and environment are set by the Swedish Radiation Protection Institute (SSI). The Swedish Nuclear Power Inspectorate (SKI) sets corresponding safety criteria and is responsible for review and supervision of the safety of facilities for management of nuclear waste, the nuclear industries R&D programme and the funding system for future costs for nuclear waste disposal. All of these tasks of SKI include assessments of safety.

An assessment of the safety of final disposal of radioactive waste should cover the total disposal system (all barriers of the repository system; operational phase as well as post-closure phase of the repository).

Each step in a handling sequence must consider earlier as well as subsequent steps, so that a waste product resulting from one step is suitable for all subsequent steps.

The safety of a repository for high level waste and spent nuclear fuel should rely on a robust system of multiple barriers. The application of this principle means that the function of one of the barriers might be lost without seriously jeopardizing the safety. Deep geological disposal seems to be the only option available today which complies with this requirement. Even if the assessment relies on an assessment of the total system and not on technical subsystems, technical requirements on the individual barriers must be defined.

The safety case is defined by a set of scenarios. Also possible human actions that could affect the integrity of the repository system should be included. A safety assessment includes an uncertainty analysis of models, scenarios and data. The need for verification/validation of models and the reliance on expert judgement is pointed out as well as the need to accept "reasonable assurance" as the realistic attitude in the demonstration of safety.

Roles of the operators and authorities in Sweden

Operators have a cradle-to-grave responsibility for the environmental effects of their activities. Authorities regulate and monitor those activities, but should not take over the operators' responsibilities.

In Sweden the regulatory system very clearly assigns the responsibility for all measures for the safe management and disposal of nuclear waste to the nuclear plant owner, the waste producer. Occupational and public health and environmental preservation during all phases of nuclear

power operation, including waste management and disposal, must conform with criteria which the Radiation Protection Institute (SSI) sets in accordance with its task to protect man and the environment against harmful effects of radiation. The Swedish Nuclear Power Inspectorate (SKI) sets corresponding safety criteria for the technical performance of every aspect of nuclear power, including waste management, in accordance with its task to supervise that nuclear facilities and activities offer the needed level of safety and also the system for safeguarding nuclear material. These two authorities review and supervise facilities for management of nuclear waste according to their responsibilities. Furthermore, SKI reviews the periodic R&D programme of the nuclear industry that is required by law and supervises the funding system for future costs for nuclear waste disposal.

Basic criteria for the disposal of high level waste

International and national bodies setting criteria agree that in principle, waste management and disposal must correspond with the same safety and protection criteria as of other aspects of nuclear power operation. However, in the context of disposal of long-lived waste, the possibility of potential future exposures complicates assessment. For long time scales, conventional radiation dose criteria must be supplemented, e.g. with criteria addressing release of activity.

The International Commission on Radiological Protection (ICRP) has set fundamental criteria for practices involving radiation, viz. that the practice must be justified, and that concomitant doses must be optimized to be as low as reasonably achievable and in any case below specified dose limits (1). These criteria apply also to management and disposal of waste, although such activities would normally automatically be justified once the practice generating the waste is permitted. However, after closure, a repository for deep geological disposal will not generate any doses at all, if a complete isolation of the waste is achieved. Thus, optimisation and dose limitation must address potential exposures due to unscheduled events, possibly in the far future, as envisaged by ICRP (2). ICRP has developed the concept of potential exposures further (3), albeit not primarily with a view to long-lived waste. It is understood that the possibility of providing further guidance on waste management questions is under discussion within ICRP.

The International Atomic Energy Agency (IAEA) has issued Safety Standards covering basic requirements on radioactive waste management addressing e.g. protection of human health and the environment, the need for establishment of national regulatory systems for radioactive waste management including the need for laws, a clear definition of roles between regulator and implementor, QA-systems etc (4, 5). The OECD/Nuclear Energy Agency (NEA) has published several documents on waste management issues, e.g. collective opinions by experts on final disposal addressing the possibilities for evaluation of the long term safety of disposal of radioactive waste (6) and on the environmental and ethical basis of geological disposal (7). The collective opinion is that the long term safety can be evaluated and that there is an environmental and ethical basis for acceptance of geological disposal.

These publications reflect different aspects of radioactive waste management. Basically they are not in conflict with each other, but should rather be seen as complementary. Various national organisations have also published recommendations and statements. For instance, in 1993 the Nordic Safety and Radiation Protection Authorities published a document on "Disposal of High Level Radioactive Waste. Considerations of Some Basic Criteria" (8). This document is also in

agreement with the views of international organisations as referenced above. The recommendations in the Nordic document are presented here as a rather extensive and fairly operational set of criteria.

Summary of main guidelines and recommendations considered by the Nordic Safety and Radiation Protection Authorities

General objective
The disposal of high level waste shall aim at protecting human health and the environment and at limiting the burdens on future generations.

Long term safety
The risk to human health and the effects on the environment from waste disposal, at any time in the future, shall be low and not greater than would be currently accepted. The judgement of the acceptability of a disposal option shall be based on radiological impact irrespective of any national boundaries.

Burden on future generations
The burden on future generations shall be limited by implementing, at any appropriate time, a safe disposal option that does not rely on long term institutional control or remedial actions as a necessary safety factor.

Justification of practice
No practice involving exposure to radiation should be adopted unless it provides sufficient benefit to the exposed individuals or to society to offset the radiation detriment it causes.

Optimisation
Radiation protection related to waste disposal shall be optimised. In doing so, radiation and risks must be compared and balanced against many other factors that should influence the optimised solution.

Individual protection
Up to reasonably predictable time periods, the radiation doses to individuals from the expected evolution of the disposal system shall be less than 0.1 mSv/year. In addition, the probabilities and consequences of unlikely disruptive events shall be studied, discussed and presented in qualitative terms and whenever practicable, assessed in quantitative terms in relation to the risk of death corresponding to a dose of 0.1 mSv/year.

Long term environmental protection
The radionuclides released from the repository shall not lead to any significant changes in the radiation environment. This implies that the inflow of the disposed radionuclides in the biosphere, averaged over long time periods, shall be low in comparison with the respective inflow of natural alpha emitters.

Safety assessment
Compliance of the overall disposal system with the radiation protection criteria shall be convincingly demonstrated through safety assessments which are based on qualitative judgement and quantitative results from models that are validated as far as possible.

Quality assurance
A quality assurance program for components of the disposal system and for all activities from site confirmation through construction and operation to closure of the disposal facility shall be established to achieve compliance with the design bases and pertinent regulations.

Multibarrier principle
The long term safety of the waste disposal shall be based on passive multiple barriers so that deficiencies in one of the barriers do not substantially impair the overall performance of the disposal system and so that realistic geological changes are likely to only partly affect the system of barriers.

The Swedish regulatory authorities, SKI and SSI, are now developing national regulations

based on the Nordic document and on other relevant documents, *inter alia* publications by ICRP, IAEA and OECD/NEA.

No technical subsystem criteria are defined by the authorities. This would restrict the possibilities of the nuclear power plant owners to shoulder the responsibility given to them by Swedish law to take all necessary actions to ensure safety.

Demonstration of compliance with criteria

Criteria and regulations should be formulated in such a way that compliance can be demonstrated.

Criteria and regulations should aim at defining a required level of safety but in order to be useful they must be set up in such a way that compliance can be demonstrated by accepted methods of assessment. The very long time perspectives in final disposal of HLW and spent fuel will lead to large uncertainties in the assessments. This is a difficulty but does not change the principal requirements on evaluation and demonstration of compliance with given criteria.

Criteria and safety assessments will normally be of both qualitative and quantitative nature. Even such safety assessments that are presented in a quantitative form will to a great extent be based on understandings of qualitative nature (expert judgement). Given the qualitative elements of safety assessments the demonstration of compliance will never be very precise. Reasonable assurance is however required.

Performance assessment as a regulatory tool

Performance assessment methods offer a possibility to assess a technical system in a comprehensive manner. They may be used for assessing safety, identifying R&D needs and may also be valuable as a background for calculating costs for future radioactive waste management.

In order to fulfill their obligations as producers of waste, the nuclear power plant owners have formed a joint company, the Swedish Nuclear Fuel and Radioactive Waste Management Co (SKB), to construct and operate radioactive waste management facilities and to perform the necessary R&D work. The responsibilities given to the nuclear power plant owners and to SKB put a demand on competence and resources. The regulations require the safe operation of facilities, but also the presentation of a comprehensive R&D programme to cover e.g. storage and disposal of spent fuel every third year, and also a funding system to cover the future costs for spent fuel management and disposal. It is very evident that the nuclear power plant owners and the SKB need to have competence in evaluating the safety of nuclear waste facilities and practices. SKB has spent a lot of efforts in developing competence in performance assessment methodology and continues to do so.

Also a regulatory authority in the field of radioactive waste management needs to have competence and tools for evaluating safety and radiation protection. The Swedish Nuclear Power Inspectorate has since many years engaged itself in competence building in performance assessment methodology. SKI as a matter of fact regards performance assessment methods as a regulatory tool to be valuable in the review and assessment of disposal concepts presented by SKB. This will be even more valuable in the expected license application for the encapsulation

facility for spent fuel and the deep geological repository. An application will, according to SKB's plans, within a couple of years be submitted to SKI and referred for comments to SSI and other authorities.

Performance assessment methods may also be valuable in identifying needs for improved understanding. This may be achieved through uncertainty and sensitivity analysis. SKI has recently concluded a performance assessment project addressing the safety of a deep geological repository for spent fuel (SITE-94) (9) and earlier an effort of the same character (Project-90) (10). Project-90 was based on data for a hypothetical site for a deep geological repository for spent fuel. The geological data were chosen to be representative for Swedish geology. In the SITE-94 project real geological data from an actual site (not intended for a repository) was used. The intention with these projects have been to increase the competence of the SKI and to prepare for coming licence applications. The two projects have clearly demonstrated the usefulness of performance assessment methods as a regulatory tool.

The Swedish Radiation Protection Institute is engaged in problems of optimisation under uncertainty. Disposal of high level radioactive waste constitutes an extreme case where hypothetical, potential future exposures of "the public" (which may no longer be formed of humans!) must be balanced against current occupational exposures of the public due to accidents before closure of a repository. Obviously, no single calculation could lead to a definitive answer to this problem. However, SSI is currently concluding an environmental impact assessment project called MKB-95 (11). This has resulted in a computerised tool aimed at sensitivity analyses, so as to illustrate the effects of changing assumptions and parameters in the optimisation process.

Performance assessment methodology is valuable in many respects. It's essential in verification and validation of models of the system. It's valuable in understanding how scenarios and time frames for assessing a repository system should be formulated. It can give quantitative results on the long term performance of a repository as a radiation dose or as an indicator of safety in the very long time perspective. It is needed in the definition of technical requirements on the canister for HLW or spent fuel. Above all it is useful in assessing the long term impact on the environment from a repository system. It must always be born in mind, however, that the quantitative results are to a very great extent based on qualitative evidence, expert judgement etc.

Total systems assessment

The ultimate goal in assessing safety is to cover all aspects of risk. All phases in the development of a waste management system should be evaluated. All facilities and practices being part of the system should be evaluated. In a repository system the total system including all barriers must be evaluated. Also the consequences of human intrusion into a repository should be evaluated.

A complete system for the management of radioactive waste and spent fuel from a nuclear power programme consists of many different activities; e.g. handling, treatment, storage, transport and disposal. Most of these activities require specially designed facilities. It is quite obvious that radiation protection and safety aspects have to be identified and analysed in all these activities and related facilities. The character of risk may differ considerably between e.g. storage and transport but the need for evaluation will always be there.

Some of the effects of waste management will occur during the operation of waste management facilities (doses to workers handling the waste) and some effects may occur in the very far future (during the post closure phase of a deep geological repository). Doses during the operational phase can be monitored and controlled. The potential doses from the post closure phase of a deep geological repository will occur in the very far future and we can not rely on monitoring and control. The system must be so designed that we have confidence in its long term behaviour. However careful we are we must accept that uncertainties exist. The proper balance between risks today (possible to monitor and control) and risks in the very far future (for which we can have no reliance on monitoring and control) is a difficult matter involving not only the uncertainties in the technical system but also philosophical and ethical aspects.

Interdependencies

A waste management system can be subdivided in separate parts and steps. It may be valuable to assess individual parts separately. In evaluating a separate step it must however be recognized that the following steps may be effected by preceding steps. This fact should also be observed in the decision process. A decision may partly or strongly be binding for the following steps.

The management system for nuclear waste, especially HLW and spent fuel, will be a complex system of activities and facilities of sometimes very different character. It is necessary to see the system as being built up by individual parts. Otherwise it will not be possible to plan and construct individual facilities being parts of the total system. In doing so it must however be remembered that these activities and facilities are parts of a total system. This means that interdependencies between all these parts have to be assessed. During the planning and construction phase performance assessment and planning activities will form an iterative process. One example of these interdependencies is the Swedish programme for spent fuel disposal which includes an encapsulation facility for spent fuel and a deep geological repository. In planning for the encapsulation plant it must be remembered that the canisters for spent fuel produced in this facility will eventually be disposed of in the repository. It is quite evident that the canisters must fulfill the requirements given by the repository system. Decisions on the encapsulation plant can not be taken without consideration of the deep geological repository. In some degree of detail the total system (encapsulation facility, canister, geological repository etc) must be assessed (12).

Another example may be the Swedish repository for nuclear waste (intermediate and low level waste) from the operation of the nuclear power plants; the SFR facility, near the Forsmark nuclear power plant. Waste types already produced existed when the repository was planned. The repository thus had to be designed to be able to accept these waste types. New waste types have to conform with the given design of the repository (13).

The effects on the decision process must be observed. A decision to design and construct a facility, e.g. an encapsulation plant, will influence the subsequent steps, e.g. a repository. The level of freedom to take further decisions on the disposal system will decrease. Decisions on individual parts of the disposal system will also influence future costs for the remaining parts and for the total system.

Deep geological disposal

Deep geological disposal of HLW and spent fuel is judged to be justified both from an environmental and ethical point of view.

It is not possible to rely on institutional control of storage for high level waste or spent fuel for very long time. A more reliable solution is needed. Deep geological disposal is the preferred option for high level radioactive waste and for spent fuel. Even if no repository for HLW or spent fuel is yet in operation, this option is under development in many countries. The Radioactive Waste Management Committee (RWMC) of the OECD/Nuclear Energy Agency (NEA) has in a recent statement declared that it is justified, both environmentally and ethically, to continue the development of geological repositories for long-lived radioactive waste (7).

Deep geological disposal typically means a repository excavated in a geological formation (e.g. crystalline rock, salt or clay) at the depth of several hundred meters. The HLW or the spent fuel is encapsulated in long-lived canisters which are deposited in repository. No long-term surveillance is required. Institutional control may however be performed for some period of time.

The reason for choosing deep geological disposal is the reliance we can put on the long term stability of the geology at a carefully chosen site. Arguments for this has recently been given by the NEA, mentioned above, and by the US National Academy of Sciences (14) in its report on the Yucca Mountain project. The SKI and SSI also share this view.

There are however other views. Some experts may find it too early to take decisions on final disposal as alternatives may be available in the future. One example of this is partitioning and transmutation of long-lived radioactive waste, a technique that may transform longlived radionuclides into less harmful nuclides. This will require a lot of work and time and there is no guarantee for success, that is an overall reduction of risk at a reasonable cost. In the NEA statement it was made clear that from an ethical standpoint, geological disposal is preferred over long term storage (waiting for new solutions) which requires long term surveillance.

Multibarrier system

Taking into account the very long-term perspectives and the uncertainties associated with the safety of a repository for long-lived waste, e.g. spent fuel, a robust system is needed. A repository built up by multiple barriers offers a more robust system than a repository relying essentially on a single barrier.

Final disposal of radioactive waste puts demands on the repository system. The intention with a repository system is to isolate the radionuclides and to restrict the release if and when releases occur. For HLW and spent fuel the relevant time perspectives for the functioning of the repository will be in the order of hundreds of thousands of years and more. Even if the geological environment will be stable for very long time periods it is evident that uncertainties in the evolution of the repository system in the long term may be great e.g. because of climate changes and human actions.

A quotation from the Nordic document (8) reads:

"The long term safety of the waste disposal shall be based on passive multiple barriers so that deficiencies in one of the barriers do not substantially impair the overall performance of the disposal system and so that realistic geological changes are likely to only partly affect the system of barriers."

The system should not rely on one single barrier but rather on a set of barriers that together offer a system robust to degrading mechanisms of different kinds, release and transport of radionuclides and to disruptive events. All this is best achieved with a multibarrier system comprising for example the spent fuel matrix, a long-lived canister, back-fill around the canisters in the repository and the geological medium itself. The process for transportation of radionuclides from the repository is dominated by groundwater transport. The depth of the repository must be sufficient to withstand erosion, glaciations etc. A sufficient depth also helps in protecting the repository against possible human actions.

The safety case

To define the safety case for disposal of long lived waste the disposal system and the features, events and processes affecting the system have to be defined. Normally this is done in terms of defining a set of scenarios.

In order to assess a disposal system and to see if regulatory criteria can be met, it is necessary to define the system. In the case of a repository the multibarrier system must be defined. The different barriers (waste matrix/spent fuel, canister, back-fill, geological medium etc) and their properties must be described. Also the system boundaries must be defined. The behaviour of the system is affected by features, events and processes (FEPs) within the system but also external FEPs may influence the system. The exact definition of the boundaries is a matter of judgement. Normally climatic changes, major disruptive event (e.g. faulting) etc are regarded as external FEPs.

The development of the repository system can not be observed and verified in the very long time perspective. A basic understanding of the system and the FEPs that may affect the system is needed.

FEPs affecting the system are normally combined and described as a scenario. Different combinations constitute different scenarios. Because of the uncertainties, mainly due to the very long term perspectives, a set of scenarios is needed to cover the development of the repository system.

Safety assessment and uncertainty

Safety assessments for very long term periods will always be uncertain. If we know and understand the reasons for uncertainty and if we can quantify the uncertainty much is achieved in assessing safety.

Safety assessment methods for the normal operation of a nuclear facility can be based on actual experience and statistics. In addition to deterministic evaluation, probabilistic methods utilising existing data can be used. Corrective measures can be initiated if needed. This also includes a repository in its operational phase (before closure).

For the post-closure phase of a repository, monitoring and control are not possible in the very long time perspectives. Assessments must be based on a genuine understanding of the repository system and its environment, including also the development of the system in the very long term perspective. The underlying knowledge base is both of qualitative and quantitative nature.

The definition of the safety case can be seen as a part of the safety assessment. Other parts are then related to the modelling and consequence analysis.

Modelling includes the formulation of conceptual models of the disposal system (e.g. the repository and the geologic medium) and the translation of the conceptual models to computer codes. The input to the codes and the output are quantitative. If the models and the codes are good representations of what they are supposed to represent then the results will be good representations of the consequences. Are the models and codes good representations? This is a key question. Much effort has been spent on development of performance assessment methodology. Despite the difficulties there is confidence in the possibility to evaluate long term safety. Expert committees of the NEA, IAEA and the European Communities in 1991 (6) stated as a collective opinion that long-term safety in radioactive waste disposal can be evaluated.

As mentioned earlier the uncertainties in the safety assessments increase with time. This issue has been discussed in an IAEA document that also addresses the concept of safety indicators (15). It should be remembered that already in time perspectives of a hundred to a thousand years considerable changes in climate, living habits etc may occur. The quantitative results should for the long time perspectives not be regarded as exact representations of the effects on human health and the environment but rather as indicators of safety. Even if this is true, calculations covering long time periods are meaningful for certain purposes, such as ranking of alternative disposal options. The quantitative assessments will help in understanding the uncertainties and sensitivities in different parameters and couplings of processes within the system, and will thus help in understanding the long term behaviour of the system. A good understanding of the long term behaviour of the system is expected to be of value also for understanding of shorter time periods. For the performance assessment of the repository system, it appears that the inevitable cut-off in time should come considerably later than the time frame (say, 10 000 years) over which calculations of doses to humans retain any relevance. This view is supported by the fact that releases from a deep geological repository may increase and peak well beyond 10 000 years. It is however important to keep two things in mind: the results from calculations for such long time periods should not be regarded as a true representation of risk but rather as indicators of safety; and the risk associated with releases of long-lived radionuclides well after the short-lived gamma emitters have decayed should be compared to risks associated with stable chemotoxic nuclides.

Conclusions

In the development and review of a system for waste management and disposal it is necessary to assess the total system. All phases should be analysed: treatment, storage, transport, deposition of waste and the long term safety of the repository after closure. The complete repository system should be assessed including man made structures and the geological environment and the biosphere.

A well organised and well structured comprehensive assessment of the waste management and

disposal system is very useful in relation to defining scenarios and time frames for evaluation of the repository performance as well as in defining technical criteria for selection of a repository site. It will also be valuable in defining the technical criteria for a canister for spent fuel or high level waste. Performance assessment methods will make it possible to evaluate the long term performance of a system for final disposal of radioactive waste.

It should always be born in mind, however, that the quantitative results from performance assessment calculations to a great part are based on evidence of qualitative nature. For the long term the results from performance assessment calculations should be regarded as indicators of safety rather than exact representations of risk.

From a regulatory perspective performance assessment methods are very useful and needed in understanding if a proposed system complies with given criteria. The answer to this may not always be simple but reasonable assurance is needed.

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MANAGEMENT OF WASTE FROM THE MINING AND MILLING OF URANIUM AND THORIUM BEARING ORES

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ABSTRACT

Mining and processing of uranium and thorium bearing ores for the extraction of uranium and other minerals gives rise to the production of various radioactively contaminated wastes and other re-cyclable materials. The wastes are characterised generally by their bulk nature , their low specific activity and the long radioactive half-lives of several of the radionuclides involved. The contaminated re-cyclable materials are mostly steels, timber and plastic and the contaminants can be of higher specific activity due to various chemical and physical concentration processes. The mining and minerals processing residues , essentially waste rock and tailings, present particular waste management problems. Their bulk nature precludes underground disposal and the long radioactive half-lives present problems of assuring their long term containment. In addition, the residues contain radium at elevated levels, corresponding generally to that in the original mined ore, and presenting a source of radon progeny. The nature of mining waste is such that they are often an inexpensive source of raw materials for use in building materials. Their diversion to such use must be considered in respect of both external and radon progeny exposure.

The paper describes the nature and characteristics of the wastes and re-cyclable materials arising from mining and minerals processing activities. It discusses the various management options adopted for these wastes in various countries and addresses the process of hazard assessment for these wastes and the associated safety criteria. The paper also discusses the consideration of these wastes in terms of the RADWASS programme and the principles of radioactive waste management embodied in the safety fundamentals which form the basis for the RADWASS programme.

1. The Nature and Characteristics of Radioactive Wastes From the Mining Industry

Uranium and thorium are present in the earth's crust at average mass concentrations of around ten parts per million. They can, however, be concentrated in mineral deposits and concentrations can range up to tens of percent. Such concentrations are exceptional, but extensive deposits are found with concentrations hundreds of times the average concentration. Both uranium and thorium are naturally occurring radioactive materials and in nature are comprised of several isotopes. The parent radionuclides, U-238, U-235 and Th-232 are primordial radionuclides with half-lives of 4 500 million, 700 million and 14 000 million years. They decay by alpha particle emission and give rise to subsequent radioactive decay series comprised of various radioactive progeny as illustrated in Table 1. The members of the decay series posses significantly differing physical and radiological characteristics which greatly influence the potential radiological hazards associated with wastes arising from mining and minerals processing activities.

Table 1. Natural Decay Series

URANIUM - 238 DECAY SERIES			THORIUM - 232 DECAY SERIES			URANIUM - 235 DECAY SERIES		
NUCLIDE	DECAY	HALF LIFE	NUCLIDE	DECAY	HALF LIFE	NUCLIDE	DECAY	HALF LIFE
U - 238	ALPHA	4,5 E9 y	Th - 232	ALPHA	1,4 E10 y	U - 235	ALPHA*	7,1 E8 y
Th - 234	BETA*	24 d	Ra - 228	BETA	6,7 y	Th - 231	BETA*	24,6 h
Pa - 234	BETA*	1,17 m	Ac - 228	BETA*	6,13 h	Pa - 231	ALPHA*	3,43 E4 y
U - 234	ALPHA*	2,4 E5 y	Th - 228	ALPHA*	1,9 y	Ac - 227	ALPHA*	22 y
Th - 230	ALPHA*	8 E4 y	Ra - 224	ALPHA*	3,64 d	Fr - 223	BETA*	21 m
Ra - 226	ALPHA*	1,6 E3 y	Rn - 220	ALPHA*	55 s	Ra - 223	ALPHA*	11,2 d
Rn - 222	ALPHA*	3,8 d	Po - 216	ALPHA	0,15 s	Rn - 219	ALPHA*	3,92 s
Po - 218	ALPHA	3 m	Pb - 212	BETA*	10,64 h	Po - 215	ALPHA	1,8 E-3 s
Pb - 214	BETA*	26,8 m	Bi - 212	BETA*	60,6 m	Pb - 211	BETA*	36 m
Bi - 214	BETA*	19,7 m	Po - 212	ALPHA	304 E-9 s	Bi - 211	ALPHA*	2,16 m
Po - 214	ALPHA*	164 E-6 s	Pb - 208	STABLE		Tl - 207	BETA*	4,79 m
Pb - 210	BETA*	21 y				Pb - 207	STABLE	
Bi - 210	BETA	5 d						
Po - 210	ALPHA*	138 d						
Pb - 206	STABLE							
* DENOTES ACCOMPANYING GAMMA EMISSION								

Since the advent of nuclear weapons development and the evolution of nuclear power generation, uranium as a mineral has been widely extracted. Thorium has been exploited to a much lesser degree, however, numerous other minerals are extracted from ore bodies containing elevated levels of uranium and thorium and giving rise to radioactive wastes. The latter include, inter alia, monazite, ilmanite, rutile, zircon baddelyite, phosphates, copper, lead, zinc, silver, flourospar, gold and coal. Radium rich mineral deposits have been exploited since the first quarter of the century and uranium mining activities commenced in the 1940's and accelerated considerably in the early 1950's. To date over thirty countries have produced in excess of 30 tonnes with six countries producing greater than 100 000 tonnes in the period up to the early 1990's as illustrated in Table 2. Greater than 1 800 000 tonnes of uranium have been extracted world-wide and projections are that by the turn of the century this will be in excess of 2 000 000 tonnes.

The mining and minerals extraction process entails mining of the ore body, either underground or by way of an open pit, crushing and milling the ore, possibly concentration of the minerals by various physical processes and extraction of the mineral being exploited by chemical processes. In addition, a limited number of in-situ leaching activities have been undertaken. More often than not these steps are followed by various purification processes. The outcome of the overall process is the mineral product together with various waste residues which will be combinations of:

- waste rock

  rock removed from the mine in winning the ore but which contains mineral concentrations too low to economically extract.

- tailings

  the crushed and milled rock from which the mineral has been extracted together with any chemicals and fluids remaining after the extraction process.

- chemical residues

  spent ion exchange resins, solvent extraction sludges, active charcoal etc.

- other waste

  In addition, various mine and plant consumables such as filters, catalytic saddles, timber etc. and scrap arisings such as metal and plastic become contaminated giving rise to radioactive wastes.

Table 2. World Production of Uranium up to 1992

COUNTRY	PRODUCTION 000 tonnes	COUNTRY	PRODUCTION 000 tonnes	COUNTRY	PRODUCTION 000 tonnes
USA	**339**	**Namibia**	**53,1**	**Poland**	**1,00**
Canada	**258**	**Ukraine**	**45,1**	**Brazil**	**0,94**
Germany	**218**	**Zaire**	**26,6**	**Madagascar**	**0,77**
South Africa	**143**	**Bulgaria**	**21,9**	**Pakistan**	**0,66**
Czech Rep.	**102**	**Gabon**	**21,4**	**Belgium**	**0,51**
Russia	**101**	**Tadjekhistan**	**20**	**Slovenia**	**0,38**
Kasakhstan	**81,7**	**Hungary**	**19,9**	**Sweden**	**0,20**
China	**79,6**	**Romania**	**18,4**	**Japan**	**0,09**
France	**70**	**India**	**5,78**	**Mexico**	**0,04**
Uzbeckhistan	**61**	**Spain**	**3,78**	**Finland**	**0,03**
Niger	**56,8**	**Portugal**	**3,56**		
Australia	**54,2**	**Argentina**	**2,15**	**TOTAL**	**1 810**

As a primary product, uranium bearing ores are generally exploitable at concentrations in excess of several hundred parts per million. As a by-product, however, uranium can be economically extracted at concentrations of around one to two hundred parts per million as in South African gold mines. At the extreme, uranium is extracted from copper and phosphate ore deposits at concentrations of around fifty parts per million. A representative average grade in the region of 0,1% or one thousand parts per million indicates that associated with

the world wide production to the end of the current century will result somewhere in the region of 2 000 million tonnes of mill tailings from primary uranium producers.

The total arising of other wastes are less well documented, however, the estimated waste arisings for South African gold/uranium mining operations together with the characteristics of such wastes are indicated in Tables 3, 4 and 5.

Table 3. Illustrative Concentration Levels in a Selection of Waste Streams From South African Mining Operations

WASTE TYPE	TOTAL ACTIVITY Bq g^{-1}
Gold tailings	2 - 200 +
Calcine	30 - 2 000
Calcine Scale	200 - 300 000
Pyrites	< 1 - 400
Pyrite scales	1 - 25 000
ADU Section Scales	up to 30 000
Rubber lined components e.g. pipes	up to 200
Contaminated Soils	< 0,2 - 50 000
PMC scales	up to 350 000
Gypsum	up to 10
Waste rock	< 0,1 - 200

Table 4. Annual Waste Production Estimates From South African Mining Operations

MINES	TAILINGS M tonnes	WASTE ROCK M tonnes	SCRAP METAL SCO-I[2] tonnes	SCRAP METAL > SCO-I[3] tonnes
Gold (29)	121	37	15 000	2 000
Tailings re-cycling[1]	48	0,1	1 000	350
Other	60	Unknown	6 000	400

1. Reprocesses already existing tailings dams it does not add to the growing inventory of tailings.
2. Estimates of SCO-1 contaminated scrap are based on the indications that about 10% of all ferrous scrap is contaminated. More accurate statistics are not available at present. Use is made of the Surface Contaminated Object (SCO) classification criteria embodied in the IAEA Regulations for the Safe Transport of Radioactive Material.
3. Estimates of scrap contaminated above SCO-1 levels are based on actual data statistics in waste reports.

Table 5. Demolition wastes

Decommissioning and Demolition

There are a number of contaminated plants on mines awaiting demolition and site rehabilitation which will give rise to significant amounts of waste arisings in the future. The majority of these plants are associated with the gold mines and include acid, pyrite, calcine, uranium and gold plants. About 40 contaminated acid, pyrite, calcine, and uranium plants and their sites require to be demolished and rehabilitated. In many cases the plants have been completely or partially stripped but the sites have not been decontaminated and rehabilitated.

Figures from recent demolition activities indicate the following approximate quantities per site:

• metal SCO-1:	up to 10 000 tonnes
• metal > SCO-1:	100 - 200 tonnes
• Rubber, resins, plastics :	a few hundred tonnes
• Contaminated soil and rubble (< 1 000 Bq g^{-1} LLA):	up to 100 000 tonnes
• Contaminated soil, rubble and scales (> 1 000 Bq g^{-1} LLA):	a few hundred tonnes

In the undisturbed ore body, the uranium and thorium isotopes will exist in their natural ratios and the radioactive progeny will be present within the ore body in secular equilibrium with the parent radionuclides. If some separation has taken place for instance by the intrusion of water into the ore body, this state of equilibrium may have been disturbed. The mining and minerals processing activities significantly alter this state of equilibrium, primarily separating the uranium or thorium from their decay series progeny. Depending upon the extraction efficiency, the bulk waste arisings, i.e. the tailings, will contain only a few percent of the original uranium or thorium concentration. It will, however, still contain most of the radioactive progeny. In the case of uranium mining, this means that the radioactive content of the tailings will be dominated by Th - 230 with a radioactive half life of 80 000 years. Should the ore being processed contain elevated levels of uranium, which are, however, not economically viable for extraction at that time, clearly the majority of this uranium would be present in the tailings and the longer half life of U - 238 would in such cases predominate.

The various other wastes arise due to contamination of materials during the mining and extraction processes. This may be contamination by ore or can arise due to the selective concentration of particular species from the decay series by chemical and physical processes. In particular, radium is found to be concentrated and taken up in several waste arisings.

In terms of overall waste management, due consideration must also be given to the mine following closure. Both underground and surface mining operations result in the original ore body being in an altered state which may have given rise to exposure pathways that did not exist prior to mining activities. Such pathways require consideration in closing-out of the facility.

2. Radiation Hazards Associated With Mining Waste and Management Options

Possible Hazards

In view of the bulk nature of mining and processing wastes and the limited considerations that were originally given to their long term management, disposal options are generally limited to some form of in-situ stabilisation. Whilst there are some significant differences between uranium mining methods and extraction technologies and the resulting wastes, the radiation hazards associated with their management are generally dominated by Th - 230, Ra - 226, Rn - 222 and its short lived progeny and the longer lived radon progeny. In addition, the wastes associated with mining and processing of thorium bearing ores and other radioactive ores all have their own particular characteristics. However, consideration of uranium mining wastes covers all the potential issues associated with the latter.

There are numerous mechanisms by which radionuclides associated with mining and minerals processing waste can enter or give rise to exposure pathways to humans. These may arise during both the operational and post close-out phases and are elaborated below:

- Airborne dispersion from tailings piles and waste rock dumps of particulate material or radon emanating from the piles. This may arise during the operational phase and following close-out if cover material is eroded due to precipitation, wind or frost damage.

- Seepage from tailings and waste rock piles can transport dissolved radionuclides into ground water. The dissolution of nuclides is often enhanced by the presence of pyrite causing the wastes to be acid generating. Subsequent migration can cause the nuclides to enter exposure pathways involving the use of the groundwater or discharge of the groundwater to surface.

- Seepage at ground level may give rise to contamination of surface waters and is also a potential source for airborne dispersion.

- Intrusion on sites were mining and minerals processing residues are present by way of construction, agricultural and recreational activities. The associated exposure pathways include inhalation - both radon progeny and long lived alpha emitting species, direct exposure and ingestion due to uptake or contamination of feed stuffs.

- Diversion of mining residues from operating or closed-out sites. Tailings material often provides an inexpensive source of land fill or construction sand and waste rock an equally inexpensive source for the manufacture of aggregate for various construction purposes. Scrap materials including steel, timber and plastic are often desirable sources of raw material for re-cycling operations and give rise to a variety of exposure pathways depending on the re-cycling operation and the subsequent use of the re-cycled material.

- Closed-out mines may become flooded following operations and give rise to sources of enhance nuclide migration into surface and ground waters. They may also be a source of radon emanation into the atmosphere.

- The mechanisms of migration and transportation included above can also be affected by the impacts of natural phenomena such as earthquakes, floods etc. which may cause damage to engineered and natural containment barriers.

Waste Management Options

Waste management associated with operating mining and minerals processing facilities generally entails the siting, design and operation of tailings dams, siting and placing of waste rock piles, control of surface drainage, control and discharge of mine water, suppression of airborne dispersion from the site and management of consumable waste arisings such as chemical processing residues, scrap steel and timber.

At the time of mine closure, rehabilitation activities generally involve the dismantling of plant, demolition of buildings, stabilisation of tailings and waste rock piles, sealing of the mine and general site rehabilitation. Practice in tailings management has involved relocation, in-situ stabilisation and cover. The latter is aimed at preventing radon emanation, moisture infiltration, gamma radiation, oxidation, intrusion and erosion. Various methods of tailings cover have been used and depend on the local geological, climatological, and geographic characteristics. Often clay, sand and various grades of rock are used, vegetation is also appropriate in some locations and water cover has been used. Mines have been backfilled with mining residues and open pits have been backfilled, re-contoured or flooded.

In view of the limited attention paid to waste management on many sites during the early years of uranium mining, some countries have experienced the contamination of other properties in the vicinity of mining and minerals processing sites. This has been caused by the diversion of materials or from the uncontrolled dispersion of material from the sites. It has resulted in extensive rehabilitation efforts having to have been expended in the cleaning of these properties.

Where mining and minerals processing activities have given rise to the contamination of groundwater various engineered approaches have been adopted. These include isolation of the contaminated groundwater compartments and treatment of the contaminated water and increased natural flushing. In both instances, some institutional control on the site and its environs is necessary, such measures may be required over relatively long time scales - up to hundreds of years.

3. Hazard Assessment and Safety Criteria

Operating Phase

During the operating phase, consideration must be given to; which mining activities will be subject to regulatory control and which can be exempted, the control of effluent discharges from regulated facilities, the management of waste arisings stored or intended for disposal on the site and clearance of materials from the site. All these aspects can reasonably be addressed by deterministic hazard assessments and compared against laid down dose limits and subjected to optimisation assessments. The exception in this regard may be contamination of groundwater, where, due to the uncertainties in groundwater movement and

transport of contaminants into and by the groundwater, elements of the assessment may have to be subject to probabilistic considerations.

Exemption gives rise to particular problems due to the existence of natural background levels of uranium and thorium, together with their progeny, and the variable nature of these background levels. This is compounded by the presence of radium, which, at relatively modestly elevated levels above average levels can, in building related scenarios, give rise to dose levels well in excess of recommended exemption levels of dose and in fact in excess of public dose limits. Exemption levels quoted in the Basic Safety Standards Ref [1] for natural uranium, natural thorium and radium-226 respectively are; 1 $Bq.g^{-1}$, 1 $Bq.g^{-1}$ and 10 $Bq.g^{-1}$. It is explicitly stated, however, that these values are not applicable to bulk materials. Control over building on land with radium levels greater than around 0,1 Bq $.g^{-1}$ of radium-226 are recommended in some countries Ref [2] although modelling of the resulting exposures is dependent on numerous factors including soil porosity, building construction method and building ventilation rates. Many countries adopt values in the range of 0,2 $Bq.g^{-1}$ as exemption levels. These correspond to the values laid down in the United States for clean-up of uranium mill tailings and contaminated vicinity properties associated with uranium mining and milling operations Ref [3, 4].

The concept of effluent discharge control in respect of mining and minerals processing facilities is complicated by the range of sources, the extended nature of some sources, such as tailings piles, the various transport pathways and mechanisms and the existence of discrete controlled discharges together with those less amenable to control such as surface run-off and wind-borne dispersion. Nevertheless, assessments can be carried out for comparison with laid down criteria and release control mechanisms are possible. Dose limits vary between different countries spanning the range from a fraction of 1mSv for discrete facilities to 5mSv Ref [5].

The management of waste on mining sites is generally dictated by the tailings and waste rock management programmes. The associated hazard assessment is more related to the eventual close-out of the mine and will be addressed later. One factor giving rise to problems in respect of on-site waste management is the handling of other wastes exhibiting concentrated levels of contamination. Mine tailings contain radium-226 concentrations generally up to around 10 $Bq.g^{-1}$ and in exceptional cases higher. Waste with concentrations up to tens of thousands of $Bq.g^{-1}$ can arise and pose particular waste management problems. Such material would generally not be acceptable for disposal in near surface repositories Ref [6], rather deep geological disposal being indicated. Practice in some mining operations has been to dispose of such materials within tailings dams or waste rock piles and, at this stage, there is no consistent management strategy for such materials. The classification of these types of waste also gives rise to problems, particularly when the waste is in the form of non-active, components contaminated with relatively thick contaminated surface deposits such as scales.

Clearance of materials from mining and minerals processing operations gives rise to many of the issues raised in terms of exemption. Materials which could be considered as candidates for clearance from mining operations could include tailings for land fill and construction purposes, waste rock for building aggregate and scrap steel, timber and plastic. On the assumption that the former materials if released will in fact be used for construction purposes, the selection of an appropriate dose criteria poses problems. Selecting a value at a fraction of

the recommended public limit would result in a very restrictive derived levels for radium-226 in the material and basically preclude the clearance of any material with levels marginally above "average" levels of around 0,03 - 0,05 $Bq.g^{-1}$. The current practice being adopted in South Africa is to employ a process of conditional exemption whereby the end use of cleared material must be specified and must not be use in building construction. This process has many associated procedural and legal complications. Similar problems arise also with the "clearance" of former mining land.

The clearance of scrap materials is also complicated by the fact that a large fraction of contaminated materials arise from the deposition of scales on material and the concept of surface contamination as a criteria for clearance is questionable. Also, as a criteria for release, the concept of bulk contamination, taking into consideration the mass of the non-active object raises questions as invariably subsequent handling and processing of the material results in the scales being removed and giving rise to loose radioactive material of much higher unit density. These problems are presently addressed in South Africa by assuming that all surface deposited material is "loose" and by applying surface contamination levels for clearance at the lower end of the spectrum of values used in various countries Ref [7].

End of Mine Life

For many operations which commenced in the 1940's and 1950's, very little consideration was given to closure and to the potential radiological hazards that may be associated with the mining and processing residues. The potential hazards have become apparent within the past twenty years and hence the associated efforts to rehabilitate mines and sites and to implement engineered waste management solutions. The approach generally adopted is to demolish the buildings and to clear the site of contamination with a view to allowing unrestricted future use of the site. The tailings and waste rock piles are then effectively the resulting radioactive waste requiring disposal. The solutions adopted to date have included in-situ stabilisation, removal to engineered on surface containment cells, backfilling into mines and pits and disposal in water bodies. These solutions have necessarily involved the conduct of hazard assessments to demonstrate their adequacy. Such assessment has involved both deterministic and probabilistic modelling techniques both separately and complimentary. The most difficult challenges facing acceptance of the assessments is selection of the time periods required for demonstration of adequacy combined with predicting the performance of the engineered covers, migration barriers and erosion control features over long periods. The impact assessment of the closed mine poses similar problems.

Design criteria for performance demonstration have generally been limited to between two hundred and a thousand years and radiological criteria have often been set in terms of radon emanation rate from the surface of covers, gamma dose rates and ground water contamination levels. Typical levels being in the range of 700 $mBq.m^2$ s, 0,2 $\mu Sv.-h^{-1}$ and 0,2 $Bq.l^{-1}$ Ra-226 Ref [7].

Because of the need to prevent intrusion into the stabilised tailings , the need to prevent removal of the material for construction purposes and because of the extensive nature of the wastes, and the vulnerability of the covers and barriers, the need for ongoing institution control over these facilities appears to be inevitable.

4. Mining Wastes in the Context of the RADWASS Fundamentals

In view of the relatively low specific activity of mining wastes and the very large bulk volumes, mining wastes have often been viewed in a different light to radioactive waste arising from other parts of the nuclear fuel cycle. The approaches currently being adopted to their management, however, go a long way to respecting the fundamental principles of radioactive waste management adopted by the Member States of the International Atomic Energy Agency.

The principle of protecting human health is respected by adoption of safety standards in line with those recommended by ICRP Ref [8] and endorsed in the Basic Safety Standards Ref [1]. The second principle of protecting the environment is particularly pertinent when considering uranium mill tailings which are often acid generating and contain other non-radioactive potential pollutants such as arsenic, barium, cadmium, chromium, lead and several others. Again the rehabilitation and containment processes often take these other pollutants into account and provide for their isolation from the biosphere. Protection beyond national borders will generally be achieved by the engineered solutions being implemented, although, if mining facilities are close to national borders, the need for demonstration remains. The fourth and fifth principles dealing with protection of future generations and the burden on future generations do raise some questions with mining waste because of the need for perpetual institutional control and the inevitability of the eventual failure of containment structures over the time scale associated with the radioactive half-lives of the radionuclides involved. Nevertheless, the concerns are ameliorated to an extent by the relatively passive nature of the required institutional controls and the relative ease to repair any degradation of the covers.

Compliance with the sixth principle of managing the wastes within an appropriate national legal framework varies considerably between countries. Countries within which a nuclear industry operates often control mining wastes in terms of their nuclear regulatory programme. Such control, however, is often limited to uranium mining wastes and does not cover mining of radioactive ores from which uranium is not extracted. Principles seven and eight requiring radioactive waste generation to be minimised and interdependencies to be considered are features of more recent mining operations and those subject to good regulatory control regimes. The ninth and final principle addressing safety of waste management practise is less of an issue with mining operations due to the low specific activity and the generally chronic nature of the associated hazards.

In conclusion, mining wastes, whilst being characterised by relatively low specific activity and giving rise to chronic modes of exposure, can cause exposures well in excess of public dose limits and can give rise to substantial collective doses. Because of the attractive nature of mining residues for construction purposes and the utility of many of the waste arisings for re-cycling, and the disposal of such wastes often on the surface, particular attention needs to be given to control over their diversion. In addition, due to the relatively significant radiation doses that can arise from radon progeny clearance levels for bulk material must be carefully selected and administered. Because of the long radioactive half lives associated with mining waste, the concept of waste management options involving eventual removal of restrictions

on the disposal sites cannot be realised. The need for consideration of all mining wastes involving elevated levels of natural radioactivity to be managed effectively and not only those associated with uranium mining require due consideration.

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Symposium S-14:

„Safety of Large Radiaton Sources“

Chair and Organizer: D. Beninson
Co-authors: Lubenau, P. Ortiz, J. Croft

Safety of Large Radiation Sources

In this Seminar the main protection problems related to the use of large radiactive sources will be examined, in an interactive discussion with the participants.

The subjects to be covered are:

1. **Types and uses of large radiation sources.**
2. **Protection in the case of normal exposures, for fixed and mobile sources.**
3. **Protection against potential exposures for fixed and mobile sources.**
4. **Special protection problems; ozone and other gases production, corrosion, etc.**
5. **Good operation practice.**
6. **Regulatory aspects of the use of large radiation sources.**

The discussion will be carried out in the framework of the ICRP recommendations and with reference to examples of regulatory practice.

Retrospective Dose Assessment

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Introduction

Radiation dose reconstruction has become a significant environmental sciences specialty over the past decade. Models to integrate limited radiological measurements have been developed and are being validated and improved. For external exposures, our main tools are computer simulations of radiation output from a "source term", be it the A-bombs dropped on Japan, or a hospital fluoroscope. Validation without actually duplicating the exposure conditions leaves a level of uncertainty that is difficult to reduce. The second major class of radiation dose reconstructions relates to those doses absorbed as a consequence of radionuclide intake. Here too, uncertainties exist as to the exact "dosage" quantity and quality and pharmacokinetic behavior in an individual or population are usually not available. I wish to restrict my remarks to this latter case, but also realize that some of what I have to say may be germane to external exposure problems.

Simply stated, my thesis is that the reconstruction of doses from the outside inward, can be assisted by personal biological radiation dosimetry from the inside out. I believe that the case can be made for the fact that the deficiencies and limitations of both sets of tools, environmental and biological dosimetry, are in different dimensions and that the strengths of one approach can assist in reducing the weaknesses in the other. Ideally, an integration of biological and environmental dosimetry for the same case will be synergistic and the resultant "collective" uncertainty will be minimized. This is particularly true of very low doses, where the sensitivity and detection limits of available techniques are stretched to their utmost. For radionuclide-related dose reconstruction, environmental measurements of concentrations and pathways infrequently sufficient to determine a biological source term, the quantity potentially taken in by the individual of population. Adding the appropriate biological, demographic, and environmental factors generates the consequent dose estimate. In the best of all worlds, the findings might be confirmed by actually measuring excreta, body-burdens, and/or biological indicators of radiation damage. We rarely have been able to realize this ideal, because those exposed may have had doses that were below the sensitivity threshold for the available techniques.

I would like to review some of these techniques and present an argument for increased research and development of biological indicators which may provide more sensitive output. To keep this in perspective, I feel that there is room for improvement and advances in the environmental tools as well as for biological dosimeters. A host of sophisticated computer programs now provide a platform for rapid integration of data and integrated output of results. Atmospheric and environmental transport models have continued their evolution to powerful tools.

Remote sensing of biological indicators, such as infrared reflection and emission signatures of radiation damaged plants are now available. Aerial surveys, yielding planar spatial isoconcentrationn gradients of surface contamination are being improved by integrating global-positioning-systems with focal emission rate data. Faster processing and more sensitive, stable detectors are also providing improvements. On the ground, improved sampling techniques, provide concentration gradients and temporal and spatial profiles important for reconstructing prior situations. Thermoluminescent dosimetry of stable signals form radiation exposed geologic and biologic samples add to our armament. All these techniques present opportunities for dynamic improvement. Added to these tools are the retrospective, historic description and documentation of events and measurements of past exposure related information. At times, I am not too certain if there is adequate support and commitment for the future development activities. I add my personal plea for a more integrated international program to improve sensitivity of techniques, to calibrate and "certify" the techniques and to explore and develop new methods.

Biological Dosimeters

The object of this overview is to indicate some important roles for biological indicators of radiation exposure in our overall planning for modern radiation retrospective dosimetry. These are either bioassays, direct physical measurements of radiation and radioactivity, or they are biological signals which are derived from living systems which record radiation effects.

Biological assays of radioactivity are simply collections of excreta from body burdens, the data from which when input to pharmacokinetic models provide and indirect estimate of the body burden and hence the associated radiation dose. This is particularly important for alpha emitting radionuclides whose emanations are too difficult or impossible to detect in-vivo. Whole body and partial body counting techniques are especially sensitive for body burdens of gamma-emitting radionuclides. Energetic beta-emitters can be assayed using bremmstrahlung counting and appropriate standards. (An important contribution from Chelyabinsk, Russia will illustrate this tool). Post mortem tissue sampling, and in some cases, tissue biopsy, (e.g. tooth), can also provide direct input into body burden estimation.

A renewed interest has developed for the use of electron spin resonance of biological crystals which had received radiation doses. Because of the long stability of the signal in exposed crystals the technique has even been applied to dating of mastodon tusks, using the integrated natural background radiation signal over many thousand years. More practically, it has been used to measure the absorbed dose signal from the jaw of a survivor of the a-bombing of Hiroshima 39 years ago. The crude device had poor sensitivity, (i.e. 1 Gy), but did demonstrate the feasibility of the method. For small chips of tooth enamel, a sensitivity of about 0.1 Gy is reported and a ten-fold improvement is quite possible. (We shall hear more of this in today's Symposium).

I would like to make a plea here for a concentrated effort to truly exploit this technique to its limit, to effectively engineer and develop both an in-vivo and an in-vitro methodology that is internationally calibrated and accepted. My journeys through this technique over the past eight years have proven to me that many well-meaning scientists working in competition have not yet assembled the critical mass needed to achieve this goal in a reasonable time. For the dose reconstructions planned, as well as for accident management needs, there is no reason to continue in present manner. I view this techniques as a biological dosimeter as useful as the thermoluminescent dosimetry as proven to be for personal and environmental dose documentation. This is one of my challenges to you.

Cellular markers or indirect indicators of radiation dose are either cytologic indicators or variations on measurement of somatic gene mutations. The advances in cell and molecular biology have not in my opinion carried adequately over to the arena of biological dosimetry. Years ago we were promised an automated **chromosome aberration scoring** device. We are still waiting for a dependable karyotype scoring device. For the most part, the scoring of chromosomal aberrations is still done manually and tediously as it was some three decades ago. Where is the application of modern pattern recognition, so well developed for space applications, but absent in microscopic assays? There have been a few heroic attempts to solve the problem, but it appears that there is no truly integrated biological-engineering heavy attack on the problem. Since the greatest interest is in stable chromosome translocation scoring, the least plentiful aberration, it is all the more imperative to push this technology as far forward as is possible. In this case, sensitivity and accuracy is a function of the sheer number of cells that are scanned, and is limited to the visual fatigue index of the scoring technician. This is another serious challenge for us to address.

The **micronuclei test** is very useful, accurate and simple. It has a major drawback in that it does not distinguish between dividing and non-dividing cells; thus as an unstable aberration assay it is limited to very short times after exposure and is not very helpful in long term retrospective dosimetry.

A newer cytogenetic tool to identify chromosomal translocations has now been developed. The use of **fluorescent in-situ hybridization** now appears as one of the more powerful techniques for identifying stable damage to chromosomes from radiation. The technique is complex and at present limited to a few laboratories. It is costly and it remains to be seen if some automated modification will be available which can increase its applicability and sensitivity to detect small radiation doses. It has sufficient promise though, that I would encourage a quantum increase of multi-disciplinary expertise focussed on this assay, especially in determining its ultimate sensitivity. This is my third challenge.

For several years we have been presented with an array of somatic gene mutation techniques. One that has received much attention recently is the **glycophorin-A assay**, using a fluorescent labeled monoclonal antibody to the protein

and measuring the loss of a red cell allele in irradiated people. It is purported to be a very stable indicator, with lifelong persistence. I am not enough of a biochemist to go into great detail, but I am concerned that the technique has severe limitations as a radiation biodosimeter. It does not appear very reproducible in the same individual, even though millions of red cells are measured in the assay. There seems to be very great variability among individuals exposed to the same dose, and there has not yet been a careful calibration using persons with known exposures. It is simple, relatively cheap and potentially useful if it can be calibrated and validated.

As with the micronuclei test mentioned above, the **HPRT - mutation assay,** (hypoxanthine phosphoribosyl transferase mutations in T-lymphocytes), is limited to very short times after exposure; it's "signal" faded rapidly. Another assay, the **HLA-A,** (human leukocyte antigen A assay), is tedious, insensitive and not promising at this time. The **beta globulin test** likewise offers some promise, but it is too early to state if this fluorescent antibody indicator of a single base change in hemoglobin can provide a practical tool. There may be other useful techniques in various stages of development.

Individual Biological Dosimetry: an Integral Part of Radiation Dose Reconstruction

What is advocated here is the full utilization of these biomarkers as integral elements in future radiation dose reconstructions. What I have seen thus far is the occasional use of one or more assays, more as an afterthought than an integral part of the protocol. The reason for this which is usually put forth, is that the techniques are too expensive and technology intensive and with insufficient radiation reliability or sensitivity. However the limitations of environmental radiation dose reconstruction also entail significant uncertainties about the variables used in modeling, no matter how sophisticated the tool.

While we are improving the validation and calibration of these models, why not also put in the effort to render the biological tools more sensitive and reproducible? Incorporating statistically appropriate individual measurements in the reconstruction matrix can add a powerful dimension to our efforts at dose reconstruction. I envision a time in the near future when there may be a convergence of these two approaches with a resultant synergy of action that produces dose reconstructions with minimal uncertainty and maximal credibility. I can see where the strengths of each technology can assist in reducing the weakness of the other, to help us answer important questions about prior radiation exposures. Major resources are expended on epidemiology studies, with great emphasis on confounding factors, on subtracting natural disease rates from the subject population, etc. However all this emphasis on reducing uncertainties on the "ordinate" of the relationship, must be accompanied by an equal dedication to minimizing uncertainties about dose. Population estimates based solely on broad environmental models, while sensitive, can only be verified by use of individual measurements. Despite 30 years of work, individual doses in survivors of the

atomic bombing of Japan, still pose large levels of uncertainty. New biological dosimetry is helping to reduce the uncertainty; and so it should in the other populations at risk from prior exposure. That is my final challenge to you.

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The Measuring and Modelling of Strontium-90: an Integrated Retrospective Dosimetry Issue

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Introduction

Dose reconstruction for internal exposure is generally structured on a paradigm of release-transport-intake-uptake-dose. In some cases when it is necessary to reconstruct individual dose for the long time after intake and historical information on the releases is limited, bioassay measurements can be used for retrospective dosimetry of long-lived radionuclides with long biological residence times. Internal dose is related to the time integral of the body burden, therefore, individual measurements of body burden, metabolic model and some general suggestions on intake pattern are three necessary parts for dose reconstruction process in such cases.

^{90}Sr is long-lived bone-seeking radionuclide with a long biological residence time in the body. The world list of data on strontium in man was restricted to a few experimental findings with a single intake, information on global fallout and the measurements of dial painters. A large multitude of measurements of ^{90}Sr in human body for the residents of the Techa river (Southern Urals, Russia) contaminated by fission products in 1949-1956 has been published in open literature only recently (Kozheurov, 1994; Degteva et al., 1994). The necessity of dose reconstruction from long-lived radionuclides for the population living on the territories contaminated as a result of Chernobyl Accident calls the utilisation of the Urals experience to optimise the efforts. An analysis of a unique and abundant Urals data set on strontium in humans presents also a great interest for general purposes of retrospective dosimetry.

Population exposure in the Urals occurred as a result of failures in the technological processes at the Mayak plutonium facility in the 1950's. Construction of the Mayak facility began in 1945 and was completed in 1948. Initially this complex consisted of three main parts: Reactor plant, radiochemical facility, and waste-management facilities. The major sources of radioactive contamination were: (1) the discharges of 2.7×10^6 curies of liquid wastes into the Techa River (1949–1956); (2) an explosion in the radioactive waste storage facility in 1957 (the so-called Kyshtym Accident) that formed the East Urals Radioactive Trace (EURT) due to dispersion of 2 ×

10^6 curies in the atmosphere; (3) the resuspension of 600 curies with dry silt from the shores of Karachay Lake during a heavy thunderstorm (1967); and (4) routine gaseous aerosol releases within the first decades of the facility's operation (Degteva et al., 1996). The significant portion of activity for the Techa River and EURT consists of long-lived radionuclides, mainly ^{90}Sr. These releases resulted in the long-lived contamination of surrounding territories. The activity released from Karachay Lake also consisted of ^{90}Sr and ^{137}Cs, and it superimposed on already existed EURT.

There are two cohorts of accidentally exposed population in the Urals: the residents of the Techa riverside communities and the residents of the area covered by EURT. The object of primary interest is Techa River population. The Techa River cohort is important because some of whose members have received relatively high doses and a significant increased risk of leukemia with increasing dose was observed (Kossenko and Degteva, 1994). The residents of the villages along the Techa were exposed to both external irradiation (from contaminated river water, sediments, flood plane soils) and internal irradiation due to ingestion of radionuclides (mainly ^{90}Sr) with drinking water and diet.

Whole-body counting system for measuring ^{90}Sr via bremsstrahlung

Since 1974 the large scale investigations of the ^{90}Sr content in the skeleton of Techa riverside residents with the whole body counting system SICH-9.1 were started. A detailed description of SICH-9.1, of the measurement procedure and results were given elswere (Kozheurov, 1994). The shielding for the whole-body counter was made of cast iron rings with a wall thickness of 200 mm. The inner surface has linings of lead (5mm), cadmium (1.5 mm), and copper (1 mm). Four Phoswich detectors are used for measuring. The geometry of measurement is as follows: the detectors are fixed in the central vertical plane with a 30° inclination towards the vertical. The distance between the axes of each pair of detectors is 35 cm at the base. On the bed frame, fabric is stretched which caves in under the weight of the body in such a way that the medium plane of the body is at a distance of 25 cm from the detectors. During the measurement the proband's bed is moved through the detector array. The scanning length is 2 m with stops at the ends lasting 10% each of the whole measurement time. The motion is controlled by impulses from the analyzer real-timer. The duration of the measurements is 20 min. The scanning device used in SICH-9.1 is discussed in (Zaitsev and Kozheurov, 1978), and the calibration technique of

the whole-body counter is described in (Kozheurov et al., 1978). Minimum detectable activity (MDA) for SICH-9.1 is equivalent to 1.85 kBq of ^{90}Sr.

For the calibration of WBC two surrogate human structures were made by two different laboratories. Both phantoms were made of natural human skeletons, paraffin imitations of soft tissues and dry paper imitations of lung. Different methods of introducing ^{90}Sr into the bones of the phantoms were used. In one of the phantoms the isotope was introduced by being dripped into uniformly distributed holes drilled the bones. The bones of the other phantoms were impregnated with ^{90}Sr solution in a vacuum chamber. Each laboratory performed independent experiments aimed at determining bremsstrahlung yields relations and the influence of human soft tissues and the phantom paraffin on the absorption of the bremsstrahlung. Independent activity measurements were carried out on each phantom. After scanning the phantoms and making the appropriate corrections the difference in calibration coefficients was determined to be 6%. This value represents the estimate of the systematic error in ^{90}Sr counting by means of the spectrometer. Water-filled phantoms made of plastic tanks laid out in such a way as to imitate a human body were used for calibration of ^{137}Cs and ^{40}K. The length of the phantom could be changed by removing one or two tanks. Such calibration was done in 1974 and did not confirmed during twenty years period of WBC operating. It is highly desirable to ensure that the old measurement system is once again calibrated in depth on the basis of measurements of a special antropomorphic phantom.

Strontium Registry

The measurements of ^{90}Sr in humans for the residents of the Techa river were started in 1951. The first findings were ^{90}Sr concentrations in different bones received from autopsies (Table 1). Since 1960 the surface beta-activity on the teeth has been measured; this was made possible with the detectors suitable to perform such measurements in the mouth of a person (Kozheurov, 1994). From 1962 through 1979 daily urinary excretion was estimated by radiochemical methods. And since 1974 the Techa river residents have also been examined for their body burdens with the whole-body counter (Kozheurov, 1994). In parallel with whole-body counting the measurements of beta-count of forehead bone are carried out. About 30,000 measurements have been performed on more than 14,000 persons from the contaminated. Both men and women of all ages have been measured. This has given a unique experience of the arrangement of individual monitoring of ^{90}Sr human body contamination. A list of available data is given in Table 1.

Table 1. Techa River Strontium Registry.

Measured values	Number of measurements	Number of subjects measured	Period of observation
Sr-90 whole-body content	28,500	14,200	since 1974
Tooth beta-count	30,200	15,300	since 1960
Forehead beta-count	23,100	12,300	since 1976
Sr-90 urinary excretion	2,910	1,564	1962-1979
Sr-90 concentration in bone	350	160	1951-1973

The results of the multitude of measurements showed a clear dependence of the ^{90}Sr body burden on the year of birth, and formed the basis for the elaboration of age-dependent model for strontium retention in human skeleton.

Metabolic Modelling of Strontium

Our age-dependent model of strontium retention in human bone is descridbed in detail elsewere (Degteva and Kozheurov, 1994). This model represents mathematical function similar to ICRP-20 retention function (in a simplified form), in which parameters are dependent on age at intake. The main task for the model development was to estimate the parameters and to validate the model under different conditions of strontium intake. The approach to the validation of metabolic models established in ICRP Publication 20 and accepted now, was the following: to collect all available sets of data on the metabolism of a particular radionuclide in humans which are suggested to reflect Reference Man (ICRP-23) under various schedules of intake; to fit model calculations for available schedules of intake based on the metabolic parameters of Reference Man to the sets of data. Thus, checking of model fitness and parameter verification are carried out.

As a starting point for parameter fit the general view of some age-dependent functions was identified by combining relatively extensive information concerning the development of the human body with the findings of experiments on different species of animals. Then the available sets of human data were determined for the more precise estimates using empirical fitting technique. Besides the Techa river data, the information on strontium in human bone from global fallout (Leggett et al., 1982; Loutit, 1967; Papworth and Vennart, 1984; Warren, 1972) and experimental findings with single injection (Bishop et al., 1960; Kereiakes et al., 1968; Kohn et al., 1962; Likhtarev et al., 1975; Woodard and Dwyer, 1972) were involved in parameter estimation and the validation of our model. Such approach has allowed to derive the model which takes into account changes in metabolic parameters throughout the entire life beginning from bone formation and growth in childhood and including loss of skeleton calcium in old age. The model gives reliable description of ^{90}Sr retention from one month to thirty years from the moment of intake.

Another possibility of the model verification was given by a set of data on direct assessments of individual strontium excretion rates dated 25-35 years after the basic intake for persons aged 30-70. Excretion rate was estimated by two different methods: using radiochemical analysis of bioassay samples and on the basis of repeated whole-body counter measurements (Table 2). The latter approach made it possible to receive the most detailed information concerning age- and sex-dependencies of the late phase of excretion, in comparison with earlier studies (Muller, 1970). The comparison of model prediction with experimental results has shown that our model overestimates excretion rate for the ages 30-45 years and does not take into account distinct sex differences in excretion rates for the elderly persons. This fact give us reasons to continue our studies and to improve our new model for strontium metabolism. Nevertheless, the model presented served to bring together into a relatively simple framework the majority of experimental data on strontium in humans and can be useful as workable age-dependent model in radiation protection.

Table 2. The comparison of strontium excretion rate estimates in late phase of retention for humans.

Study description	**Muller, 1970**	**Present paper**	**Kozheurov, 1994**
Sample information - sex - age, yr - ^{90}Sr body burdens, kBq	9 luminizers 1 M + 8 F 22 - 41 1.85 - 70.3	40 residents of the Techa riverside 21 M + 19 F 37 - 50 7.4 - 145.0	717 residents of the Techa riverside 367 M + 350 F 30 - 76 3.0 - 164.5
Methods of measurements	Radiochemical analysis of bioassay	Radiochemical analysis of bioassay in parallel with whole-body counting	Repeated WBC measurements made within 5-10-year intervals
Time since exposure, yr	2.2 - 8.2	29 - 30	25 - 35
Statistical evaluation techniques	Model extrapolation using additional data on stable Sr	Direct assessment	Direct assessment
Resultant excretion rate estimates	$(7.7\pm1.0)\ 10^{-5}$ per day 0.028 ± 0.004 per year	$(8.3\pm3.6)\ 10^{-5}$ per day 0.030 ± 0.013 per year	Distinct age- and sex-dependence. Mean value for ages 30-40 years: 0.025 ± 0.016 per year

Conclusions

The results of a long-term investigation of the Techa river population have given a unique experience in the arrangement of individual monitoring of ^{90}Sr human body contamination for the purposes of retrospective dosimetry. Techa River Strontium Registry data base can serve for the development of age-dependent metabolic models for long-lived bone-seeking radionuclides. The data of concern are of major interest to the reconstruction of dose to the red bone marrow and bone surfaces of the members of the Techa river cohort which could give a possibility of direct radiation risk assessments for long-term chronic exposure conditions.

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THE ABSORBED DOSE IN TOOTH ENAMEL BY ^{90}SR BODY BURDEN

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SUMMARY

The residents of the villages along the Techa river, Southern Urals, Russia, were exposed mainly external by γ-rays from ^{137}Cs and internal by β-rays from ^{90}Sr and its daughter ^{90}Y, as a result of the heavy release of radioactive waste from the plutonium production facility Mayak into the Techa in 1950-51. The residents living at the lower part of the river were exposed only little externally but mainly internally due to their ^{90}Sr body burden. Teeth from the latter group of residents which were extracted in 1994-95 in the course of dental care were collected and used for retrospective dosimetry more than 40 years after the main ^{90}Sr ingestion.

The absorbed dose in the enamel of the teeth was measured by EPR dosimetry. In addition the ^{90}Sr content in the roots of the teeth was measured by low-level beta counting. Both measurements at the teeth were done in 1995. The whole-body burden of the tooth donors was measured via bremsstrahlung in 1992-1994. The absorbed dose in the enamel of molars was found to correlate with the retained ^{90}Sr whole-body burden (Fig. 1). The equivalent dose in the enamel of the molars per ^{90}Sr body burden retained 42 years after the ingestion was determined to $3*10^{-5}$ Sv/Bq. The accumulated background dose in the enamel of the in average 60 years old residents was found to be about 100 mSv. This results in an average annual dose rate for the background radiation of 1.7 mSv/a. The results suggest that the absorbed dose measured in the enamel of molars can provide information about the ^{90}Sr burden as a mean of the whole skeleton. In contrast to the encouraging results obtained from the dose measurements in the enamel of molars, no correlation with the ^{90}Sr body burden was found for the dose measured in the enamel of incisors. Similar discrepancies with dose measurement in the enamel of incisors were found earlier by other authors also in the case of predominantly external radiation. At present the reasons for these difficulties are not known but might be due to stronger sensitivity to environmental influences and/or problems in preparing the very thin layers of the enamel of incisors for the EPR measurement. Until these problems are resolved we recommend to exclude measurements of the enamel of incisors for retrospective EPR dosimetry.

No correlation could be found for the absorbed dose in the enamel of neither molars nor incisors with the ^{90}Sr content in the dentine of the roots of the teeth (Fig. 2). Due to the short range of the β-rays from ^{90}Sr and its daughter ^{90}Y, the dose in the enamel is expected to result predominantly from the ^{90}Sr content in the adjacent dentine within the tooth crown and less from the content in the root dentine. The results of the present investigation let suppose that the strontium metabolism in the dentine of the crown is comparable with the that of the mean of all parts of the skeleton. In contrast to the dentine of the root which is under extreme mechanical strain and hence remodels strongly. Therefore, measurements at the root dentine are not expected to provide conclusive information about the ^{90}Sr burden of the overall skeleton.

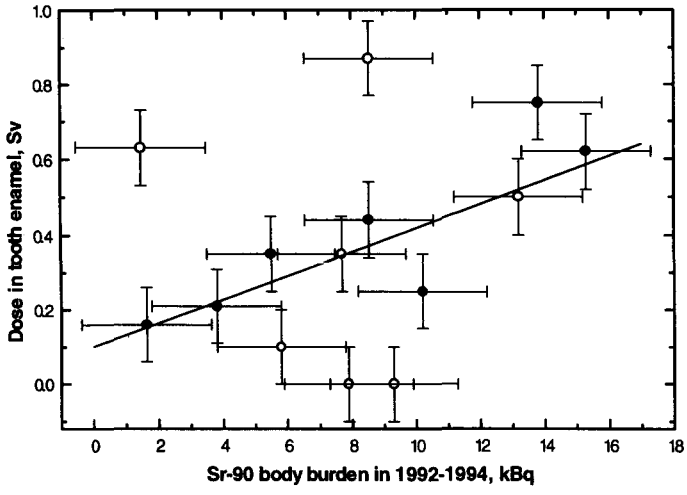

Figure 1. The absorbed dose in tooth enamel as measured by EPR dosimetry versus the ^{90}Sr whole body burden. The dose in the enamel from molars (●) were found to correlate with whole body burden. No correlation is obvious for the dose in the enamel of incisors (○) with the whole body burden.

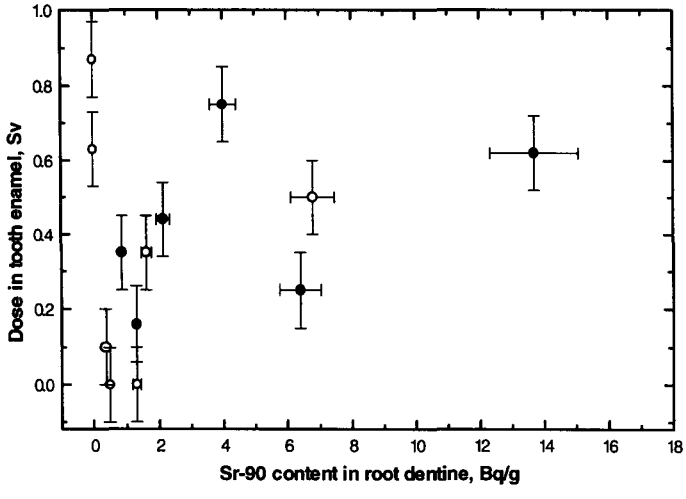

Figure 2. The absorbed dose in tooth enamel as measured by EPR dosimetry versus the ^{90}Sr content in the root of the tooth. The dose in the enamel from molars (●) and incisors (○) were found to correlate not strongly with the ^{90}Sr content in the root.

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Retrospective Dosimetry using EPR and TL Techniques: A Status Report

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Methods of retrospective dosimetry, including luminescence and electron paramagnetic resonance spectroscopy (EPR), rely on measurement of accident dose absorbed by naturally occurring materials - ceramics in the case of both thermoluminescence (TL) and optically stimulated luminescence (OSL) and organic materials and bio-minerals in the case of EPR. Each of these methods relies on measurement of radiation defects resulting from accidental exposure. Since defects also result from natural sources of radiation over the lifetime of a sample, analysis is usually restricted to materials for which the natural dose may be determined and subtracted from the measured cumulative dose.

The transient, or accident radiation dose 'D_X' is determined as follows

$$D_X = D_{TL} - (R_\alpha + R_\beta + R_\gamma + R_\chi)A$$

Where

D_{TL}	=	TL measurement of total accrued dose
A	=	Sample Age
R_α	=	alpha dose rate
R_β	=	beta dose rate
R_γ	=	gamma ray dose rate
R_χ	=	cosmic ray dose rate

Luminescence dating techniques rely heavily on an accurate assessment of cumulative dose from natural radiation sources, and dating research has provided us with the bulk of our knowledge in this area [1]. Virtually all of the work on natural dose determination can be directly applied to retrospective techniques. With EPR techniques the cumulative dose from diagnostic x-rays is also of importance.

Electron Paramagnetic Resonance Techniques

EPR analysis of tooth enamel can provide measurements of cumulative dose to teeth. The attraction of enamel as an individual dosimeter lies in the ubiquity of the material and the fact that

wisdom teeth are routinely removed and others are extracted as a result of age related periodontal disease.

Interlaboratory Comparisons of EPR Techniques

EPR dosimetry of tooth enamel is in a rapid stage of development. Two interlaboratory comparisons have produced recent validation of different aspects of the technique. The first, performed as part of ECP-10, was a blind study [2] involving 9 laboratories from the Former Soviet Union (FSU), Europe and an invited laboratory from the U.S. The results of this intercomparison showed marked variation in accuracy produced by individual laboratories, however one laboratory produced results which exceeded expectations (Fig. 1). The study was designed to assess the reliability of different methodologies under ideal conditions. The intercomparison involved the preparation of a homogeneous mixture of grains from unirradiated teeth which were irradiated after crushing. The project did not examine effects which may be associated with the preparation procedure or effects associated with irradiation of whole teeth versus crushed teeth. The results are currently being assessed by the individual members of the project, and the analytical procedures used by the most successful laboratory are being adopted in whole or in part.

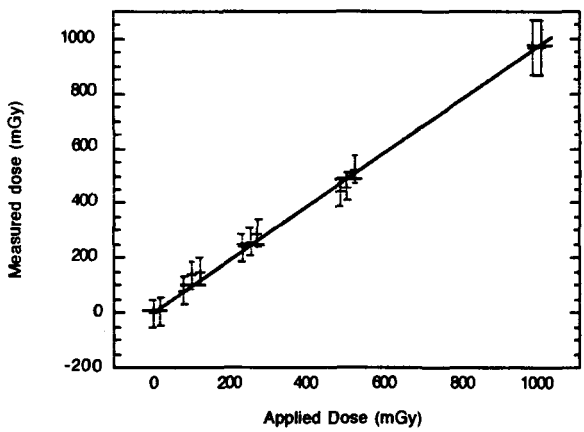

Fig. 1. Most accurate results of the ECP-10 intercomparison

A second, bilateral intercomparison [3], performed by the University of Utah and the Ukrainian Scientific Center for Radiation Medicine (USCRM), was designed as a pilot study for future multilaboratory comparisons. This study involved irradiation of teeth prior to crushing. The design of the blind study allowed assessment of applied dose as well as dose from prior x-rays. Results from both laboratories showed mean deviations of measured versus cumulative dose (including x-rays) of approximately 20% for doses on the order of 0.2 Gy.

Problems associated with EPR analysis of enamel

EPR of enamel is not without its problems, and additional research is needed to clarify and minimize uncertainties. The presence of organic and mechanically induced signals [4,5,6] superimposed on the radiation sensitive signal has been recognized as a potential limiting factor in the ultimate accuracy of the technique. The need for a zero dose background spectrum which is subtracted from the spectrum of an irradiated tooth means that variations in the large organic signal will be translated into significant errors in the overall dose estimate. The approaches to this problem have been threefold. The first involves examination of each spectrum generated from an irradiated tooth and comparison with a well prepared zero dose background tooth [7]. Deviations in the organic signal are, with experience, recognized and eliminated by treatment in NaOH. Once organic signals match, the spectra may then be subtracted. The validity of this approach is evidenced by results (Fig. 1) from the ECP-10 intercomparison. A second approach involves modeling of the EPR signal and computer isolation of the organic and radiation sensitive components. Progress in identifying and isolating the organic component from the mechanical portion has recently been made with the use of organic and synthetically grown apatites [8]. The third approach involves variations on the measurement procedure itself and utilizes differences in signal saturation as a function of microwave power. The differential power method [9,10] allows determination of cumulative dose without the need for a zero dose background signal. Additional work is needed with methods two and three above, however the implementation of the first method offers an immediate reduction in uncertainties due to the background organic signal.

Another recently recognized problem associated with EPR dosimetry of teeth involves the effect of light exposure on the EPR signal of teeth [11]. This problem was identified by observing large variations in dose estimates obtained by analysis of incisors and canine teeth which may be exposed to sunlight. This effect poses a serious limitation in the ability to perform EPR dosimetry on canines and incisors, however it appears to have no effect on analysis of molars or wisdom teeth. It does reduce the total number of collected samples which can be analyzed with confidence, and in some cases it may mean a retrospective deletion of previously measured samples..

Yet another potential source of error of EPR dosimetry of enamel concerns the effect of pre and post crushing sensitivity of the radiation induced signal. Recent research [12,13] indicates that an overestimate of dose may be obtained if grain sizes smaller than 100 μm are used to measure dose delivered prior to crushing. This study, while not yet confirmed by other laboratories also

indicates that use of grain sizes in the range of 250 to 600 μm does not produce this overestimate. This effect also requires further examination and explanation.

Dental x-rays can represent a significant source of uncertainty in dose estimation due to the difference in energy dependence of hydroxyapatite versus that of soft tissue. A seven to one difference in deposited dose in enamel versus that in soft tissue at typical x-ray energies of 60 KeV represents a threat to the utility of EPR as a biological dosimeter. Fortunately, two factors reduce this uncertainty. The first is the purported rarity of dental x-rays in the FSU, an assertion which needs additional documentation, and the second is the differential dose deposition in the outer and inner portions of teeth due to attenuation of the x-ray beam. This effect has been used previously to distinguish dose due to x-irradiation [14,15,16] and was used by USCRM to determine x-ray dose in the teeth used in the bilateral intercomparison mentioned above [28]. Because of the potential overestimation of dose due to dental x-rays, it may be necessary to routinely measure interior and exterior portions of teeth. Molars or wisdom teeth, because of their large size, may again be required for this multiple analysis.

Riding on each of these uncertainties is another potential limitation to the accuracy and ultimate lower limit of detection of EPR techniques. This source of uncertainty is the non-isotropic response associated with measurement of crystalline grains. Theoretically grains sizes sufficiently small will result in an isotropic EPR response which is independent of sample orientation. In practice, use of such small grain sizes may present a problem of sensitivity difference mentioned above. Additionally, the effect of anisotropy is enhanced with increasing microwave power. This places limits on the effectiveness of the two power method discussed previously. A recent addition to the EPR dosimetry process, a constant rotating goniometer [17], reduces the effect of anisotropy making measurements at all microwave powers, and particularly the higher powers, significantly more accurate. Since the organic component of the EPR signal seems to be most effected by anisotropy, this procedure may well result in reduced limits of detection. Fig. 2 shows the result of three spectra obtained following separate shakings with no rotation and three spectra obtained following separate shakings with rotation during spectral measurement. The spectra have been passed through a low bandpass filter to remove high frequency noise and more clearly show the improved reproducibility.

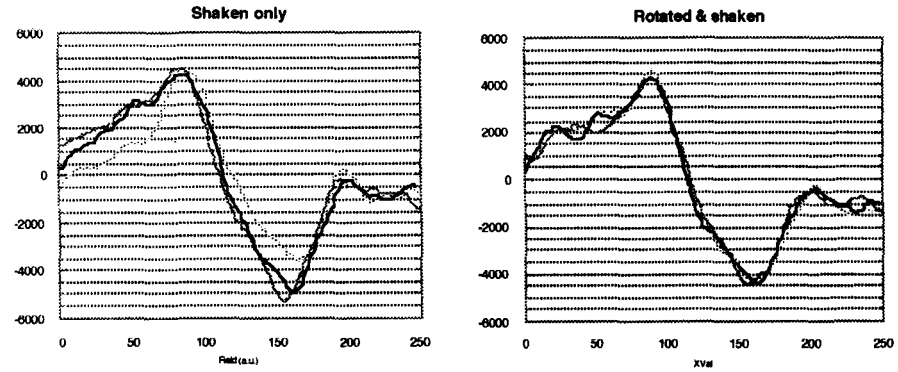

Figure 2. Effect of constant rotation goniometer on EPR reproducibility.

Many of the uncertainties addressed above are serious and can result in significant inaccuracies if not properly recognized and addressed. However, none of them seems insurmountable nor represents a fundamental limit to the utility or accuracy of the technique. Assessment of errors has received recent attention [18] and methods for time optimization of analysis are also being developed [19]. In fact, it seems highly likely that continued research will result in steady improvements in the speed, reliability, and lower limits of detection of teeth as an EPR dosimeter.

A related EPR technique involves measurement of deorganified dentine and comparison with measurements from enamel [20,21]. This technique has the potential of allowing differential measurements of dose due to external gamma rays as well as internal emitters such as Sr-90. An advantage to the use of dentine as a dosimeter is the fact that the organic signal present in enamel is eliminated in the deorganified dentine. Use of dentine also eliminates any possible effects of UV irradiation. At present the lower limit of detection of dentine dose does not reach that of enamel. However, this area of research is still in its infancy.

Luminescence Techniques

Luminescence dosimetry of environmental materials was first associated with accident dose reconstruction during the early 1960's. Thermoluminescence techniques were applied to the reconstruction of doses at Hiroshima and Nagasaki [22,23,24,25,26], the Nevada test site [27,28], and most recently, the regions downwind of the Chernobyl Nuclear power plant [29,30,31]. Several reviews of retrospective techniques used in these studies detail procedures, associated problems and experimental validations which have been conducted [32,33,34]. Perhaps the most extensive effort at technique validation prior to current efforts associated with the Chernobyl

accident was the TL study of Hiroshima and Nagasaki conducted as a part of DS-86 [24]. Validations included tests of dose rate effects, neutron sensitivities, preparation effects and numerous intercomparisons and intercalibrations. A partially blind, multilaboratory study involving NBS-irradiated annealed quartz removed from a Nagasaki brick sample produced measurements of dose which agreed with applied values to within better than ±10% (1s) at each of 3 dose levels ranging form 82 mGy to 417 mGy. Comparison of identical tile samples, when measured by both the high temperature and pre-dose TL techniques, produced agreement of ± 10% [35]. In an intercomparison involving different tiles from the same sampling sites, the laboratories also produced dose estimates agreeing to within ± 10%. The TL results showed close agreement with theoretical calculations in the city of Nagasaki. Interlaboratory agreement was again very close at a site in Hiroshima examined by the laboratories, however the overall measurements in Hiroshima at a distance from the hypocenter of approximately 1.4 km was approximately 20% higher than theoretical calculations. Further measurements using both the pre-dose and high temperature techniques at even greater distances has tended to verify these results [36,37,38]. The reason for the discrepancy between the measured and calculated gamma ray dose values in Hiroshima remains unclear, as does an even greater discrepancy between theoretical and measured neutron fluences in that city[39,40,41].

Concurrent with the Hiroshima/Nagasaki effort was an evaluation of doses delivered to regions downwind of the Nevada Test Site [27,28]. This effort evaluated transient doses of less than 100 mGy and relied exclusively on the use of the predose TL technique. Because the doses of interest were low compared to natural dose accumulation, rigorous diagnostic tests [27,28,42,43] were developed and applied. The study allowed comparison of measured doses in building bricks versus evaluated exposure estimates based on soil sampling measurements, reevaluation of original monitors reports and a comprehensive review project.

The Chernobyl nuclear accident has lead to several recent applications [29,30,31,34] of TL techniques. Again collaborative efforts were employed allowing intercomparisons of separate techniques as well as similar techniques applied by different laboratories.

The multi-national effort sponsored by the European Community (ECP-10) in collaboration with the Former Soviet Union, continues to examine the feasibility of retrospective techniques applied to the Chernobyl accident. The program has contributed to the validation of more recently developed techniques, has incorporated measurement into Monte Carlo models at actual exposed locations has investigated correlation of measured dose depth profiles in ceramics with theoretical calculations and has mounted a series of studies designed to insure interlaboratoy reliability [44].

Perhaps the biggest question currently facing luminescence dosimetry efforts involves optimization of methods for integrating results into modeling efforts. This problem is being addressed by the ECP-10 effort and requires close cooperation of modelers and measurers at the earliest stages of reconstruction.

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RISK EVALUATION AND DECISION MAKING

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THE THRESHOLD ERA

In the early days of radiation protection, the harm to be avoided (skin damage and deterioration of bloodforming organs) was of *deterministic* nature, with a dose threshold that had to be exceeded for harm to occur. "Tolerance doses" and dose limits were chosen with a margin of safety to make sure that deterministic harm could not arise.

In this case, the doses to control would be the sum of all dose contributions, irrespective of source, since it would be the total dose that would cause the harm. Conceptually, this would have called for inclusion of medical exposures and doses from natural sources in the dose to be controlled, but in practice the latter contribution was believed to be insignificant and not controllable but fairly constant, while the medical exposures fell under the responsibility of the medical doctor. However, in principle, the assessment and control was *individual-related.*

THE STOCHASTIC CASE

On the assumption of dose thresholds for harmful effects, unconditional safety could be guaranteed if the sum of all dose contributions did not approach the threshold. When it was realized that some harm (cancer and hereditary harm) was of *stochastic* nauture and showed dose-response relations in which no thresholds could be assumed, the situation changed strikingly. No dose could be said to be absolutely safe, and the choice of a dose limit implied a decision on acceptable risk rather than the identification of safe doses.

The assumption of a linear, non-threshold relation between radiation doses and the probabilities of the various types of stochastic effect over the dose region of interest in normal operations made it justified to consider the dose contributions from each source or practice without regard to doses from other sources, as long as the total dose was not too high. Each dose increment is expected to cause one and the same detriment increment irrespective of other exposures. This opened the door to *source-related* assessments and decisions.

OPTIMIZATION OF PROTECTION

The awareness of stochastic harm made the situation more complex in some respects but more simple in other ones. To continue with a control system that depended on a fixed dose limit no

longer seemed appropriate; it would have been unnecessarily stringent in some cases and too relaxed in others. The position of the dose limit as the primary means of control was not retained; instead, the main protection principle became the attempt to keep all doses "As Low As Reasonably Achievable, economic and social factors being taken into account" (optimization of protection, sometimes referred to as ALARA). It was obvious that, at some level of protection, the cost and efforts of further dose reduction would not be justified.

Various methods may be applied to reach ALARA. In the first ICRP interpretation [1], the principle was illustrated by means of differential cost-benefit analysis, but (as the Commission says in its 1990 recommendations [2]) the methods range from simple common sense to complex techniques of cost benefit analysis or multiattribute analysis. In all cases the objective is to minimize the sum of the costs of protection and the radiation detriment. Protection costs may include more than money, and the radiation detriment may include more than detriment to health. However, to make comparisons possible, all these components must be expressed in monetary terms, implicitly as in applying common sense or explicitly as in cost-benefit analysis.

THE VALUE OF α

If the collective dose is taken to indicate the radiation detriment, cost-benefit analysis must be based on some *a priori* decision on the amount of money that it would be appropriate to pay to eliminate a unit collective dose. This amount (expressed in, *e.g.* US$ per man·Sv) is often referred to as "α". If all doses are in the region of stochastic harm, the choice of α implies a choice of a corresponding amount of money (we may denote it "A") appropriate to pay for saving, in a statistical sense, a not identified human life. The relationship between A and α is $\alpha = r \cdot A$, if r is the probability coefficient for radiation-induced death or death equivalent harm (in ICRP Publication 60 [2] r is given as 7.3% per Sv for an average member of the population at large and 5.6% per Sv for the average worker).

The appropriate value of A can easily be shown to fall between two extremes. The lowest value may be derived from the annual gross national product *per caput* of the productive population. It would be short-sighted to pay less per man·year saved, and for equity reasons at least the same amount should also be paid in saving non-productive man·years. The highest value may be derived from an estimate of the potential saveable man·years per year in the population and the highest fraction of the annual gross national product that could be used for statistical lifesaving without jeopardizing society's capability to sustain the same capacity for lifesaving over future years. The range for the choice of A in developed countries is then found to be roughly 0.5 - 5 M$ and the available range for the choice of α would be 0.035 - 0.35 M$. The geometric mean of the extremes is about 100.000 $ per manSv.

It is sometimes said that cost-benefit methods would be unethical because we should not "set a price" on a human life. This claim was considered by an expert group appointed by the Pontificial Academy of Sciences in 1983, but it was concluded that it cannot be unethical to use a method designed to save the maximum possible number of lives for the money available for lifesaving.[3] The ethical problem arises with the decision to set aside a certain amount of money for lifesaving rather than for some other purposes, not with the optimized use of that money.

There is, however, another method to derive the value of α, namely what is called "willingness-to-pay". In this case, a review is made of the amount of money that has either actually been spent for protection in various cases or has been said to be appropriate to spend. This sum has then been divided by the number of lives assumed to be saved by the protective action. The disadvantage with this method is that the resulting quotient is not necessarily an indication of the appropriate value of A.

If the total amount of money spent on protection in a particular situation is denoted T and we assume that this is exclusively intended to save N lives, then we have the relation $T = A \cdot N$ and we may conclude that $A = T/N$ is a measure of the willingness-to-pay to save a human life. However, in most cases there is an additional sum of money (W) which is paid to achieve other objectives than lifesaving. In some cases money must be paid for measures taken to prevent prescribed limits from beeing exceeded. In other cases, there may be a willingness to pay for protective actions taken to demonstrate good will without particular attention to lifesaving. We may therefore write

$$T = A \cdot N + W$$

in which case

$$T/N = A + W/N$$

This means that the quotient T/M is often not an implicit measure of A. A striking example is an often quoted value of 72.000.000.000 $ per life in protection against formaldehyde.[4] In that particular case, the protective action must have been needed to protect against non-lethal effects, and because N was very small, we may assume that $T/M \approx W/N$, in which case there is no basis for any estimate of A.

If $W = 0$, we might think that the implied value of A is T/N, but this is only true if T (the amount of money actually paid) was less than or equal to the total amount available. If this amount is small, as is often the case in medical practices, T may not have been derived as $A \cdot N$ (*i.e.* after an *a priori* assessment of a potential N) but paid because it happened to be available, and N is then merely the result. If more money than T was available (as may be the case in the nuclear industry), it is more likely that T was actually derived from knowledge of the potential value of N. If T was exclusively intended for lifesaving and $W = 0$, then, but only then, we might take T/N as an indication of A based on willingness to pay. The range of results reviewed in a recent Swedish study [5] was about 0.1 - 7 M$, amazingly close to the available values mentioned above.

DOSE LIMITS AND SOURCE-RELATED DOSE CONSTRAINTS

For public exposures, the primary dose limits have lost their importance, except as benchmarks for radiation protection authorities in their overview of all sources and practices. Instead, ICRP now recommends source-related dose constraints to be imposed in order to prevent the ALARA result from being achieved at the cost of an unacceptably high risk to some few individuals.[2] This extra safety requirement is needed because harm and benefit are usually not equally distributed and it would not seem reasonable to expect people to accept unusual risks where they see little benefit.

The limits actually enforced by competent authorities would be derived limits of measurable quantities (releases, activity concentrations, etc) as deemed appropriate from ALARA

considerations under the additional requirement that the source-related dose constraint is not exceeded for critical groups.

ETHICAL PROBLEMS

There are a number of ethical problems inherent in risk evaluation for purposes of decision making. Already radiation risk assessment is based on a number of assumptions which imply ethical positions. The existence and magnitude of radiation risks at low doses can never be proven and is only inferred from credible assumptions. It is in line with the precautionary principle to postulate, as ICRP does, that the risk does exist.

Once this is postulated, the quantity "risk" has to be defined. There are two types of risk to consider: (1) the risk to the most exposed individuals, and (2) the overall risk due to a practice or operation. In general, "risk" may be seen as a combination of event probabilites and consequences. In the individual case, the worst consequence is death and "risk" may then mainly be assessed as the probability of radiation-induced death (although it matters how and when death is expected to occur). In the source-related overall risk, there are the two cases of either actual exposures or potential exposures. In the case of actual exposures and collective doses of at least tens of man·Sv, the probability of *some* consequence is near one. The "risk" is then described by the expected number (>1) of deaths, assumed to be proportional to the collective dose. The case of potential exposure is more difficult to handle.

The most exposed individuals

The main objective related to the risk to the most exposed individual is limitation of the probability of death to fall below some chosen level. In the case of actual exposures, there is a corresponding dose constraint or prescribed limit related to the particular radiation source or practice. One ethical question is then whether future individuals shall be protected by the same cautious dose constraints as present individuals. This is a pertinent question in the case of radioactive waste disposal.

The overall risk from the practice

The main objective in controlling the overall risk from a practice is to keep the expected consequence as low as reasonably achievable. If the collective dose is at least some ten man·Sv, there is an expectation of at least some death. If protection is optimized by means of cost-benefit analysis, then a unit collective dose may be costed by $\alpha = r \cdot A$, using for A the valuation of lives that is achievable according to the previous discussion. If the collective dose is less than a few man·Sv, the most likely number of deaths is zero. If we do not want to sub-optimize but look for a method which gives a total optimization, we should use the same value for α also in this case.

The ethical question then arises whether we wish to use the same value for α also for collective dose contributions which are distant in space and time. With regard to distance in space, an IAEA expert group has recommended "As a basic principle, policies and criteria for radiation protection of populations outside national borders from releases of radioactive substances should be at least as stringent as those for the population within the country of release." [6]

The policy with regard to distance in time has been much discussed. From economists there are often proposals for discounting of committed future detriment costs. However, from the

ethical point of view, this is not an economical issue. Discounting future detriment costs merely means that future detriment is not added at full value to present detriment. A common practice is therefore to avoid discounting but disregard radiation doses which would occur in a very distant future, usually 10.000 years is mentioned.[7] There are two reasons not to include these very distant doses without any obvious conflict with ethical principles. One is that the dose assessment becomes so uncertain that it does not provide a useful basis for optimization calculations. The other is that dose reduction by isolation of the radioactive material may not be effective for more than 10.000 years, so that the dose contribution from later years will be the same for all options and would therefore not influence the optimization result

ATTENTION TO ATTITUDES

Decision-making is influenced by attitudes to risk. This is true both for those who evaluate the risk, those who take the decisions and those who are expected to accept the decisions. Distinctions are often made between "objective" risk and "subjective" or "perceived" risk and there are numerous discussions about "acceptable" risks.

In the author's opinion, all these discussions are simplified if we relate acceptability not to risk but to the practice or situation that causes the risk. To ask whether a risk is acceptable is similar to asking "Is this stone too heavy?" Such questions cannot be answered out of context.

Whether a practice or a situation is acceptable or not (to somebody) depends on the balance of advantages and disadvantages. Risk is one of the latter, but so is cost, disturbance, discomfort, etc. This is balanced against various types of benefit. Without knowing the benefit as well as disadvantages other than risk, it is not possible to decide on acceptability.

Everyone has an attitude towards proposed practices and our attitude will influence the balance. Some aspects or foundations of our attitudes may be inherent, depending on our social standing and environment, culture, education, religion, etc. In particular situations, however, there are a number of well known factors that form our attitude. Usually they are said to influence our risk perception, but this view may be too narrow since they influence our attitude to more than just the risk. The most important factors are:

I see a benefit	I don't see a benefit
I control the situation	I have no influence
I understand the risk	I don't understand the risk
The situation is familiar	The situation is new
I accept this voluntarily	It is imposed on me
I trust those responsible	I don't trust them

The items in the first column form a positive attitude, those in the second column contribute to a negative attitude, something which one author calls "outrage".[8] If there are several outrage factors, the practice or situation is not accepted.

The decison-maker is influenced by the same factors, but does not always share them with those exposed to the risk. To what extent should then attention be paid to their attitudes? It is known that risk reduction has rather little influence on acceptance, reducing the outrage is more effective. [8] Some of the outrage factors are subjective but some are indisputable, *e.g.* a

new, unfamiliar type of risk, children being exposed, commitments for the future. The existence of such factors should call for particular caution.

UNDERSTANDING

A major problem for the decision-maker is to device a control system which is understandable and therefore more likely to be accepted. In the old days, the situation was clear-cut. There was a limit. Below the limit it was safe. Simple as that. Today we have a limitation system, very properly based on protection optimization, with constraints and prescribed limits which differ from case to case. Even below the limits no absolute safety can be guaranteed. "You have no reason to worry, but..." We want to have the very best protection, but may ask ourselves if there is a proper way of exchanging some degree of protection and equity for a more understandable system. Since the unswer is likely to be no, how could we nevertheless improve understanding?

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SEMINAR ON RISK EVALUATION AND DECISION MAKING

HOW RISK EVALUATION IS BUILT INTO OPTIMIZATION AND DECISION-MAKING: IS IT CLEAR AND ARE THE RESULTS APPROPRIATE?

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INTRODUCTION

The concept of optimization of protection is practical in nature. In all aspects of dealing with protection of man against ionizing radiation, it is necessary to make frequent decisions as to the level of protection to be provided, given that all levels of protection achieve doses below the relevant dose limits. Optimization provides a basic framework of thinking - that it is proper to carry out some kind of balancing of the resources put into protection, and the level of protection obtained, against a background of other factors and constraints, so as to obtain the best that can be achieved in the circumstances.

The procedures used in the more formal studies of optimization of protection are all one form or another of "decision-aiding" procedure. They are intended to clarify, for the people who have to decide on the level of protection, the various factors involved, to quantify them where this is reasonable and necessary and to systematize the trade-offs between the various factors.

OPTIMIZATION AND DECISION MAKING

In its first major report on optimization (1) ICRP stated the general concept in broad terms but the report focussed on the use of cost-benefit analysis as a technique to be used in analysing protection problems to determine the optimum solution. It was probably useful in introducing the ideas of balance and trade-off to concentrate on that reasonably simple and quantified technique. In a later report (2) however, it was shown how the general concept could be implemented at different levels of decision and in different contexts using various appropriate techniques, one of which is cost-benefit analysis. The main message of this second report was that the optimization process was a way of structuring the inputs into decisions and the result - the decision - was independent of technique but was determined by the inputs to the decision and the relative evaluation of those inputs against predetermined criteria. In this context the value of unit collective dose or "α" is one such predetermined relative evaluation criterion. This structured approach is shown in Figure 1, which also makes it clear that while optimization of radiological protection can encompass some of the relevant factors there will often exist others that affect the final decision. It is the inclusion, or more often exclusion, of these other factors that has led to difficulties in obtaining acceptance for decisions.

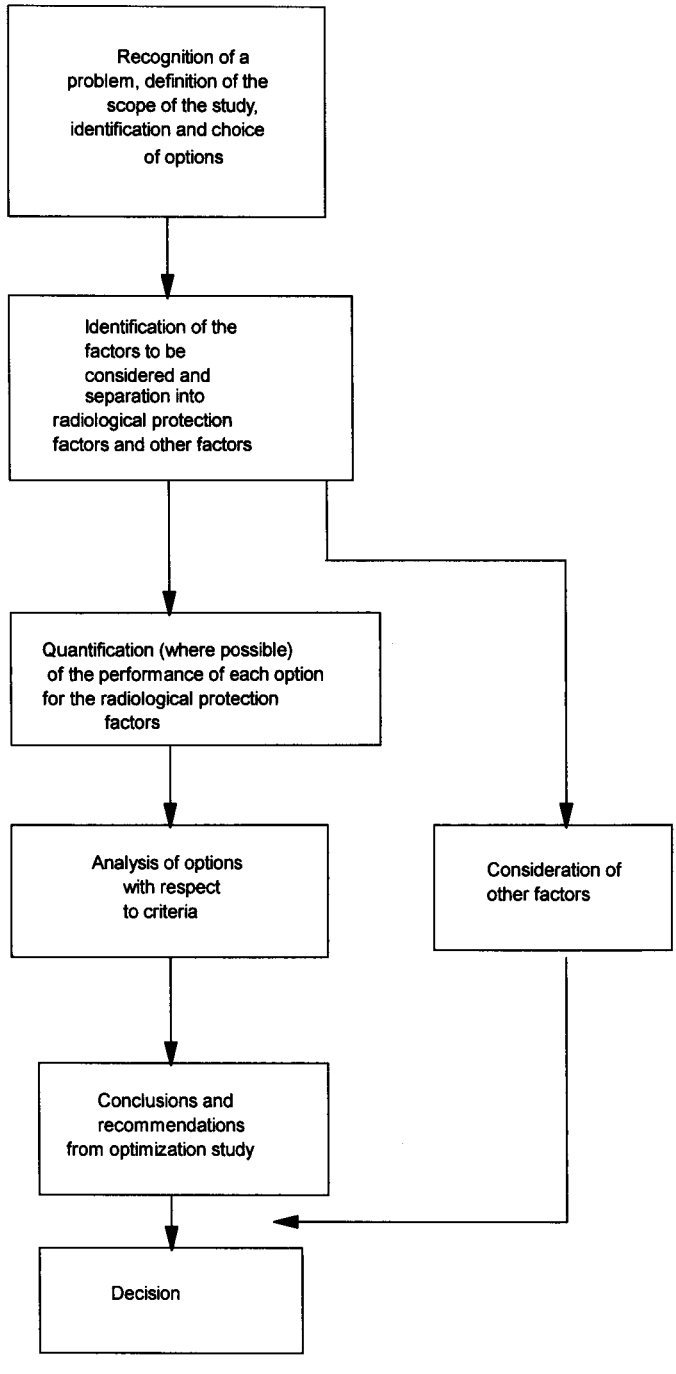

Figure 1. Structured approach to optimization of protection in the context of decision making

RISK EVALUATION IN THE OPTIMIZATION PROCESS

Some aspects of risk evaluation are clearly fundamental to and fully incorporated in the optimization of protection. Even taking the basic measure of detriment - dose - the quantity effective dose equivalent incorporates social judgements. The major such judgements built into the quantity relate to the relative importance given to mortality, morbidity and hereditary disease through the tissue weighting factors, notably those for gonads, thyroids and skin.

Other aspects are incorporated in the trade-off between dose reduction and the cost of achieving it. These components are incorporated in the value of unit collective dose and the broad judgements made are described in a companion paper for this Seminar (3).

In this system collective dose is generally not given additional weighting as a result of any risk "evaluation" and is usually costed largely on the basis of lost statistical production through unidentified premature death. Individual doses approaching the dose limits are sometimes weighted higher to account for some "aversion" to higher doses.

The general characteristic of the evaluations incorporated in procedures for optimization of protection, whether based on cost-benefit analysis (1) or multi-attribute utility analysis (2) are that they relate to the amount of radiation dose and the associated detriment irrespective of the source of the dose and normally irrespective of the recipient of the dose. This is not so much a criticism of the procedures themselves, which are in principle capable of incorporating further judgements as inputs, as it is of the users of the procedures and their willingness to recognize and take account of attitudes.

The other aspect of optimization of protection when applied according to the ICRP guidance, is that each problem has a unique solution (or optimized level of protection). The procedure relates and balances all the inputs, and because these will differ for each problem studied, the outcome of each optimization study will differ in the level of protection that is deemed to be reasonably achievable.

DIFFICULTIES WITH OPTIMIZATION

Application of this system of optimization therefore raises two major questions:

1. Does the application of these techniques adequately reflect the attributes attached to the risk by public and politicians?

These attributes include:

The source of the doses, whether this is for example, a hospital x-ray set, the discharges from a power reactor or the residues from weapons testing affects peoples attitudes (evaluation of risks and benefits) even though the doses may be identical.

The voluntariness or perceived control over the doses whether for example from a discharge to the atmosphere or from choosing to take a transatlantic flight.

Whether the doses result from deliberate acts(practices), from accidents or are "natural".

The time of delivery of the doses, now or in the far future and the associated uncertainty.

Considering only these attributes the answer to this question at the present is that they are not adequately included in optimization, and this may be one of the reasons why proposed decisions that are quite acceptable to the radiation protection community may not be so acceptable to the other communities. To modify the optimization procedures so that more "protection" is explicitly allocated to sources that people do not like is seen by some purists as a misallocation of resources but it could be that it is a better reflection

of society's wishes in the allocation. The problem is not with the procedures but in deciding what inputs to the decisions we want to take into account and how we evaluate the relative importance of these inputs.

2. Is the specificity of the optimization process, which may be reasonable when dealing with the control of doses to individual workers, appropriate when dealing with dose control for members of the public?

It is theoretically clear that the circumstances will affect the result of any particular optimization study. For example the optimized levels of discharge to the environment will depend on the pathways to man and the rates of transfer of radionuclides along these pathways, the costs of discharge reduction and the availability and costs of other options such as storage on site. These will generally vary from site to site so the outputs of optimization studies will be site and installation specific. As another example the levels to which contaminated environments should be cleaned up will differ depending on the cost and complexity of decontamination procedures, the habits including food consumption of the local population and even the extent of the contaminated area.

However, it is becoming increasingly clear that the public, politicians and decision-makers see such differences as confusing and undermining confidence rather than as evidence of scientific rigor.

It is very difficult to explain why it is safe to live in a contaminated environment in a dose rate of - say - 5mSv/a (deemed appropriate as result of an optimization study) while in another environment decontamination has been carried out to reduce doses to less that 1 mSv/a (even if this is also based on an optimization and not a misguided application of the dose limit for the public). How should both levels be compared with the pressure for intervention for Radon in homes at levels which are generally around 10 mSv/a or the lack of interest in moving entire populations from regions of high natural background such as Kerala?

We may have it theoretically correct but is that comforting when "it" is so difficult to explain and defend? Possibly there is a greater role than so far adopted for generic values. One example that was essentially imposed on the radiological protection community after Chernobyl by the politicians is the Codex Alimentarius values for contamination of foodstuffs - we may feel it is too simple and crude but we cannot fail to admit that it works.

The answer to the second question is again probably no but there may in this case be a more reasonably achievable solution, namely to adopt robust generic values rather than maintaining the fine distinctions that result from optimization studies - especially if the studies have failed to encompass all the attributes of the risk.

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Does radiation risk exist?

(Synopsis of a paper)

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RISK

Risk is related to human actions (and natural occurrences). It expresses the possibility of harm associated with a given human action, the probability of the harm and its nature and magnitude. By definition risk is a multidimensional concept. Risk is influenced by the social, economical and ecological environment in which the human actions take place.

RISK ASSESSMENT AND RISK MANAGEMENT

Risk assessment and risk management are parts of a dynamic process with the objective to decide on the tolerability of risk and on measures to keep risk within accepted limits. It enables all relevant parties to express their concerns and preferences regarding the different options for the human action involved and regarding the relative importance of criteria to decide on the tolerability of risk.

Risk assessment has three phases: problem definition, risk analysis and risk characterization. Risk analysis is primarily a technical and scientific endeavour. With regard to problem definition and risk characterization consultation between risk assessors and risk managers (and other parties concerned) is a must.

RISK ANALYSIS AND RISK CHARACTERIZATION

In the risk analysis phase the effects of the relevant physical and social stressors generated by the human action are estimated. In case ionizing radiation is one of the relevant (physical) stressors, radiation effects are studied and their probability, nature, distribution over the affected population, etc are estimated.

The results of the risk analysis are expressed in terms of relevant risk characteristics. Which characteristic is relevant is not a purely scientific question. In the case of medical treatment years of healthy life lost may be a suitable characteristic, but in the case of the siting of an airport daytime noise level, nighttime noise level, loss of nature, concentration of air pollutants, threat of crashes, etc. may be more appropriate. Which characteristics are relevant and which characteristics may be neglected is matter of judgment.

CONSEQUENCES FOR ACTIONS INVOLVING RADIATION EXPOSURE

Cancer treatment, biological research, oil exploration and generation of electrical energy are examples of human action in which exposure to ionizing radiation may be involved. The risk characterization involves usually more characteristics than only radiation dose (as a metric for the possible effects from radiation exposure). Furthermore, in deciding on the tolerability of risk the benefits of the action, including those directly related to the application of radiation, are of importance. For different types of action different criteria and norms are applied; this may hold for a judgement of the contribution of radiation exposure to the risk as well.

This viewpoint implies that the contribution of radiation exposure to the risk will, in principle, judged differently if its is associated with medical diagnosis or treatment, with air transport or with

electricity generation by nuclear energy. The risk characteristics for these types of action are different, as well as the benefits derived from the different forms of action.

How to deal with cumulative radiation exposure in this respect? Conceptually this questions pertains to the influence of the social, economical and ecological environment on the risk of a given form of action. The present data on possible effects of radiation exposure indicate that the contribution of radiation exposure to the occurrence of effects in non-accident situations ('low level exposures') is in good approximation independent from other radiation exposures. This means that different radiation exposures need not be combined; a - what Lindell calls - source related exposure would suffice. This does not mean to say that risk need not be justified; radiological examinations need not be repeated if the information is already available, even if the examination in itself was justified.

Risk has many dimensions and many metrics to quantify the dimensions (in so far they can be quantified). Too many risk characteristics may obscure decision making. A way out is to aggregate some dimensions, as has been done in defining the effective dose. In the past such aggregation has had the effect to emphasize the role radon inhalation (the change from whole body dose and separate lung dose to effective dose). However, it may be questioned if the change from the 1977 weighting factors to the 1990 weighting factors in the effective dose aggregation had any relevance. The relative dose values for given exposures changed only in a limited way. It might have been more fruitful to direct efforts to finding methods, criteria and norms to judge the tolerabilty of risk and its radiation component for different types human actions.

WHAT SHOULD WE KEEP:

- exposure quantities as risk characteristic
  - dose equivalent
  - collective dose equivalent
  - dose equivalent committment
  - effective radionuclide concentration
- criteria dependent on the type of action
  - dose constraints (for all exposure characteristics)
  - radionuclide concentration constraints
  - distributional criteria (space and time)
- risk management measures
  - knowledge and experience and quality control
  - striving for risk reduction where this is reasonably achievable

WHAT CAN WE DO WITHOUT?

- effective dose
- general dose limits for members of the public
- licensing only taking radiation exposure into account

ASSESSMENT OF OCCUPATIONAL RADIATION EXPOSURE IN CHINA

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Abstract **Since the eighties, the doses received by the workers of the nuclear industry system in China have been below 5 mSv, excluding uranium miners. Workers involved in the radioisotope and radiation applications received doses in the range of 1~2 mSv. Stringent and effective measures are required to be taken for the radioisotope and radiation applications due to high accident possibility. Average annual effective doses to underground workers in variety of occupations such as uranium, coal and non-ferrous metal mines are 19.3, 8.3 and 33.2 mSv respectively on the rough estimate basis. The nuclear industry contributes only 0.17% to collective dose. Contributions by coal and non-ferrous metal mining to collective dose account for 85.15% and 14.3% of the total respectively. The data available from coal and non-ferrous mines are less, associated with high uncertainty.**

Keywords: **Occupational exposure, nuclear industry, coal mine, non-ferrous metal mine**

Introduction

Occupational radiation exposures may cover three categories: 1) exposure of the workers of the nuclear industry system, including nuclear fuel cycle and the related researches, 2) exposure of workers engaged in the radioisotope and radiation applications, 3) exposure of workers involved in mining and milling operations. The third category involves the largest fraction of workers of such three kinds, which would lead to both the highest collective dose among others and higher annual individual dose. In the light of the fact that the data on the third category of exposure is too far from enough to accurately estimate the resulting collective doses and that the regulatory authority has not defined whether or not this category of exposure belongs to occupational exposure, this paper here is only to estimate the collective doses arising from such operations from the viewpoint of pure science.

1. Levels of radiation exposures from the nuclear industry[1]

1.1 Occupational exposure in normal condition

Table 1 lists annual collective dose equivalents and average annual individual dose equivalents to workers of the nuclear industry. The data listed in this table were in part gained from individual dose monitoring results, which were collected from reactor operation and reprocessing and the related scientific research institutions; the remainders were obtained by calculation according to the concentrations and duration of aerosol in radioactive working places related to uranium mining and milling, fuel fabrication and isotopes separation system. The doses arising from inhalation of radon progeny were estimated on the basis of a conversion factor of 10 mSv WLM^{-1}.

Table 1 Annual collective dose equivalents and average annual individual dose equivalents

Nuclear industry system	Annual collective dose equivalent				Average annual individual dose equivalents, mSv	
	Prior to 1987		1980~1987			
	man·Sv	distribution	man·Sv	distribution	Prior to 1987	1980~1987
Uranium mining	62.5	59.5	59.7	61.42	26.1	19.3
Uranium milling	3.04	2.89	5.31	5.46	1.82	1.8
Fuel fabrication	2.6	2.48	2.9	2.98	4	3
Isotope separation	0.31	0.3	0.12	0.12	0.13	0.08
Reactor operation	13.9	13.2	16.6	17.1	14.2	12.3
Reprocessing	21.6	20.56	10.6	10.9	18.9	5.9
Research institute and Isotope production	1.1	1.05	1.97	2.02	2.3	1.86
Total	105.05		97.2			

The Table 2 shows the monitoring results of individual dose arising from occupational external radiation exposures from the operation of nuclear industry, which indicates that during the period 1967~1976 the workers received the highest doses, with about 10% of workers having received doses in excess of dose limit (50 mSv). The reasons for higher doses to be received by the workers during that period are mainly as follows: 1) reactors and reprocessing plants were put into production in succession in that period, and there was a lack of operation experiences in the initial phase; 2) the implementation of various regulations and rules was seriously interrupted, which resulted in high probability of accident occurrence. Following the end of Culture Revolution that took place during 1966~76, updated various regulations and rules related to safety led to a progressive reduction of doses to the workers. Average annual individual dose dropped by a factor of 50% in the period 1977~1981 and once again decreased by a factor of 50% in the period 1982~1987 in comparison with those several years ago.

Table 2 Individual dose monitoring data of occupational external exposure due to nuclear industry

Items	Years							
	1959~66		1967~76		1977~1981		1982~1987	
	Range	Mean	Range	Mean	Range	Mean	Range	Mean
Number of individuals	289~1064	563	1265~3267	2067	3464~5356	4452	5253-6322	5751
Annual collective dose, man Sv	0.73~3.42	2.32	6.06~127.87	42	38.78~60.49	46.34	19.17~29.88	26.12
Annual effective dose, mSv	1.6~11.7	4.1	4~64.8	20.3	8.8~11.6	10.4	3.6~5.3	4.5
Distribution ratio NR_{15}, %	0~36.8	5.21	2.78~62.46	31.52	15.18~22.4	19.3	4.49~8.76	5.91
Distribution ratio NR_{50}, %	0	0	0.06~44.4	10.7	2.43~5.43	3.58	0.09~0.58	0.34

1.2 Accidental exposure

Table 3 shows the annual collective dose equivalents to the workers of the nuclear industry resulting from accidental exposures, excluding those from uranium mines, and not allowing for those from fuel element fabrication and isotope separation plants because of very small magnitude. Internal doses received by the workers arose mainly from the contamination by Pu, accounting for about 92.6% of the total; the remainders were from fission products and tritium, about 3.2% and 3.1%. The exposure pathway was principally from inhalation, which led to about 73% of the total. It is worth while to be noted that internal dose due to plutonium incorporated into body through wounds accounted for 26% of the total. The annual collective dose equivalents shown in Table 4 do not cover overexposure of local skin. During the period 1958~1987, 29 skin injury accidents occurred, involving 51 sufferers. Absorbed doses to skin were estimated to be in the range of 5~196 Gy, with 86% of irradiated area being found on the part of hand. These accidents, except for 1 ^{60}Co accident and 2 accelerator electron beam accidents, were caused by β-ray radiation.

Table 3 Annual collective dose equivalents received by workers of nuclear industry due to accidental exposures

Nuclear systems	Annual collective dose equivalents, man Sv			Percentage to total
	External exposure	Internal exposure	Total	
Reactor	2.65	2.2×10^{-2}	2.67	19.2
Reprocessing	0.31	1.25	1.56	7.2
Research institutes and Isotope production	0.22		0.22	20
Nuclear metallurgy and machining		0.46	0.46	
Total	3.18	1.73	4.91	

2. Levels of radiation exposure resulting from radioisotope and nuclear technology application

2.1 Occupational exposures in normal conditions[2]

The monitoring of individual doses to workers in the radioisotope and nuclear technology applications were carried out at a later time, which began in middle 1980's and has developed gradually. Table 4 shows the occupational radiation exposures of the workers in the field of the radioisotope and nuclear technology applications. The average annual individual dose and collective dose received by workers in industrial radiography, as shown in this table, are relatively high, which includes X ray and γ ray radiography. The internal exposures of workers of nuclear medicine in Beijing were monitored in 1994 and the results showed that the average annual individual effective dose was about 1.1×10^{-1} mSv[3] which is about a few percent of external dose.

Table 4 Occupational external exposure to workers of radioisotope and nuclear research application (1985-1989)

	Workers		Annual individual dose	Annual collective dose (man Sv)	
	monitored (10^3)	Radio monitored	equivalent (mSv)	Workers monitored	All group
Nuclear medicine	0.77	12.8	1.39	1.25	7.78
Radiotherapy	0.36	5.23	1.25	0.52	8.36
Industry Radiography	3.08	35.94	2.01	5.23	16.22
X diagnosis	13.67	14.55	2.06	27	185.5

2.2 Accident exposure

Table 5 shows the radiation accidents leading to radiation damage, which occurred in several irradiation facilities with sealed sources.

3. 1 Occupational radiation exposure of workers in underground mines

For coal mine, equilibrium equivalent concentration (EEC) of radon in underground mines is estimated to be 250 Bq m^{-3}, that would result in average individual dose of 8.3 Sv a^{-1}, and collective effective dose of 5×10^4 man Sv a^{-1} to 6×10^6 underground workers nationwide. For non-ferrous metal mines, if taking typical value of EEC of radon to be 1×10^3 Bq m^{-3}, annual individual effective dose is estimated to be 33.2 mSv and annual collective effective dose to be 8.4×10^3 man Sv to 0.25×10^6 underground miners. For other kinds of mines, the estimating of dose resulting from radon and its daughters is difficult due to less data available from these mines, but high concentrations of radon and its daughters in these mines were found extensively, which may be up to 3.2×10^4 Bq m^{-3}.

Table 5 Accidents leading to irradiation damage at irradiation facilities with sealed sources[3]

Date	Place	Summary	Consequence
29-February, 72	Sichuan	Accidentally entered the irradiation room with a 2.64×10^{14} Bq ^{60}Co source.	resulted in 55~147 rad radiation exposure
December, 72	Wuhan	A 5×10^{12} Bq ^{60}Co source for radiotherapy purpose fell off on copper filter plate, lasting 16 days.	20 patients and 8 workers received doses in range of 5~147 rad dose
September, 80	Shanghai	Entered the irradiation room with a 2.2 $\times10^{15}$ Bq ^{60}Co source due to power failure and interlockout of order.	A worker incurred a 5.22 Gy acute radiation exposure
June, 85	Shanghai	Entered a 1.5 MeV Van de Graff accelerator hall while main motor running.	A worker incurred a 25~210 Gy radiation exposure
March, 86	Beijing	Entered a 0.2 pBq ^{60}Co source room when it is on irradiation position while driven system in failure and door open.	2 workers received doses of 0.7 and 0.8 Sv respectively
May, 86	Hena	Entered a 0.3 Bq ^{60}Co source room while it is on irradiation position.	2 workers incurred 3.5 and 2.6 Gy acute exposure respectively
March, 88	Liaoning	Took source by bare hands when removing a radiographic source of 1.1×10^{12} Bq in failure.	6 workers received doses in range of 0.1~126 Gy on hands.
June, 90	shanghai	Entered a 0.85 pBq ^{60}Co source room when a protection door broken and taken away while the another in failure due to power loss.	7 workers were exposed to 2~12 Gy radiation, resuling in 2 deaths
November 92	Wuhan	Interlock out of order.	4 workers were exposed to radiation, 1 of them incurred acute radiological sickness

4. Conclusions and discussions

Table 6 shows the average annual individual dose and annual collective dose to workers of various occupations arising from occupational radiation exposures in China, which were typical values in 1980's on the whole. As shown in the table, the underground miners in non-ferrous metal mines received the highest annual individual dose among others, even higher than uranium miners by 65%, with the coal miners' dose ranking the second; coal miners received the highest annual collective dose, followed by non-ferrous metal miners; sum of both accounts for 99.45% of the occupational radiation dose. Though the relevant Chinese regulatory body has not yet defined the exposures of these kinds as part of occupational exposure it is necessary, from

the viewpoint of people's health, to take effective measures to reduce the concentrations of radon in working places, especially for non-ferrous metal miners.

Table 6 Average annual individual doses and annual collective doses to occupational workers in China

Categories of occupations	Average annual individual dose (mSv a^{-1})	Annual collective dose	
		Dose (man Sv)	Ratio of dose to dose sum (%)
Uranium mine	~20	$\sim 1\times10^{2}$	0.17
Nuclear stages other rather uranium mine	4.5	$\sim 2.2\times10^{2}$	0.37
Nuclear and radiation applications	2	5×10^{4}	85.15
Coal mine	8.3		
Non-ferrous metal mine	~33	8.4×10^{3}	14.3
		5.872×104	

It should be noted that (1) the stringent and effective measures should be taken in nuclear technology and radiation applications because of highly potential exposure and high frequency of accident occurrence though less annual individal doses in normal conditions; (2) although the nuclear industry has good safety record, there would be possibility of severe accident with very low probability of occurrence but not equal to zero so that to prevent the accident from occurring by reducing the dose to workers and by reinforcing management will be necessary; and (3) the data given in Table 6 is associated with uncertainty to a great extent, mainly from non-ferrous metal mines and coal mines and for the purpose of accurate dose estimate much more typical data are required. Rough estimates indicate that more attention should be paid to this problem.

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IRPA9

1996 International Congress on Radiation Protection

Proceedings / Volume 2

Vienna, Austria, April 14-19, 1996

Organised by the Austrian Association for Radiation Protection

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EUROPEAN RESEARCH INTO THE MEASUREMENT OF EXTERNAL ENVIRONMENTAL RADIATION DOSES

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INTRODUCTION

In a Commission of the European Communities (CEC) research programme spanning the past 10 years all aspects of external environmental radiation dose measurements have been thoroughly investigated (1,2). The aim of this work was to harmonize environmental radiation dose measurements within different countries. The initial studies involved research into practical calibration techniques and this led to the establishment of two unique calibration facilities. One facility developed by the Physikalisch Technische Bundesanstalt (PTB), Braunschweig, Germany in the Asse salt mine at a depth of 970 m where there is an extremely low level of environmental radiation (3). The other calibration facility is the Risø National Terrestrial Radiation Measurement Station (4) where reference measurements of a well determined natural environmental photon spectrum can be made. At Risø a cosmic measurement station has also been established at a wooden pier extending into the Roskilde Fjord, this station is used for the determination of a detector's response to cosmic radiation.

This paper describes the calibration techniques that have been investigated and the intercomparisons that have been performed. The relevance of this research programme was demonstrated by an intercomparison of external photon dose rate measurements by several detectors in the vicinity of a nuclear power station.

CALIBRATION AND TYPE TESTING

Many commercial dose rate instruments, electronic dosemeters and most thermoluminescence dosemeters (TLDs) were calibrated and radiation type tested. These tests included the determination of the inherent responses of the detectors using the Asse facility, namely the contribution to the readings from the active and passive detectors from internal radioactive contamination or from electronic noise.

The energy responses of all the detectors were measured over the photon energy range from 60 keV to 6 MeV.

The cosmic responses of the detectors were evaluated from measurements made on board boats in the North Sea and in the Roskilde Fjord and these results were then compared with measurements made at the Risø cosmic field station.

Free-field and shadow-shield calibration techniques were intensively studied at Risø and several intercalibration experiments using these techniques were carried out among the participating States. Monte Carlo (MC) calculations using the MCNP (Monte Carlo Neutron Photon) code were used to determine the air kerma rates produced by scattered photons from ^{137}Cs, ^{60}Co and ^{226}Ra sources as a result of surrounding media in different free-field and shadow-shield calibration geometries (6).

INTERCOMPARISONS

a) The assessment of external photon dose rates in the vicinity of nuclear power stations.

In 1991 experiments were carried out to investigate the responses of instruments to additional exposures above normal environmental levels. Four environmental dose rate instruments having different detectors, a high pressure ionization chamber, a Geiger-Müller counter, a proportional counter, and a scintillation counter, were used to make continuous measurements over a four month period of the air kerma rate at a location close to the Hinkley Point Nuclear Power Plant in the UK. Three types of TL dosemeters normally used for monitoring environmental radiation were also used to compare the responses of solid state passive dosemeters with those of the active dose rate meters.

The results clearly demonstrated that accurate estimations os doses in the environment arising from a nuclear facility can only be obtained if continuous measurements are made and if the responses of the detectors used to the different radiation components at that location are accurately evaluated. By correcting the measured air kerma values by the accurately determined detector responses the standard deviation, expressed as a percentage of the

mean value of the total air kerma for the instruments and TLDs was reduced from 20% to 5% (5).

b) An international comparison of active and passive detectors used for environmental radiation measurements.

During 1994 an international intercalibration experiment was performed at Risø National Laboratory with participants from the USA, Eastern and Central Europe and the EU member countries. One aim of the experiment was to intercompare the "home" calibration of detectors and dosemeter responses of environmental radiation monitors used in the above countries and to try to make a link between the different reference standards used. A special feature of this recent intercomparison was the testing of a variety of recent designs of electronic dosemeters to see how they respond to environmental radiation. Another important feature was the use of highly sensitive TLD materials over ultra short integration periods (e.g. 3 hours) to determine the natural background air kerma rate. A total of 12 active dose rate meters, 2 spectrometers, 21 electronic dosemeters and 8 different TL materials were used in the experiments which included (i) free-field and shadow-shield calibrations, (ii) measurement of the natural radiation at the terrestrial field site at Risø and (iii) measurement of the cosmic ray component both on a platform at sea and on a boat at sea.

Table 1. Summary of mean results of air kerma rates from field, cosmic and terrestrial radiations, respectively, for each detector type.

Detector Type (Dosemeter type)	Field, cosmic + terrestrial ($nGy.h^{-1}$)	Cosmic (sea) ($nGy.h^{-1}$)	Terrestrial - field - cosmic ($nGy.h^{-1}$)
HPICs (Dose rate meter)	74.4±1.9	36.4±1.6	38.0±0.8
Pl. Scintillators (Dose rate meter)	71.4±7.8	31.5±6.5	39.8±3.4
GM (Dose rate meter)	68.7	37.8	30.9
Spectrometer (NaI)	45.9±1.9	5.7±1.1	39.9±2.2
TLDs, MCPN	70.0±0.3	28.7±0.1*	41.4±0.4
TLDs, GR-200	67.9±3.6	29.8±1.6*	37.7±1.4
TLDs, Al_2O_3:C	102.5±34.6	37.2±9.6*	65.3±25.0
TLDs, $CaSO_4$:Dy	72.0	30.4*	41.6
Electronic dosemeters (GM)	130±22.4	96.7±19.1*	34.0±4.8
Electronic Dosemeters (Solid state)	96.3±62.5	61.0±53.7*	35.3±12.7

* Estimated from measurements made at the Cosmic Pier Station
N.B. The ± values represent 1 SD.

Table 1 shows the mean results for terrestrial and cosmic measurements for each detector type. Substraction of the cosmic (sea) responses from the field measurement responses yielded mean values of the air kerma rates due to the terrestrial radiation component.

Apart from the Al_2O_3:C material, which will be investigated in future intercomparisons, the TLD results for the terrestrial radiation show good agreement with the active dosemeter results.

CONCLUDING REMARKS

The considerable experience and expertise gained by the participants during the CEC research programme is to be disseminated in a European Radiation Dosimetry Group (EURADOS) Technical Recommendation. This publication will give practical advice on all aspects of environmental photon dose monitoring.

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TEMPTS TO DETERMINE RADON ENTRY RATE AND AIR EXCHANGE RATE VARIABLE IN TIME FROM THE TIME COURSE OF INDOOR RADON CONCENTRATION

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INTRODUCTION

For the study and explanation of the diurnal variability of the indoor radon concentration a(t) [Bq/m^3], which is proportional to the ratio of the radon entry rate A [Bq/h] and the air exchange rate k [1/h], it would be of advantage to know separatetely the diurnal variability of both determining quantities A(t) and k(t). To measure directly and continuously the radon entry rate A(t) is possible only in special studies (mostly in experimental rooms) and also continuous measuring of the air exchange rate k(t) is possible also only in special studies for a short time. But continuously measuring radon meters are now common, do not trouble people in normal living regime during day and night. The goal of this endeavour would be the evaluation of the time courses of both determining quantities from the time courses of the indoor radon concentration directly without additional experimental work and so a better utilisation of such measurements.

APPROACHES TO THE TIME ANALYSIS OF THE RADON CONCENTRATION

The variation in time of the radon concentration a(t) in a room with volume V (m^3), radon entry rate A(t) and air exchange rate k(t) is studied here using the linear (compartmental) model

$$a(t)' = - k(t)\, a(t) + A(t)/V \,, \qquad (1)$$

expressing the balance of removal and entry of radon by the differentiate a' [Bq/(h m^3)] and assuming that the air exchange rate is much more quicker than the decay constant of radon. This differential equation has for the initial condition $a(t_0)$ a general analytical solution which could be used in principal for the determination of the unknown functions A(t) and k(t) by numerical methods. This approach seems to be very complicated and unrealistic. The other way is to use the differential equation itself.

Mostly the entry and air exchange rates are evaluated with the assumption of their constancy in time during the measuring period. The solution is in this case simple

$$a(t) = A(1-e^{-kt})/kV + a_0 e^{-kt} \quad \text{for } t>0=t_0 \qquad (2)$$

and the three unknown values a_0, k, and A can be obtained by numerical methods from three concentrations a_1,a_2,a_3 measured in times t_1,t_2,t_3 (at least two of them not in steady-state). The insufficiency of this approach is in the unability to assure the constancy of k and A for a longer time than several hours.

DETERMINISTIC SOLUTION OF AN EQUIVALENT DIFFERENCE EQUATION

Measurements result in average concentrations a_i during the i-th time interval $\{t_i,t_i+d\}$ of duration d [h], therefore also average values A_i and k_i can be only obtained from difference equations equivalent to the differential equation:

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where the average differences a_i' have to be calculated by a numerical calcul from the measured, or better from smoothed values a_i. To calculate the two unknown values k_i and A_i one needs to evaluate at least two adjacent intervals,i.e. two difference equations. Assuming

the same values k and A for both intervals of a concavely increasing course or a convexly decreasing course of experimental values gives good results but fails for convexly increasing and concavely decreasing courses of the radon concentration resulting in meaningless negative air exchange rates. The reason can be understood from the concavely increasing or convexly decreasing time courses of the exact solution (2) of the differential equation assuming constant k and A.

One possibility to extend the applicability of this approach is to assume linear changes for both quantities $k_i = k + \kappa t_i$ and $A_i = A + \alpha t_i$ - decreasing or increasing in accordance to the sign of κ and α. Then four equations for four adjacent intervals have to be used for a general deterministic evaluation. Three equations can be used if the change in time of one of the quantity can be neglected and two equations are enough, if A and k are nearly the same for two adjacent intervals and the courses are properly concavely increasing or convexly decreasing.

STATISTICAL SOLUTION OF THE EQUIVALENT DIFFERENCE EQUATION

The deterministic approach has two important disadvantages. First the "uncompromising" fit of the deterministic solution to the uncertain input data a_i and a_i, gives manytimes completly meaningless results, second it does not give uncertainties for the calculated results. This can be overwhelmed by a linear regression analysis using the difference equation as the statistical model with index i as the independant variable. This approach gives beside of the four mutually interdependant results A, k , α and κ also their covariance matrix, of course the price for this is a larger number of intervals used for the evaluation with the same tendency of linear change assumed for the entry rate and air exchange rate. The optimal number of intervals suitable for evaluation can be determined by computer using the covariance matrix. Setting the difference equations symetrically over the used intervals and shifting automatically the evaluation to the next interval the evaluation should give reasonable results, but only for properly monotonically increasing or decreasing radon concentrations.

Unfortunately the statistical approach results sometimes in multifolds of the real values k and A, of course with the right ratio for the steady state radon concentration. The reason for this is that the minimalisation process finds the lowest minimum but off the real region. Overwhelming of this disadvantage needs to use constraints, but the evaluation gets more and more complicated. Another disadvantage of the approaches is that it cannot be applied around maximal and minimal radon concentrations because a monotonic tendency of the radon entry rate or the air exchange rate is not justified.

STOCHASTIC APPROACH

A promising approach to evaluate the time courses of the radon entry rate and of the air exchange rate from the whole time course of the measured radon concentration is hoped to be the stochastic approach used for simulation of sequences of data , the theory of which is described e.g. in (1). The equation (3) is the generic model for the evaluation and unrealistic solutions are avoided by limits for the air exchange rate and by realistic estimated starting values.

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The compartmental or linear kinetic model used for the description of the indoor radon concentration assumes a perfect and immediate mixing of the entered radon in the whole

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CONCLUSION

Two approaches are given for the determination of variable in time radon entry rates and air exchange rates from continuously measured indoor radon concentrations by numerical solution of the equivalent difference equations in deterministic or statistic form. Unfortunately these approaches are not always successful. But the analysis of the reasons for the failures leeds to a better understanding of the problem. It is hoped that the more sophisticated simulation approach shall solve the problem.

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PAPER TITLE

COMPARISON BETWEEN LARGE SCALE RADON RISK MAPS AND RESULTS OF DETAILED RADON SURVEYS

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A large number of detailed measurements is available for testing the reliability of regional radon risk maps. The comparison illustrates general reliability of large scale radon risk maps on the significant probability level. On the other hand, the local variability caused by a number of geological and nongeological factors confirms the necessity of detailed measurements for proposing the optimal remedial technology.

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Author 20294
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Mini-Presentation

PAPER TITLE Use of SSNTD for radon measurement in the Czech Republic

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MAJOR SCIENTIFIC TOPIC NUMBER 1.1. (see page 7)

ABSTRACT (See instructions overleaf)

In the Czech Republic striking number of solid state nuclear track detectors (SSNTD) has been used in last five years (more than 150 000). The bare detectors Kodak LR115 have been choiced for year-mean estimation. The investigation level in the Czech Republic is defined for EEC (radon progeny). Therefore the response (track density) depending on factor F must be corrected.

The uncertainty caused by the differing F is greater than for the case of SSNTD in diffusion chamber, but still acceptable for this purpose (searching houses with elevated EEC).

One has prefered to place detectors in radon-prone areas (defined with help of geology etc.). So the mean of results is higher than in other countries: 95 $Bq.m^{-3}$ with GSD 2.5 for 115 000 detectors evaluated to the end of 1994. The increase of mean in the past is surprising - we suppose it is caused by tightening of dwellings etc.

There are regions with elevated EEC level - medieval mining town Jachymov with mean about 500 $Bq.m^{-3}$ and villages round the pluton body in Central Bohemia (so called Devil Burden), characterized by mean about 300 $Bq.m^{-3}$. In these regions remedial actions were started.

ON THE SIMULTANEOUS INDOOR MEASUREMENT OF THE EXHALATION RATES OF RADON GAS FROM OPEN SOIL AND THE CONCENTRATION OF RADON PROGENY IN THE SURROUNDING AIR

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INTRODUCTION

It is well known that the outdoor measurements of radon exhalation rates and of radon concentration in atmosphere are affected by the temperature, wind and atmospheric pressure (1). So far, no clear correlations have been found between these measurements. However, with indoor measurements, especially those recorded in a basement, the temperature is constant and the air flow minimal. This means that we can expect radon exhalation rates and the concentration of radon progeny to be affected by atmospheric pressure alone.

Taking these environmental conditions into account, the simultaneous measurement of radon exhalation rates and radon concentrations was performed in the basement of SUT (Science University of Tokyo), Oshamanbe campus in order to clarify the nature of the relationship between radon exhalation rates and concentrations and changes in atmospheric pressure. Atmospheric pressure was also measured simultaneously with the above measurements. The present measurements were carried out during the winter, spring, summer and autumn of 1995. Measurements were made in a series of one week periods.

EXPERIMENT

As far as we know, there are few studies based on indoor measurements of radon exhalation rates from the soil. This is because the exhalation source, that is the soil, is essential to measurement, and earth-floored basement areas are seldom found in modern building construction. Fortunately, this campus has a large earth-floored basement room (24 m X 22.4 m X 2.6 m). This was the ideal environment for the measurement of radon exhalation rates from the soil and radon progeny concentration in the surrounding air. Our basement has an exhaust fan, which is turned off between midnight and 5:00 a.m.

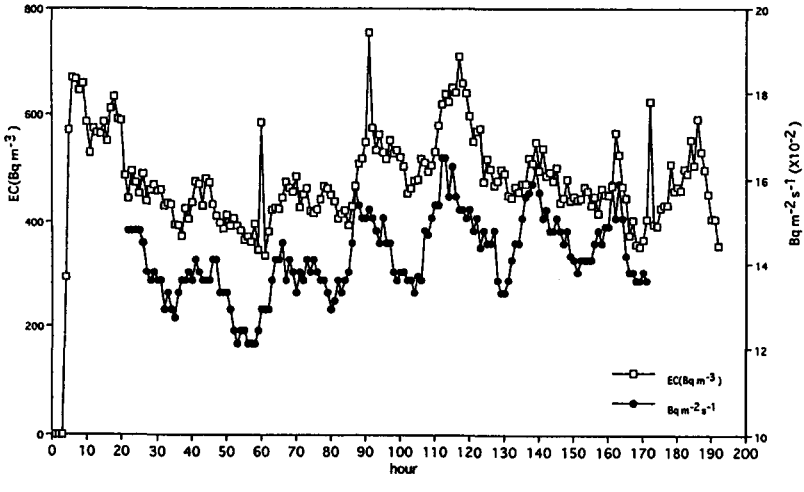

Figure 1. Simultaneous measurement of radon exhalation rates (Bq $m^{-2}s^{-1}$) and radon concentrations [EC(Bq m^{-3})], 17th-23rd May 1995.

The exhalation rates of radon gas from the soil were measured using an ionization chamber (2), whose efficiency was calibrated with a standard radon gas. The ionization currents yielded in the ionization chamber were continuously marked on recording paper. The ionization currents were then converted into exhalation rates in units of Bq $m^{-2}s^{-1}$. The collector of the ionization chamber (covering area 0.85 m X 0.25 m) was placed on the ground surface around the center of the basement room.

The concentrations of radon progeny in the surrounding air were measured using a Radon WL Meter (Thomson & Nielsen Electronics Ltd). The α - ray counts in the Radon WL Meter were measured as they accumulated every hour, and the data processed by an internal computer program. The results were printed out on the recording paper in units of EC(Bq m^{-3}), i.e. Equilibrium-Equivalent Concentration. In order to monitor the concentration of radon gas in the surrounding air near the Radon WL Meter, a dozen "Rad Track" detectors were also fixed at the same time near the inlet of the Radon WL Meter. The Radon WL Meter was set on a table at a height of 0.8 m.

RESULTS and DISCUSSION

As shown in Figure 1, it was found that the upward and downward trends of the radon concentration obtained in the surrounding air were correlative to variations in the radon exhalation rates from the open soil. In other words, the radon concentrations were found to be in phase with the radon exhalation. Sporadic spikes were observed in the tracking curves of the radon progeny concentrations, as shown in Figure 1. The reason for their occurrence is not apparent to the present authors.

Atmospheric pressure affected the yields both of the radon gas from the soil and of the radon progeny concentrations in the air. Whenever atmospheric pressure increased, both the yields of radon gas and of radon concentration decreased, and vice versa. That is, the former tracking curve varied with the latter curves in an inverse relation. An example of this phenomenon is shown in Figure 2.

Periodic daily variations were clearly observed in the tracking curves of both the exhalation rates of radon gas and the radon concentrations, as shown in Figure 2 and Figure 3. The peaks of the radon exhalation rates in Figure 2 and of the radon concentration of (no.1) and (no.2) data in Figure 3 occurred at about 6:00 a.m. every day. The data of (no.1) and (no.2) were obtained simultaneously by two different Radon WL Meters, which were set on tables at the center and corner points of the basement room, respectively.

Seasonal variations in radon exhalation rates from the soil are shown in Figure 4. The autumn level of the radon exhalation rates was the highest, while those in winter and spring were median. The lowest rates were exhibited by the data observed in the summer.

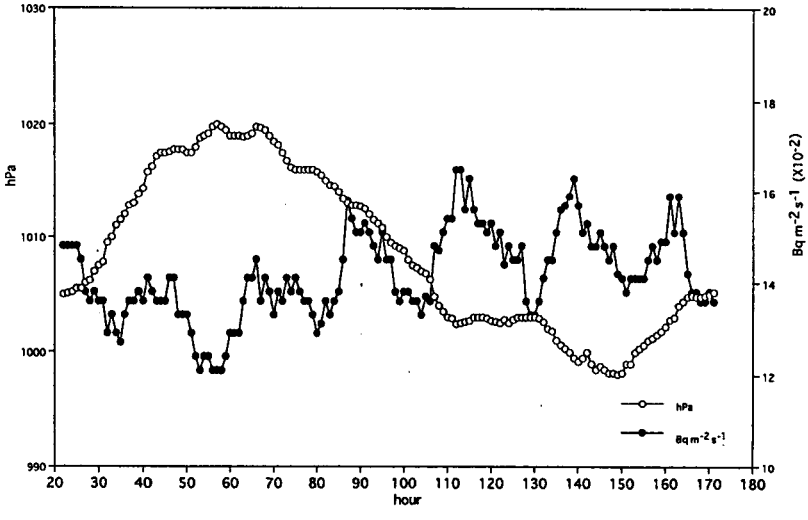

Figure 2. Dependence of radon exhalation rates (Bq $m^{-2}s^{-1}$) from soil on changes in atmospheric pressure (hPa), May 17th-23rd 1995.

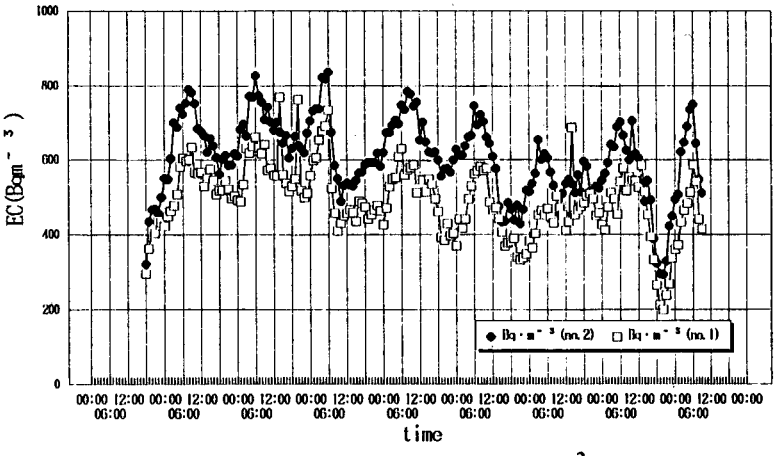

Figure 3. Daily variation of radon concentration [EC(Bq m^{-3}] in air, July 28th-August 5th 1995. The data of (no.1) and (no.2) were obtained at the center and corner points, respectively, in the basement room. Two different Radon WL Meters were used simultaneously. The peaks of radon concentrations occurred at about 6: 00 a.m. daily.

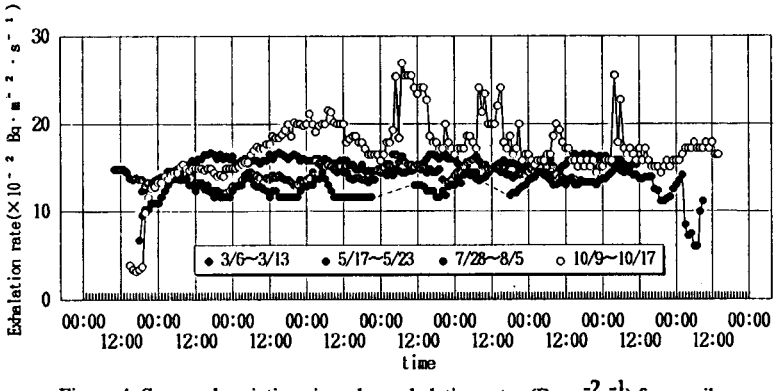

Figure 4. Seasonal variations in radon exhalation rates (Bq $m^{-2}s^{-1}$) from soil during winter, spring, summer and autumn 1995.

The radon concentrations (Bq m^{-3}) obtained by "Rad Track" detectors were compared with the integrated values estimated from the tracking curves of the radon concentrations [EC(Bq m^{-3})]. The results show that the latter values were about 1.7 times greater than the former.

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EXHALATION OF RADON AND THORON FROM PHOSPHOGYPSUM USED AS BUILDING MATERIAL

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ABSTRACT

The radioactive properties of two types of phosphogypsum, were determined. Gypsum plates with different thickness were produced. The ^{226}Ra and ^{232}Th concentrations were measured by means of high resolution gamma spectrometry. The results are for type 1 ^{226}Ra : 75 Bq/kg and ^{232}Th 230 Bq/kg and for type 2 ^{226}Ra : 155 Bq/kg and ^{232}Th : 160 Bq/kg. The radon (^{222}Rn) exhalation rate was evaluated by closing the plates in airtight barrels and measuring the radon concentration. The exhalation rate of type 1 is 1.2 10^{-5} Bq/(kg s) and type 2 4.7 10^{-5} Bq/(kg s). In combination with the ^{226}Ra concentration an emanating fraction of respectively 7.6% and 14% is obtained.

The ^{220}Rn (thoron) exhalation of the plates was determined by measuring the concentration of the decay products in a chamber of 1 m^3 with normal aerosol concentrations. The exhalation rate was found to be independent of the thickness of the plates, as expected from the short half-life of ^{220}Rn. Covering the entire surface of the plates with two layers of a common Latex paint decreased the thoron exhalation by a factor of 10 to 20.

The laboratory results for the radon and thoron exhalation were converted using realistic assumptions for a room. The contribution of phosphogypsum to the average radon concentration in a room is found to be about 1 Bq/m^3 for type 1 and 4 Bq/m^3 for type 2 resulting in an annual effective dose of the order of 0.1 mSv/year. The contribution to the effective dose from the thoron exhalation is much greater, namely, 1.8 mSv/year for type 1 and 0.9 mSv/year for type 2. Painting the gypsum lowers the thoron dose by a factor of 10 to 20 making the thoron dose comparable to that of radon.

INTRODUCTION

Gypsum is widely used in Belgium as a building material. The walls in most dwellings are covered with a layer of 1 to 2 cm of gypsum before they are painted or wallpapered. A considerable fraction of the gypsum applied in Belgian dwellings comes from the phosphate industry. The reaction of phosphate ore with sulfuric acid yields a variety of phosphate products and calcium-sulfate (gypsum). On average 4 kg of gypsum are produced for each kg of phosphoric acid (1). The phosphate ore has generally a high content of ^{238}U and/or ^{232}Th in radioactive equilibium with its daughter products. Most of this radioactivity will be found in the gypsum waste which is called phosphogypsum.

The paper consists of two parts. In the first part the radioactive properties of two types of phosphogypsum are determined. Results are given concerning the ^{226}Ra and ^{232}Th concentration, the radon (^{222}Rn) exhalation rate and the thoron (^{220}Rn) exhalation with and without painting. The radiological impact of using these types of phosphogypsum in home construction are discussed in the second part.

METHODS AND RESULTS

From each of the two types of phosphogypsum 60 plates with a surface of 0.295 x 0.210 m were made. Half of the plates was produced with a thickness of 0.5 cm and the other half with a thickness of 2 cm.

The ^{226}Ra and ^{232}Th concentrations of the two types of phosphogypsum were measured by means of high resolution gamma spectrometry. The detector of the measurement system is a 20% efficient germanium cristal with a resolution of 1.70 keV (fwhm). The detector is calibrated in the same geometry with a ^{134}Cs source traceable to NIST. The results for the two types are:

Type 1 : ^{226}Ra : 75 Bq/kg ^{232}Th : 230 Bq/kg

Type 2 : ^{226}Ra : 155 Bq/kg ^{232}Th : 160 Bq/kg

The radium concentrations are lower and the thorium concentrations are higher than those found in phosphogypsum from Moroccan origin (1).

The radon (^{222}Rn) exhalation rate was measured by closing some of the plates in an airtight barrel flushed previously with radon free air. One week or more later, the radon concentration in the barrel was determined by means of the Lucas technique. From the dimensions of the barrel, the weight of the sample and the radon concentration the exhalation rate was calculated:

Type 1 : 1.2 10^{-5} Bq/(kg s)

Type 2 : 4.7 10^{-5} Bq/(kg s)

In combination with the ^{226}Ra concentration an emanating fraction of 7.6% for type 1 and 14% for type 2 is obtained.

The thoron exhalation (^{220}Rn) of the plates was evaluated by means of its decay products because of the short half-life of thoron (55.6 s). The plates were brought into a 1 m^3 chamber which was ventilated continuously at a rate between 0.3 to 0.4 h^{-1} with outside air containing natural aerosol particles. The outside air was taken at a height of 10 m where the thoron and thoron decay product concentrations are very low. Each sample was measured 3 to 4 times. The reproducibility of the equilibrium equivalent thoron concentration was about 25% due to fluctuations in the aerosol concentration in the chamber. The equilibrium equivalent thoron concentration (EEC_{Th}) is calculated from the ^{212}Pb and ^{212}Bi concentrations in the following way:

$EEC_{Th} = 0.91\ C\ (^{212}Pb) + 0.09\ C\ (^{212}Bi)$

The results given in table 1 are normalised to an exhalation surface of 4 m^2 and corrected for the influence of humidity on the exhalation of the freshly produced gypsum plates.

TABLE 1. Equilibrium equivalent thoron concentration of the phosphogypsum plates with and without painting in the experimental chamber

Type	Thickness cm	Not painted EEC_{Th} Bq/m^3	Painted EEC_{Th} Bq/m^3	Ratio Not painted / Painted
1	0.5	35	2.2	16
1	2.0	25	2.3	11
2	0.5	14	0.7	20
2	2.0	12	1.0	12

The thoron exhalation is, as expected from the short half-life of ^{220}Rn, independed of the thickness of the samples. The exhalation of type 1 is about two times higher than that of type 2. This is more or less in line with the measured ^{232}Th concentrations.

The effect of covering the entire surface of the plates with two layers of a common Latex paint was also investigated. The set-up was identical to the measurements of the un-painted plates. The measured equilibrium equivalent thoron concentrations are shown in table 1. The reduction in thoron exhalation by painting the gypsum plates is a factor of 10 to 20.

DISCUSSION

In order to evaluate the radon and thoron exhalation of the phosphogypsum plates, the laboratory results were converted using realistic assumptions for a room:

- size of the room : 4 x 5 x 3 m (60 m^3);
- ventilation rate : 0.5 h^{-1} ;
- walls and ceiling covered with type 1 or type 2 phosphogypsum (74 m^2);
- residence time : 80%.

Covering the walls and the ceiling of the room with 1 cm of phosphogypsum increases the radon concentration of the room with 1 Bq/m^3 for type 1 and 4 Bq/m^3 for type 2 resulting in a contribution to the effective dose of the order of 0.1 mSv per year.

In the calculation of the effective dose from the thoron exhalation the conversion factor between the equilibrium equivalent thoron concentration and the effective dose from ICRP publication 50 (2) is taken: 3.9 10^{-5} mSv/(Bq h/m^3). The effective dose from the thoron exhalation of the uncovered phosphogypsum is much greater than the one from the radon exhalation, namely, 1.8 mSv/year for type 1 and 0.9 mSv/year for type 2. Painting the gypsum reduces the thoron exhalation by a factor of 10 to 20 making the thoron dose comparable to that of radon

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AN ASSESSMENT METHOD OF DOSE EQUIVALENT DUE TO INDOOR ^{220}RN PROGENY BY USING ^{220}RN CONCENTRATION MEASURED AT A 20 CM DISTANCE FROM WALL

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INTRODUCTION

In indoor ^{222}Rn surveys in Japan, high ^{222}Rn concentrations were sometimes observed by passive methods(1) in Japanese traditional dwellings with soil walls. However, the ^{222}Rn progeny concentrations measured by active methods in the dwellings were not so high but thoron (^{220}Rn) progeny concentrations were rather high(2). Moreover, some studies(3,4) have made it clear that high ^{220}Rn concentrations were observed near the soil walls. Since there are the dwellings with soil walls in all over the world, it is important to evaluate the annual effective dose equivalent dose due to ^{220}Rn and its progeny(4).

PASSIVE ^{222}RN AND ^{220}RN CUP MONITORS

A pair of passive ^{222}Rn and ^{220}Rn cup monitors was developed. The cross sections are shown in **Fig. 1**. The ^{222}Rn and ^{220}Rn monitors have a 5 mm exchange opening and four exchange openings of diameter 20 mm, respectively. Calibration factors of the ^{222}Rn and ^{220}Rn monitors were $(2.15\pm0.19)\times10^{-5}$ and $(5.06\pm0.89)\times10^{-4}$ tracks$\cdot$cm^{-2}(Bq$\cdot$m$^{-3}\cdot$h)$^{-1}$ and agree well with the calculated values. The lower detection limit of the ^{222}Rn and ^{220}Rn monitors were 6.4 and 21 Bq$\cdot$m^{-3} for an exposure time of 90 d.

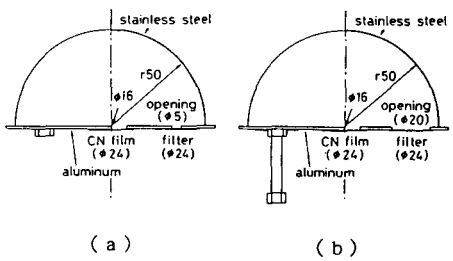

Fig. 1 Passive integrating ^{222}Rn and ^{220}Rn cup monitors (a) for ^{222}Rn measurement and (b) for ^{220}Rn measurement.

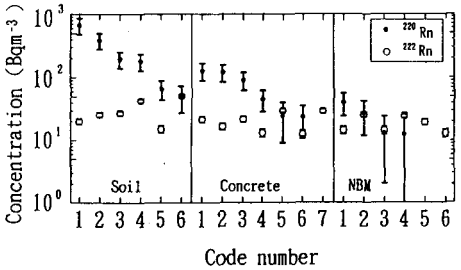

Fig. 2 Result of the survey from 29 Oct. 1991 to 30 Jan. 1992: (●) ^{220}Rn and (○) ^{222}Rn concentrations.

MEASUREMENTS OF INDOOR ^{222}RN AND ^{220}RN CONCENTRATIONS

Small scale surveys of ^{222}Rn and ^{220}Rn concentrations indoors were performed with the cup monitors over three years. An example of the surveys is shown in Fig. 2. Mean ^{220}Rn concentrations of the dwellings with soil walls were the highest compared with dwellings made of concrete and new building material (NBM).

In some dwellings with soil wall, the ^{220}Rn concentrations have been measured by using six pairs of cup monitors. An example of the results is shown in Fig. 3. The ^{220}Rn concentration could be expressed by the following equation(5):

$$Q(X) = \frac{E_{Tn}}{\sqrt{\lambda_{Tn}\cdot D}} \exp(-\sqrt{\lambda_{Tn}/D})X, \qquad (1)$$

where Q(X) is the ^{220}Rn concentration at a distance of X m from wall in Bq•m^{-3}, E_{Tn} is the ^{220}Rn exhalation rate from wall in Bq•m^{-2}•s^{-1}, λ_{Tn} is the ^{220}Rn decay rate in h^{-1}, and D is the effective diffusion coefficient of ^{220}Rn in air in m^2•s^{-1}. The E_{Tn} and D calculated by using the least squares method range from 1.6 to 6.4 Bq•m^{-2}•s^{-1} and 1.2 to 7.4X10^{-5} m^2•s^{-1}, respectively.

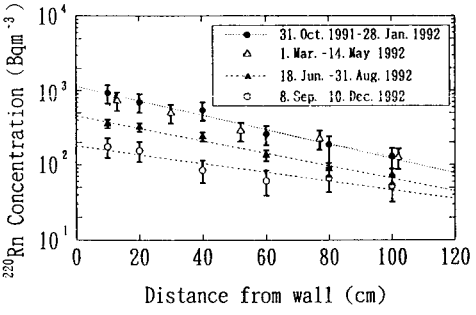

Fig. 3 Distributions of ^{220}Rn concentrations indoors.

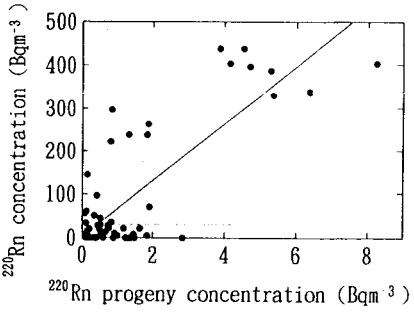

Fig. 4 Relationship between measured ^{220}Rn and ^{220}Rn progeny concentrations.

DOSE ASSESSMENT METHOD

The effective diffusion coefficient ranges from 1.2 to 7.4X10^{-5} m^2•s^{-1} as described above. If we measured ^{220}Rn concentrations at a distance of 20 cm from wall, Q(0.2), exhalation rate of ^{220}Rn from wall can be calculated to be 0.0099Q(0.2) in ranging from 0.0074Q(0.2) to 0.0124Q(0.2) from Eq. (1). Thus, ^{220}Rn exhalation rate could be estimated by uncertainty of only ±25% from the ^{220}Rn concentration at 20 cm from wall.

The ^{212}Pb (ThB) concentration in dwelling is expressed as follows(5):

$$C_B = \frac{\lambda_a \cdot \lambda_B \cdot N_B \cdot S/V + \lambda_v \cdot C_{Bo}}{\lambda_B + \lambda_v + \lambda_d}, \qquad (2)$$

where N_B is defined by

$$N_B = E_{Tn}[\lambda_{Tn}(\lambda_B + \lambda_a + \sqrt{\lambda_{Tn}(\lambda_B + \lambda_a)})], \qquad (3)$$

C_B is indoor ^{212}Pb concentration in Bq•m^{-3}, λ_a is the attached coefficient of free nuclei to aerosol particles in s^{-1}, λ_B is the decay rate of ^{212}Pb in s^{-1}, S is the surface area of wall in m^2 from which ^{220}Rn exhales, V is the volume of the room in m^3, λ_v is the ventilation rate of the room air in s^{-1}, C_{Bo} is outdoor ^{212}Pb concentration in Bq•m^{-3}, and λ_d is the deposition rate of attached ^{212}Pb atoms in s^{-1}. Here, we assume that the ratio of surface area exhaling ^{220}Rn to volume of the room is about 1/3 and does not vary on dwellings. The values of the another parameters are assumed to be; λ_a = 0.0139 s^{-1}(6), λ_v= 1.39X10^{-4} s^{-1}(7) and λ_d = 5.6X10^{-6} s^{-1}(6). Moreover, ^{220}Rn progeny concentration is equal to about 1.1 times ^{212}Pb, if ^{212}Pb and ^{212}Bi are in equilibrium. Consequently, ^{220}Rn progeny concentration could be expressed by ^{220}Rn exhalation rate as follows:

$$C_{TP} = 1.3E_{Tn}, \qquad (4)$$

where C_{TP} is ^{220}Rn progeny concentration in Bq•m^{-3}. As E_{Tn}=0.0099Q(0.2) described above,

$$C_{TP} = 0.013Q(0.2). \qquad (5)$$

Equation (5) indicates that indoor ^{220}Rn progeny concentration could be estimated from the ^{220}Rn concentration measured with passive cup monitors at 20 cm from wall.

In some dwellings, ^{220}Rn progeny concentrations were measured by active methods(2, 8) during the surveys of ^{220}Rn concentrations with cup monitors. Figure 4 shows the relationship between measured ^{220}Rn and ^{220}Rn progeny concentrations. The correlation coefficient between ^{220}Rn and ^{220}Rn progeny concentrations was 0.79. The relationship between ^{220}Rn progeny and ^{220}Rn concentrations is therefore given by

$$C_{TPm} = 0.015Q(0.2)_m. \quad (6)$$

where C_{TPm} and $Q(0.2)_m$ are the measured concentrations of ^{220}Rn progeny and ^{220}Rn. The relation of Eq. (6) gives fairly good agreement with the theoretical relation in Eq. (5).

Annual effective dose equivalents due to ^{222}Rn and ^{220}Rn progeny were estimated by using the measured values of small scale surveys. The results classified by building materials are shown in **Table 1**. The effective dose equivalent factors of ^{222}Rn and ^{220}Rn progeny are estimated to be 70 and 280 $\mu Sv \cdot y^{-1}$ per $Bq \cdot m^{-3}$ using a mean breathing rate of 0.8 $m^3 \cdot h^{-1}$ indoors and an average occupancy factor of 0.8 indoors(9). An equilibrium factor between ^{222}Rn and its progeny was assumed to be 0.4 indoors. On the other hand, ^{220}Rn progeny concentration were calculated by Eq. (5). The annual effective dose equivalents due to ^{220}Rn progeny in the dwellings with soil wall was calculated to be 0.67 $mSv \cdot y^{-1}$ and may be equal to that due to ^{222}Rn and its progeny.

Table 1 Estimation of annual effective dose equivalents due to ^{222}Rn and ^{220}Rn progeny.

	^{222}Rn		^{220}Rn	
Building materials	Concentration ($Bq \cdot m^{-3}$)	H_{eff} ($\mu Sv \cdot y^{-1}$)	Concentration ($Bq \cdot m^{-3}$)	H_{eff} ($\mu Sv \cdot y^{-1}$)
soil	19.1±0.9	530±30	159.7±12.4	670±50
concrete	23.3±1.5	650±40	41.9± 3.3	180±10
NBM	17.5±1.2	490±30	22.9± 3.0	100±10
UNSCEAR 1988(9)		1100		160

CONCLUSIONS

A pair of passive cup monitors with a different air exchange openings was developed for measuring simultaneously ^{222}Rn and ^{220}Rn concentrations. Indoor ^{220}Rn concentrations were very high in traditional Japanese dwellings with soil walls. The ^{220}Rn concentration decreases exponentially with the distance from wall. The effective diffusion coefficient of ^{220}Rn in dwelling and the exhalation rate of ^{220}Rn from wall were evaluated from the distribution of the ^{220}Rn concentrations. Then, indoor ^{220}Rn progeny concentration could be estimated from the ^{220}Rn concentration at a 20 cm distance from wall. From the results of the surveys, the average annual effective dose equivalent due to ^{220}Rn progeny was expected to be 0.67 $mSv \cdot y^{-1}$ in the traditional Japanese dwellings.

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COMPARED DEPOSITION OF RADON-222 AND RADON-220 (THORON) PROGENY AND NUCLEAR ABERRATIONS IN THE RESPIRATORY TRACT OF RATS AFTER EXPOSURE UNDER DIFFERENT AEROSOL CONDITIONS.

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INTRODUCTION

The inhalation of airborne short-lived radon decay products occured from the two main isotopic forms of radon, ^{222}Rn and ^{220}Rn (thoron), which are ubiquitous in the human environment and contributed to the greatest part of the natural radiation exposure. The laboratory rat has been particularly used to estimate the hazard to human health following exposures to radon and progeny (1). These data on lung cancer risk in rats are consistent with the epidemiological findings for uranium miners in all exposure categories considered and in typical aerosol characteristics, so that the exposure-dose conversion factor estimated by lung modelling are comparable in the two species (2). It has been shown that measurements of deposition of ^{214}Pb in excised lungs and nuclear aberrations in alveolar macrophages allow us to estimate the dose delivered to the deep lung after radon progeny inhalation (3). In contrast, experimental data for thoron and thoron progeny are quite limited. The aim of this study was to develop an experimental device suitable for the exposure of few rats to thoron, - to measure the deposition of thoron progeny within the excised lung of rats by gamma spectrometry of ^{212}Pb and to compare this deposition for a similar cumulative exposure with the incidence of micronuclei and binucleated alveolar macrophages observed after radon progeny inhalation in well characterized atmospheres.

EXPERIMENTAL METHODS

An available and convenient source of thoron gas is ^{228}Th, involatile at room temperature, which decays rapidly via thoron gas in ^{216}Po and ^{212}Pb. It is possible to increase dramatically the rate of emanation by heating. Such a small experimental design was performed for nose-only animal exposures (Algade-Cogéma, France). It consists in a small glass volume (15 liters) in which the heated generator was previously located and where the rats were restrained in specially designed tubes so that only the noses of the animals protruded into the chamber. Due to its very short half-live, ^{216}Po is in equilibrium with thoron so that the activities are comparable. A smoke aerosol was introduced to allow the attachment of ^{212}Pb and limit its plate-out on surfaces. Six rats were exposed in a single session to thoron and progeny for 3 hours under these experimental conditions.

Two animals were euthanazied to determine the gross deposition in the excised lungs by gamma spectrometry of the emission of ^{212}Pb at 239 keV. The retention of ^{212}Pb was measured within an annulus shaped NaI detector (Crismatec, France). All these measurements were performed according to the recommendations decided between the European laboratories involved in animal studies on radon (4).

Two of the remaining rats were euthanazied 6 days after thoron exposure and the two last ones, 12 days after the end of thoron exposure. Micronuclei and binucleated cells were scored in alveolar macrophages extracted by bronchoalveolar lavage, 8 days after exposure. Cytospins of cell suspension were air-dried, fixed in ethanol-ether (1/1 v) and stained with Giemsa before light microscopic observation.

RESULTS AND DISCUSSION

Table 1 shows the inhalation parameters measured in exposure chambers and lead deposition measured in excised lungs of thoron exposed rats compared to those previously measured in rats exposed to radon (^{222}Rn) in the inhalation facility of Razès under different aerosol atmospheres and using a quite similar potential alpha energy concentration (PAEC).

Table 1. Inhalation parameters of radon and thoron exposures and Pb retention in the deep lung of rats under different conditions and for a similar potential alpha energy concentration (PAEC).

RADON INHALATIONS			THORON INHALATION	
Aerosol parameters	High equilibrium	Low equilibrium	Aerosol parameters	High equilibrium
^{222}Rn activity	4632 kBq/m3	9925 kBq/m3	^{220}Rn activity	1268 kBq/m3
^{218}Po activity	2408 kBq/m3	7146 kBq/m3	^{216}Po activity	1100 kBq/m3
^{214}Pb activity	1713 kBq/m3	1488 kBq/m3	^{212}Pb activity	330 kBq/m3
^{214}Bi activity	1667 kBq/m3	397 kBq/m3	^{212}Bi activity	< 1 kBq/m3
Equilibrium factor	0.36	0.16	Equilibrium factor	0.30
PAEC	9.3 mJ/m3 450 WL	8.9 mJ/m3 429 WL	PAEC	15 mJ/m3 720 WL
Cumulative dose (after 3 hours)	27.9 mJ/m3.h 7.9 WLM	26.7 mJ/m3.h 7.5 WLM	Cumulative dose (after 3 hours)	43 mJ/m3.h 12 WLM
^{214}Pb activity in 3 excised lungs	5.0 ± 2.9 kBq/g	5.3 ± 3.2 kBq/g	^{212}Pb activity in 2 excised lungs	0.36± 0.2 kBq/g

Table 2 shows the percentage of micronucleated and binucleated alveolar macrophages recovered from the rats exposed to radon and euthanazied 8 days later. The controls corresponded to 5 unexposed rats killed at the same age.

Table 2. Percentage of micronucleated and binucleated alveolar macrophages (AM) one week after radon and thoron exposures (mean value ± SD).

	% of micronucleated AM	% of multinucleated AM
Controls	0.19 ± 0.10	1.34 ± 0.69
radon at high equilibrium	1.22 ± 0.26	2.91 ± 0.74
radon at low equilibrium	1.47 ± 0.42	2.30 ± 0.78
thoron at high equilibrium	0.59 ± 0.06	2.64 ± 0.98

If the micronuclei index is similar for radon inhalations under different aerosol characteristics, the number of micronuclei after thoron is lower as regards the level of concentration but higher than that found spontaneously in unexposed rats. However it is now possible to expose rats to an aerosol mixture of thoron and progeny inducing an increased incidence of alveolar cell nuclei aberrations. Further studies are still in progress to improve the measurements of the aerosol characteristics, thoron progeny deposition and dose distribution compared with radon progeny exposures.

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THORON PROGENY CONCENTRATIONS IN LIVING SPACE

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INTRODUCTION

In the UNSCEAR 1988 report(1), it was estimated that the inhalation of short-lived decay products of radon(^{222}Rn) and thoron(^{220}Rn) accounts on average for about one half of the effective dose equivalent from all natural sources of radiation. In recent years, several studies on radon and its progeny in mines and in dwellings have been performed, but for indoor thoron progeny there are only a few measurements. In Europe, the mean thoron progeny levels represented by equilibrium equivalent thoron concentration in dwellings seem to lie between 0.1 and 1.0 Bqm^{-3}(2), and the thoron progeny makes a contribution of about 10 to 30% of the effective lung dose equivalent from all indoor radon plus thoron and their progeny. However, Doi(3) reported that thoron concentration in Japanese dwellings may be is higher than in other countries because the interior walls of typical Japanese dwellings are covered with a soil based plaster. Therefore, the dose due to indoor thoron progeny may be higher in Japan.

Thoron concentration depends on a distance from wall which is a source of thoron, but less information is available concerning the distribution of indoor thoron progeny concentrations. It is necessary to study the characteristics of thoron progeny for estimation of the absorbed dose. The concentration distributions of indoor thoron progeny have not been studied and calculated in detail. Thereby more research of thoron progeny should be required.

This report shows (i)the theoretical distributions of indoor thoron progeny concentration that could be predicted by a model calculation using the numerical method(4) and (ii)the development of a quiet portable monitor for measuring both thoron and radon progeny(5).

CONCENTRATION DISTRIBUTION

The behavior of thoron and its progeny almost depend on the diffusion in the room because of a very short half-life time of thoron which emanates from the wall. For describing movement of the nuclides in the dwelling, we introduce an effective diffusion coefficient D[m^{2}s^{-1}] which expresses the diffusion in the room macroscopically, then we suppose the effective diffusion coefficients of thoron and its progeny have the same value. Distinguishing between attached (superscript 'a') and unattached or free (superscript 'f') atoms, the formation and the extinction of thoron decay chain is considered as follows:

$$\begin{array}{ccccccccc} & & \lambda_T & & \lambda_A & & \lambda_a & & \lambda_B & & \lambda_C \\ ^{220}\mathrm{Rn} & \rightarrow & ^{216}\mathrm{Po}^f & \rightarrow & ^{212}\mathrm{Pb}^f & \rightarrow & ^{212}\mathrm{Pb}^a & \rightarrow & ^{212}\mathrm{Bi} & \rightarrow \\ & & \downarrow & & \downarrow & & \downarrow & & \downarrow & \\ & & \lambda_d & & \lambda_d & & \lambda_d+\lambda_v & & \lambda_d+\lambda_v & \end{array}$$

Thoron and its progeny concentrations were calculated by two dimension and steady state model. For considering the distribution of thoron and its progeny, the plane was divided into K^2 cells. Considering the diffusion, deposition and ventilation at each cell, 5 K^2 sets of the simultaneous equations are obtained. Since all differential equations are equal to zero under the steady state conditions. These equations could be solved directly by the method of LU-factorization. Substituting the appropriate parameters, distributions of thoron and its progeny concentrations are calculated .

MEASUREMENT

The measurement method for radon and thoron progeny is based on the solid state nuclear track detector. A schematic diagram of the potential alpha-energy monitor head is given in Figure 1. The measurement system is composed of the monitor head, a diaphragm pump, an integrating flow meter and a timer. The pump is selected to be as silent as possible for long term measurements in the living space. The total weight of the system is about 2kg. The system is portable, and can be easily moved from dwelling to dwelling. The monitor head has four windows, two for radon progeny and two for thoron progeny. The thickness of absorbers are different for radon or thoron progeny because of the difference of the emitted alpha particle energies.

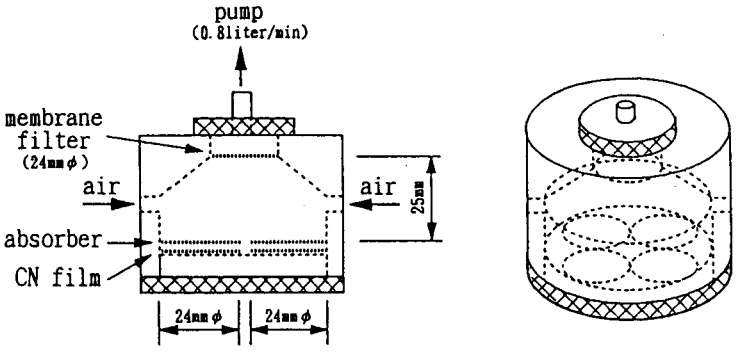

Figure 1. Schematic drawing of a potential alpha-energy monitor for radon and thoron progeny.

In equations(1) and (2), the equilibrium equivalent radon concentration(E.E.R.) and the equilibrium equivalent thoron concentration(E.E.T.) could be obtained by the collecting volume(V) and etch pits on the CN film(N_R,N_T), where N_R and N_T are the count of etched tracks in radon and thoron window, respectively. The unit of V is m^3.

$$E.E.R. = \frac{6.79\ 10^{-2}\ N_R}{V} \qquad (1)$$

$$E.E.T. = \frac{7.59\ 10^{-3}\ N_T}{V} \qquad (2)$$

RESULTS AND DISCUSSION

Figure 2 shows the distributions of thoron, ^{212}Pb(unattached) and ^{212}Bi concentration in dwellings calculated by this method. Thoron emanates only from the whole left side wall and diffuses to the right. The effective diffusion coefficient ,D=0.005 m^2s^{-1}, in Figure 2 is normal value in the ordinary dwellings(6). Thoron concentration is decreasing exponentially with the distance from the wall of thoron emanation. Both the atoms of ^{212}Pb and that of ^{212}Bi, which are closely related to human exposure, are the constant concentrations whether at the center or the corner in the room. Therefore, it is easy to estimate the absorbed dose due to inhalation of thoron progeny which has uniform concentration in the room.

In the model dwelling, the equilibrium equivalent thoron concentration calculated by plausible parameters is compared to the experimental value. The calculated concentration was 3Bqm^{-3}. It is little higher than experimental value of 1.1 Bqm^{-3}. If the ventilation rate equals to 2.0h^{-1} which is four times higher than the above calculation, the calculated concentration becomes to be 1.1 Bqm^{-3}.

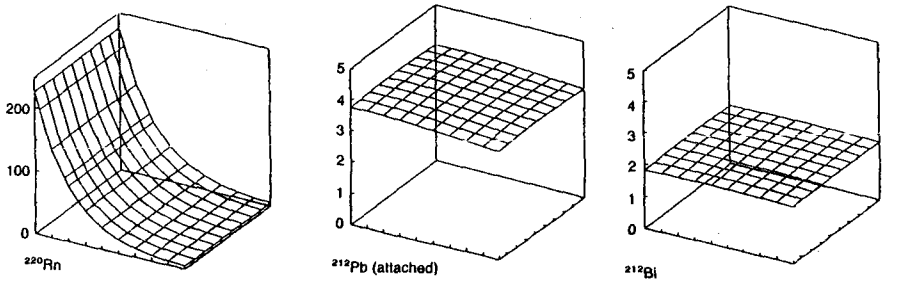

Figure 2. The general views of concentration distributions. The vertical axes show the activity concentrations.

Figure 3 shows measurement results at four locations as a example. The measurement period is one week at each location. Figure 4 shows the RT-ratio, which is defined by the ratio of E.E.T. to E.E.R. The indoor RT-ratio is about 0.1, which is about three times higher than outdoor. The indoor thoron progeny may make a significant contribution to the public dose equivalent of Japanese.

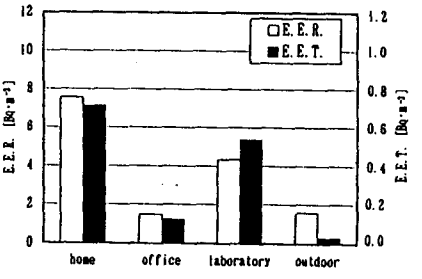

Figure 3. Results of measured E.E.T. and E.E.R.

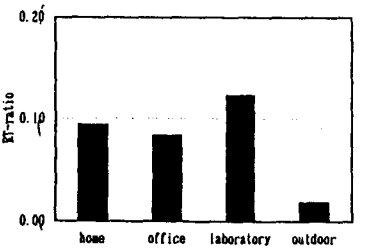

Figure 4. RT-ratio(E.E.T./E.E.R.).

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BODY CONTENT OF ^{210}Pb: A MEASURE OF CUMULATIVE EXPOSURE TO RADON ?

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INTRODUCTION

The content of the long-lived ^{222}Rn daughter ^{210}Pb in the human body has been proposed as a measure of cumulative exposure to radon/radon progeny for occupational and residential environments. However, there are various sources that contribute to the ^{210}Pb accumulation in the body. For a reliable interpretation of any measured ^{210}Pb value, it is imperative that the significance and variability of these different sources are sufficiently well understood. In this study, the accumulation of ^{210}Pbe in the human body was modelled for a variety of exposure conditions, applying the most recent knowledge on lead metabolism in man.

SOURCES OF ^{210}Pb IN THE BODY

Age dependent values for the uptake of ^{210}Pb from the various natural sources are summarized in Table 1. Under normal residential conditions, consumption of food is the most important route by which ^{210}Pb enters the human body. Whereas the short-lived ^{222}Rn daughters account for the majority of the lung dose, they contribute relatively little to the ^{210}Pb accumulation in the body. For normal radon levels, the blood uptake of ^{210}Pb by direct inhalation of air borne ^{210}Pb is a factor of five higher than uptake from the short-lived ^{222}Rn daughters (Table 1). Cigarette smoking may significantly contribute to the deposition of ^{210}Pb. Additional small components of ^{210}Pb uptake result from the presence of ^{222}Rn and ^{226}Ra in the human body. The contributions to ^{210}Pb accumulation in the body can be clearly grouped into two categories: components that are independent of the level of ^{222}Rn and components that are related to environmental ^{222}Rn (Table 1).

CALCULATIONS FOR DIFFERENT EXPOSURE MODELS

The accumulation of ^{210}Pb in various organs and tissues of the body and the excretion pattern were predicted on the basis of the age-dependent uptake values given in Table 1. The distribution and retention after systemic uptake was described applying the age-dependent metabolic model for lead as described in ICRP Publication 67 [1]. The model calculations were performed employing the computer program SAAM-31 [2].

RESULTS

In Fig. 1, the ^{210}Pb activity in whole body, total skeleton, soft tissues and blood during life-time are shown for "normal" exposure conditions. The effect of regular smoking on the accumulation of ^{210}Pb under otherwise "normal" conditions is shown in Fig. 2. The effects of variations in the ^{222}Rn level and hence in the ^{222}Rn dependent components of ^{210}Pb uptake are shown in Fig. 3. For higher radon level environments these components dominate by far the ^{210}Pb accumulation. Results for the simulation of elevated occupational exposures during a certain time period are shown in Fig.4. It shows the time course of ^{210}Pb activity for the two bone compartments (trabecular and cortical) and the whole skeleton. The subsequent decrease after returning to "normal" exposures is very different for the two types of bone. Whereas the half time for trabecular bone is about 4.4 years, it is about 19 years for cortical bone.

Table 1. Average daily uptake to blood of ^{210}Pb from natural sources (mBq · d^{-1})

Age group	Ingestion [a]	Direct inhalation of airborne ^{210}Pb [b,c]	Production of ^{210}Pb from inhaled ^{222}Rn daughters [b,d]	Production of ^{210}Pb from ^{222}Rn dissolved in body fluids [e]	Production of ^{210}Pb from ^{226}Ra in body [f]
3 mo	12	0.5	0.1	0.01	< 0.01
1 y	16	0.9	0.2	0.02	< 0.01
5 y	24	1.4	0.2	0.03	0.01
10 y	28	1.9	0.3	0.04	0.01
15 y	25	2.4	0.4	0.05	0.02
adults	20	2.7	0.5	0.07	0.03
(smoking) [g]		(10)			

[a] Derived from reference profiles for food consumption and activity concentrations in diets and water [3]. Absorbed fractions according to ICRP Publication 67 [1]

[b] Assuming age dependent reference breathing rates [3]. Applying new lung model of ICRP [4].

[c] Assuming air concentration of 0.5 mBq · m^{-3} of ^{210}Pb [Unpublished observations]

[d] Derived for a typical indoor atmosphere of 40 Bq m^{-3} of ^{222}Rn, 33 Bq m^{-3} of ^{218}Po, 22 Bq m^{-3} of ^{214}Pb, and 15 Bq m^{-3} of ^{214}Bi [3]; ^{210}Pb equivalent ($8.46 \cdot 10^{-5}$ Bq ^{210}Pb per m^3)

[e] Derived for indoor atmosphere with 40 Bq m^{-3} of ^{222}Rn. (0.085 mBq·d^{-1} of ^{210}Pb per Bq ^{222}Rn in body)

[f] Assuming 0.85 Bq of ^{226}Ra per kg of calcium [5]. Assuming 30% of the produced ^{222}Rn and its daughters are retained long enough to decay to ^{210}Pb

[g] Estimate for 20 cigarettes per day (20 mBq ^{210}Pb per cigarette; 10% inhalable) [5]. Applying new lung model of ICRP [4].

DISCUSSION

The uptake of ^{210}Pb via the lung through inhalation of ^{210}Pb and the short-lived ^{222}Rn daughters depends not only on their concentrations in air but also on the physical properties of the inhaled particles on which the Rn daughters and ^{210}Pb collect. Whereas extensive measurements of ^{222}Rn concentrations have been made in residential and occupational settings, little information is available on indoor concentrations of ^{210}Pb. Preliminary results indicate that changes in indoor ^{222}Rn concentrations are accompanied by corresponding changes in airborne ^{210}Pb.

The calculated ^{210}Pb contents are in good agreement with available autopsy data [6]. From these data, a total skeleton content of 15 Bq of ^{210}Pb can be derived for normal exposures, which is in very close agreement with the results of the present calculations.

Estimates of body ^{210}Pb are finally aimed to estimate cumulative exposures to the short-lived ^{222}Rn daughters, since it is the lung dose from these radionuclides which determines the health risk from radon. However, the short lived Rn daughters contribute relatively little to the ^{210}Pb accumulation in the body (2% under normal conditions vs. 12% from the direct inhalation of ^{210}Pb and 86% from ingestion). The relative importance of the short lived Rn daughter component for the ^{210}Pb accumulation at increasing Rn levels depends strongly on the relationship between the concentrations of short lived Rn daughters and ^{210}Pb in air at increasing indoor Rn levels, which until now has not yet been established. It must be concluded therefore that all components contributing to body ^{210}Pb must be known with sufficient accuracy and for a large range of Rn levels before a relationship between body ^{210}Pb and lung dose can be postulated. Without a sound knowledge of these components and their variabilities, no reliable estimates for the lung dose can be obtained, even when the body content of ^{210}Pb can be assessed very accurately. The smaller the contribution of Rn daughters to the ^{210}Pb accumulation, the smaller will be the lung dose per unit of exposure and the larger will be the inherent uncertainties.

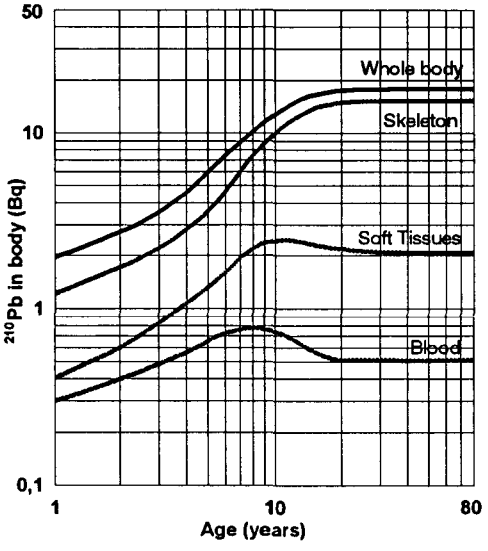

Figure 1. ^{210}Pb accumulation in whole body, total skeleton, soft tissues and blood during life time for normal exposure conditions (Table 1).

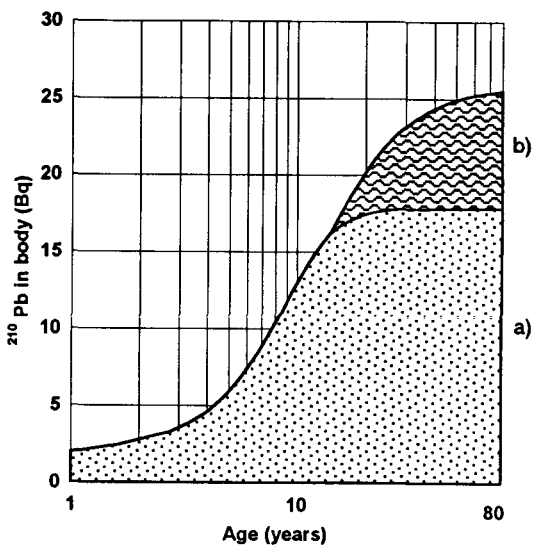

Figure 2. Effect of regular smoking on ^{210}Pb accumulation: a) no smoking; b) 1 pack of cigarettes per day (10 mBq ^{210}Pb / day).

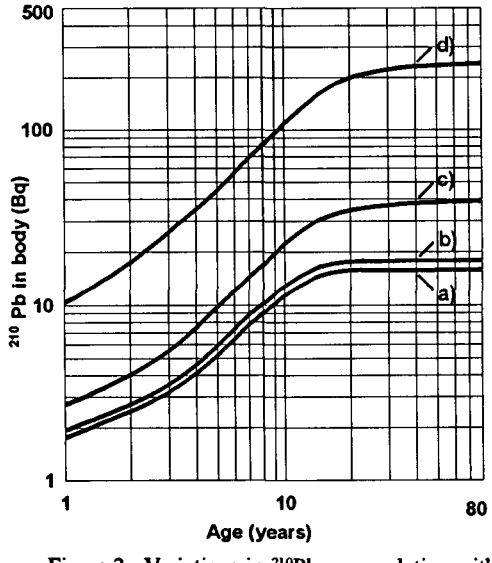

Figure 3. Variations in ^{210}Pb accumulation with radon level: a) no radon; b) "normal" radon level; c) tenfold increase in radon; d) hundredfold increase in radon.

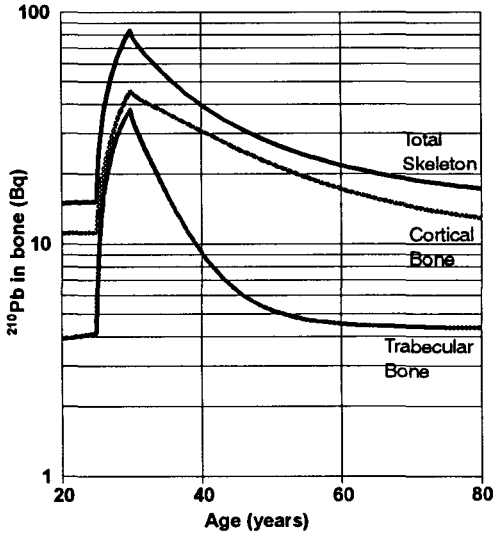

Figure 4. ^{210}Pb activity in bone, assuming a hundrefold increase in radon for a period of 5 years (from 25 years to 30 years), with normal exposures before and thereafter.

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RADIATION EXPOSURE DUE TO RADON IN DRINKING WATER IN REGIONS WITH HIGH TERRESTRIC ACTIVITY

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INTRODUCTION

We performed investigations to evaluate the radiation exposure due to the private use of drinking water in the region of the German Fichtelgebirge (300 km North Munich near the Czech border), where the content of Radon in ground water is very high. Besides the drinking water pathway the different practices in the household were investigated such as taking showers, stay in a hairdressing salon, operating a washing machine.

RADON CONTENT IN WATER

In water Radon concentrations from less than 0,1 mBq/l up to more than 5 kBq/l are to be found. In Germany the mean concentration of tap-water is 7,2 Bq Rn 222 per litre [3].

In table 1 there are listed the results of our measurements in the German Fichtelgebirge (300 km North Munich near the Czech border).

Location		C(Rn-222) [Bq/l]
Swimming pool Hof	Upstream charcoal filter	250
	Downstream charcoal filter	6
	Pool, waterfall	6
	Non swimming pool	8
	Whirl pool	3
	Sauna, pool	6
Shower, Weißenstadt*	Run in, bath	685
Inn, Ochsenkopf, private well	Run in (mainstream), cellar	836
Hotel, Fahrenbühl	Run in, bath	320
	Run in, kitchen	301
	Run in, swimming pool	283
Farm house, Fahrenbühl	Run in, bath	281
	Well, backyard	114
Hairdresser, Kirchenlamitz	Run in, washbasin	397
Inn, Girgelstein, private well	Run in, washing machine	920
	Run in, dish-washer	798
	Run in, bath	710
	Run in (mainstream), cellar	898

*- Drinking water treatment facility of Municipal Water Supply, City of Hof

Table 1: Rn-222 concentration in drinking water, emanation free taken

RADIATION EXPOSURE DUE TO CONSUMPTION OF DRINKING WATER

Following [2] we assumed equilibrium between Rn 222 and his short lived daughter nuclides Po 218, Pb 214, Bi 214 and Po 214. For Rn 222 concentrations above 100 Bq/l we assumed Pb 210, Bi 210 and Po 210 concentrations of 5 mBq/l due to Radon. These nuclides are negligible for Rn 222 concentrations below 100 Bq/l. These assumptions arise from the results given in [2].

There are 3 pathways to distinguish:

- Intake of freshly taken drinking water (concentration of Rn 222, short and long lived daughter nuclides as measured, see above)
- Consumption of drinks made of heated up drinking water such as tea, coffee (Rn 222 completely off, short and long lived daughter nuclides as above)
- Use of drinking water for cooking of liquid foodstuff (duration $\geq$ 1 hour; Rn 222 completely off, short lived daughter nuclides 25% of freshly taken water, long lived daughter nuclides as above)

Based on the highest measured Radon concentration of about 1000 Bq/l for freshly taken water, assuming an average consumption rate of 400 l drinking water per year [1] and the distribution of the total consumption as 1 : 3 : 2 (70 l : 200 l : 130 l per year) into the above mentioned three pathways an effective dose of 1,5 mSv per year has been estimated. For the German average Rn 222 concentration in freshly taken water of about 5 Bq/l the effective dose yields to 7 μSv per year. In both cases the main part (about 95%) of the estimated effective dose results from the intake of freshly taken drinking water.

RADIATION EXPOSURE DUE TO TAKING SHOWERS

Figure 1 shows the measured concentration of Rn 222 and his daughter nuclides during taking showers for a measured Rn 222 concentration in water of 685 Bq/l. The maximum values of Rn 222 concentration in air and EER respectively are 21.350 Bq/m³ and 5.860 Bq/m³. A 1 hour shower taking of a family was simulated. After taking the shower a 20 minute stay in the bathroom was assumed for hair drying, e. g. As expected the air concentration is lower leaving the window ajar (approximately by a factor of 3). Assuming a stay of 20 minutes per day at the most unfavourable time the integration of EER given in figure 1 yields to an exposure of 1 WLM and hence an effective dose of 4 mSv per year using the dose conversion convention 4 mSv per WLM according to ICRP 65.

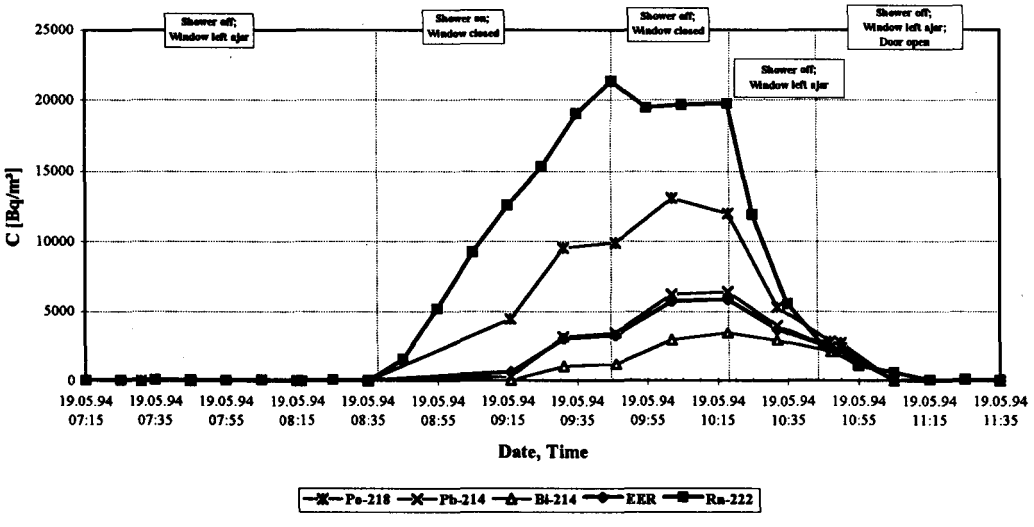

Figure 1 ***Rn-222- and Rn-222 Daughter Nuclides Concentration in Air while Taking Showers (Volume 11,25 m³; Rn 222 concentration in Water 685 Bq/l; 9 °C; 15 - 18 l/min)***

RADIATION EXPOSURE DUE TO OPERATING A WASHING MACHINE AND FOR STAY IN A HAIRDRESSING SALON

Concerning the radiation exposure due to the Radon concentration in water for a stay in a hairdressing salon and for operating a washing machine the same investigations have been conducted.

The washing machine has been running with a Rn 222 concentration in water of 280 Bq/l. The maximum values of Rn 222 concentration in air and EER respectively (volume 31,2 m^3) are 925 Bq/m^3 and 85 Bq/m^3. Assuming a 20 minutes stay in the room per week after finishing the machine operation the integration of EER yields to an exposure of 0,002 WLM and hence an effective dose of 8 µSv per year using the dose conversion convention 4 mSv per WLM according to ICRP 65.

The water used in the hairdressing salon had a Rn 222 concentration of 400 Bq/l. The maximum values of Rn 222 concentration in air and EER respectively were 285 Bq/m^3 and 120 Bq/m^3. Assuming a 2 hours stay in the hairdressing salon per month the integration of EER yields to an exposure of 0,002 WLM and hence an effective dose of 8 µSv per year using the dose conversion convention 4 mSv per WLM according to ICRP 65.

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AN ATTEMPT TO EVALUATE DOSE EQUIVALENT DUE TO THE INHALATION OF ^{222}Rn AND ITS PROGENY IN INDOOR AIR

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INTRODUCTION

An attempt to evaluate alpha dose equivalent given by inhaled ^{222}Rn and its daughter nuclides has been made based on the measured concentrations of ^{222}Rn and its progeny in indoor air and the estimated equilibrium factor between the parent and daughter nuclides. Three different dosimetric lung models were used for the evaluation of the dose equivalent, and a brief comparison was made.

MEASUREMENT

1) ^{222}Rn Concentration

An activated charcoal canister of 10 cm diameter × 2.85 cm height, which contains 70 ± 1 g of 6 × 16 mesh charcoal powder, was used for collecting ^{222}Rn in indoor air by adsorption process for a period of certain time interval. Concentration of ^{222}Rn in the air was determined by gamma-ray spectrometry using a 3″ ϕ ×3″ cylindrical NaI(Tl) detector that detects 242, 295 and 352 keV photons emitted from ^{214}Pb , and 609 keV from ^{214}Bi, both of which are the equilibrated daughter nuclides of ^{222}Rn in the canister. Calibration factors(CF) for the canister were determined with the aid of a reference canister that contains a uniformly distributed known amount of equilibrated ^{226}Ra. The humidity correction factor for the CF were deduced through a series of appraisal test for the water adsorption characteristics of the charcoal canister. CF for ^{222}Rn concentration in the air was obtained by the calibration of the canister exposed in a Rn-Chamber containing ^{222}Rn of known concentration.

The measurements of the ^{222}Rn concentration in indoor air were carried out in a laboratory room of the C. N. University where average relative humidity was maintained to be 32±2 % during the course of the measurement. Five measurements for 24 hrs and 72 hrs periods, respectively, were performed, and in each measurement 3 canisters were exposed to the indoor air. Resultant mean concentration of ^{222}Rn in the room was found to be (40.0 ± 1.9) $Bq \cdot m^{-3}$.

2) ^{222}Rn Daughter Concentration and Equilibrium Factor

In order to determine the equilibrium factor, F, between the concentrations of ^{222}Rn and its progeny in the indoor air, a series of α-spectrometry was performed for the air particulate collected on a membrane filter of 47 mm diameter with 0.22 μm pore by use of an ion implanted Si detector of effective diameter of 24 mm. As the result of the spectrometry the concentrations of α-emitting daughters, ^{218}Po and ^{214}Po, were obtained. And the F is determined by[1)]

$$F = \sum_i a_i F_i \qquad (1)$$

where $a_i = k \cdot (\frac{E_{p,i}}{\lambda_i})$, $F_i = (\frac{C_i}{C_{Rn}})$, and $E_{p,i}$ and λ_i are potential α-energy concentration and decay constant of ith daughter nuclide. C_i and C_{Rn} are the concentrations of ith daughter and that of ^{222}Rn in the air, respectively. k is a constant that depends on the used system of units.

Thus determined overall mean F in the room where the measurements were performed was 0.36 ± 0.12. Using this value of F the equilibrium equivalent concentration(EEC) of ^{222}Rn in the indoor air was determined to be (14.4 ± 1.9) $Bq \cdot m^{-3}$.

ASSESSMENT OF THE DOSE EQUIVALENT

With the value of the F or EEC refered above the potential α-energy concentration, E_p, is estimated by[2)]

$$E_p = \frac{F \cdot C_{Rn}}{1.78 \times 10^8} \quad (J \cdot m^{-3}) \qquad (2)$$

to be $(80.8 \pm 10.4) \times 10^{-9}$ J · m^{-3},

By use of the presumed mean breathing rate of 1.2 m^3 · h^{-1} and 0.75 m^3 · h^{-1} for occupational and public exposures, respectively, the potential α-energy intakes, I_p, can be estimated to be $(97.0 \pm 12.5) \times 10^{-9}$ and $(97.0 \pm 12.5) \times 10^{-9}$ m^3 · h^{-1}, respectively, using the equation[2)]

$$I_p = B \cdot E_p \qquad (3)$$

where B is the mean breathing rate in m^3 · h^{-1}.

It was attempted to evaluate the regional and total lung dose equivalent rate given by the inhalation of ^{222}Rn daughters in the indoor air based on the data mentioned above and by adopting three different dosimetric lung models, namely, Jacobi-Eisfeld(J-E), James-Birchall(J-B), and ICRP models, which appeared in the Table 1 of the reference 2 and the Table 2.9 of the reference 3. The dose equivalent rates were evaluated for two target tissues, the basal cell layer in the tracheo-bronchial(T-B) region and the epithelium in the pulmonary(p) region, in addition to the total lung dose. In this estimation the unattached fraction, f_i , of the ^{222}Rn daughters, which is given by

$$f_i = \frac{E^u_p}{F \cdot C_{Rn}} \quad \text{or} \quad E^u_p = F \cdot C_{Rn} \cdot f_i \qquad (4)$$

where E_p^u is the potential α-energy concentration of unattached ^{222}Rn daughter nuclide in J · m^{-3}, was used. The results obtained by use of f_i values of 0 and 0.05 are summarized in Table 1. With an assumption of an occupancy factor of 0.8, the annual effectived dose equivalent due to the inhalation of ^{222}Rn and its progeny is also assessed and summarized in Table 2.

COMPARISON OF THE MODELS

A brief comparison in regard to the dose equivalent rate assessed by the ICRP model in the case of occupational exposure reveals that the T-D doses by the J-E and J-B models, in general, shown to be larger, except for f_i = 0, while P dose and total lung dose by both models appeared to be much smaller. Those evaluated by the J-E and J-B models were 0.65 ~ 0.70 and 0.25 ~ 0.26 times that obtained by the ICRP model, respectively.

The annual dose equivalents given by the ^{222}Rn daughters and the total lung dose estimated by the J-E model nearly coincide with(for f_i = 0) or slightly larger (for f_i = 0.05) than those given by the ICRP model, while those estimated by the J-B model for f_i = 0 are 0.7 times those by the ICRP model and 1.6 times the ICRP for f_i = 0.05. As a summary, it may be said that the J-E model looks closer to that of the ICRP than the J-B model for the evaluation of radon dose in the occupational exposure.

CONCLUDING REMARK

Through this study it was reconfirmed that the dose due to the inhalation of ^{222}Rn and its progeny in the air comes mainly from the daughter products with negligible contribution of ^{222}Rn itself. The respiratory organ dose given by the daughter products appeared to be region dependent, and T-D dose increases with unattached fraction of the progeny, while P dose decreases with the fraction.

Table 1. Regional and total lung dose equivalent due to the inhalation of ^{222}Rn daughters based on three different dosimetric lung models.

Region	Breathing rate($m^3 \cdot h^{-1}$)	Unattached fraction(f_i)	Mean dose equivalent rate ($nSv \cdot h^{-1}$)		
			J-E Model*	J-B Model*	ICRP Model
T-B	1.2	0	1746.0 ± 225.0	1358.0 ± 175.0	1455.0 ± 187.5
		0.05	2570.5 ± 331.3	4074.0 ± 525.0	1940.0 ± 250.0
	0.75	0	969.6 ± 124.8	969.6 ± 124.8	not evaluated
		0.05	1272.6 ± 163.8	1515.0 ± 195.0	
P	1.2	0	504.4 ± 65.0	194.0 ± 25.0	776.0 ± 100.0
		0.05	479.2 ± 61.8		737.2 ± 95.0
	0.75	0	363.6 ± 46.8	121.2 ± 15.6	not evaluated
		0.05	345.4 ± 44.5	115.1 ± 14.8	
Total (m=1kg)		0	679.0 ± 87.5	not evaluated	970.0 ± 125.0
		0.05	693.6 ± 89.4		989.4 ± 127.5

* J-E Model : Jacobi-Eisfeld Model J-B Model : James-Birchall Model

Table 2. The annual effective dose equivalent(μSv) due to the inhalation of ^{222}Rn and its daughters based on three different dosimetric lung models

Nuclide*	Breathing rate ($m^3 \cdot h^{-1}$)	Unattached fraction(f_i)	Annual effective dose equivalent(μSv)**		
			J-E model	J-B model	ICRP model
RnD	1.2	0	946.2 ± 98.5	652.6 ± 74.3	938.1 ± 89.4
		0.05	1282.3 ± 141.7	1794.6 ± 221.0	1125.7 ± 112.5
	0.75	0	560.6 ± 56.0	458.7 ± 52.9	not evaluated
		0.05	680.3 ± 71.4	685.4 ± 82.2	
Rn			50.5 ± 21.0		
Total	1.2	0	996.7 ± 100.7	703.1 ± 77.2	988.6 ± 91.8
		0.05	1332.8 ± 143.2	1845.1 ± 222.0	1176.2 ± 114.4
	0.75	0	611.1 ± 59.8	509.2 ± 56.9	not evaluated
		0.05	730.8 ± 74.4	735.9 ± 84.8	

* RnD : ^{222}Rn daughters
Rn : ^{222}Rn

** Effective dose equivalent

$H_E = W_{TB}H_{TB} + W_PH_P$

where W_{TB}, W_P : weighting factor of T-B and P region; 0.06, 0.06
H_{TB}, H_P : Effective dose of T-B and P region

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THE RADON SURVEY AND ITS CONTRIBUTION TO THE RADIATION EXPOSURE

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INTRODUCTION

Nowadays, the change of life conditions, the intense increase and diversified pollution of the environmental factors have a complex effect over the health status of the population. A compound of pollutants produces a stronger negative effect than the virtual sum of its components taken separately, determining their investigation as a whole. Radioactive pollution and exposure to natural and artificial radiations play a distinct role among the contemporary health hazards. It is well known that the most important contribution to human exposure to the ionizing radiations of natural origin is due to ^{222}Rn and its short-term daughters (1). In the 80's, this nobel gas was discovered to be accumulated mostly in the indoor air of the buildings. The accumulation of these radionucludes in dwellings depends on some factors such as: geochemical and geophysical properties of the sol where is the location of the construction, the building materials, the nature of the building foundation, the type of dwellings, the dimensions of rooms, the rate of ventilation, the radon and radium content of the tap water, type of heating used and the living and working conditions (2).

In this study we present the level of radon concentration in different types of buildings. Based on our results we estimated the yearly effective dose equivalent for these groups of the population.

MATERIAL AND METHOD

Two types of dwellings were taken under study: detached houses (built in 1920) and a ten floor block-of-flats, (built in 1970), in Cluj-Napoca situated in an area with "normal" radiation background. These buildings are characteristic for Transilvania, Romania. Cluj has 350000 inhabitants and according to the statistical data from 1992 the proportion of the population living in detached houses is 36 %.

The walls of the detached houses are made red brick while stone and slog were used for the basement. There is a cellar under the kitchen of the house I. In 1975, a new room (C) was attached to the old building (room A and B), having brick-made walls, while the basement, the floor and the ceiling were made of concrete. The rooms were the same volume 5m x 5m x 3.5m. Natural gas is used for cooking and heating. The block of flats is built out of prefabricated elements and all the flats have central heating. The living rooms volume are 4m x 4m x 2.5m.

The determinations of the ^{222}Rn content were performed under normal living conditions, in the autumn of 1994 using the "PRASSI - Portable Radon Survey Meter" instrument. The air samples were collected in the middle of the premises, at 0.5 m from the floor. In order to assess and compare the values of the indoor ^{222}Rn concentrations, measurements were performed in the outside as well, at 0.5 m from the ground, during the same period of time.

We estimated the Effective Dose Equivalent (EDE) of the people considering the internal dose absorbed in respiratory tract from radon and its daughters. The following assumptions has been made: the mean radon concentrations -determined in autumn- represent a yearly average, the equilibrium factor between ^{222}Rn and its daughters is 0.5, people spend 19 hours/day in dwellings, the mean inhalation rate of people is considered to be 14 l/min (3,4). The annual number of the expected lung cancer cases induced by radon daughters exposure can be calculated based on our results and the absolute lung cancer risk (5).

RESULTS AND DISCUSSION

The mean values of ^{222}Rn concentrations in the inside and outside of the buildings are shown in table 1. The indoor values are significantly higher in the rooms A,B (house I) and A' (house II) as compared to those found in the other rooms in detached house or to the block of flats.

The building materials being identical for the rooms, the bathroom and the kitchen, the high differences between

^{222}Rn concentrations could be explained by the dissimilarities in their respective basements.
Beneath the rooms A,B and A', the high porosity pavement made of stone and slag allows this noble gas to easily spread and reach the interior of the rooms. Moreover, the room B has only one wall in direct contact with the outside air so that the speed of the indoor natural air exchange is lower than in room A, that has two free walls

Table 1 ^{222}Rn CONCENTRATION IN FAMILY HOUSES AND BLOCK OF FLATS

PLACES	NUMBER OF SAMPLING	NUMBER OF MEASUREMENTS	Rn-222 CONCENTRATION (Bq/m^3)		
			MEAN ± SD	MIN.	MAX.
DETACHED HOUSE					
1.Room A	1	351	335 ± 76	22	520
Room B	2	151	436 ± 109	42	585
Room C	5	62	91 ± 20	27	124
Kitchen	4	90	45 ± 11	14	85
Bathroom	3	55	85 ± 25	47	162
2. Room A'	-	78	799 ± 293	310	1316
BLOCK OF FLATS					
Ground floor	-	4	70 ± 25	52	88
First floor	-	72	61 ± 14	19	90
Four floor	-	8	65 ± 25	50	81
Seven floor	-	41	49 ± 11	21	70
Eight floor					
Room	1	125	68 ± 20	23	102
Kitchen	2	66	61 ± 15	33	97
Bathroom	3	5	101 ± 14	81	117
OUTDOOR	6	50	29 ± 12	14	39

and, therefore, a lower air ^{222}Rn concentration. The ^{222}Rn concentration is approximately 7.5 times lower (45 Bq/m^3). This could be explained partially by the presence of the cellar under the kitchen (where the diffused underground ^{222}Rn is temporarily deposited and easily evacuated through the windows of the cellar into the open air) and partially by the fact that the kitchen, by its function, is more frequently ventilated. The daily variations of the radon concentration in the kichen's air are show in Figure 1.

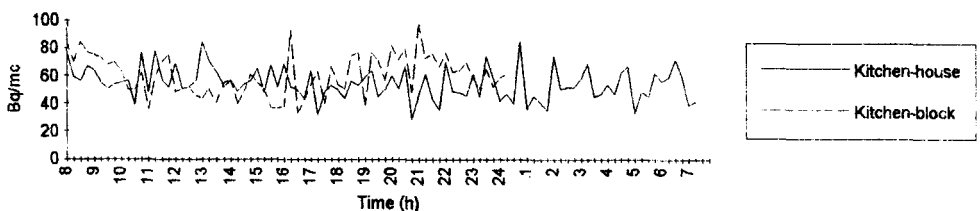

Figure 1. Distribution of radon concentration in kitchens' air.

The bathroom has no outside walls and there is no cellar under it, but the ^{222}Rn concentrations in the air are approximately 4.5 times lower as compared to those found in the rooms A and B, due to a permanently opened ventilation aperture through the chimney. Measurements in the air of the bathroom after running the shower for 15 minutes proved an increase in the radon concentration of 20 Bq/m^3.

Finally, the ^{222}Rn concentration in room C, being 3...4 times lower than those measured in the rooms A and B, can be explained by the presence of a concrete foundation that plays the role of a screen for the ^{222}Rn coming out from the soil.

Regarding the ^{222}Rn concentrations found inside the block flat, a lowering of the values along the first floors and a slight increase of them at the upper levels are to be noticed. The overall values are 2...3 times smaller than those measured in the detached house.

Out of the measurements performed in the initially ventilated rooms, during standard periods of 24 hours, it can be noticed that during the first 3 hours after closing the windows and the doors, the speed of ^{222}Rn accumulation is 50 Bq/m^3/hour in room A and approximately 90 Bq/m^3/hour in room B followed by a slower accumulation, of 22 and, respectively, 28 Bq/m^3/hour, determined during a 9 hour period. Sudden decreases of ^{222}Rn concentrations after the opening of the windows have been observed in all sampling points. During the first 15 minutes of natural ventilation, the concentrations reached values similar to those measured in the open air (Fig.2).

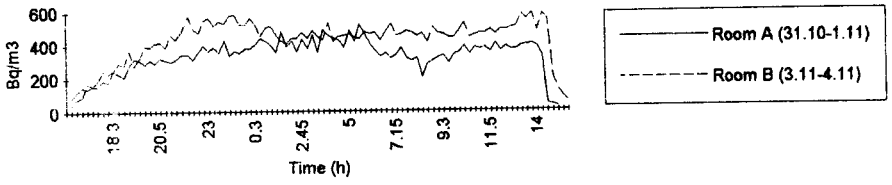

Figure 2. Radon accumulation in living rooms.

The deviations of the daily means from the previous day mean value ranged in the 10...24 % interval.
We have estimated respiratory system doses attributable to the inhalation of the radon and its daughters in dwellings for the investigated population groups from Cluj-Napoca (0.35 to 16..54 mSv/y). The maximum value of was obtained for people who live in detached houses. In Table 2. we give the Equivalent Dose Effective (EDE) and the weighted averages and the lung cancer risk for these groups of people.

Table 1. WEIGHTED AVERAGE ANNUALE EQUIVALENT DOSE EFFECTIVE INDUCED BY INHALATION OF ^{222}Rn DAUGHTERS AND LUNG CANCER RISK

BUILDING	INHABITANTS	EDE (mSv/y)	LUNG CANCER EXPECTED
DETACHED Ground floor (with cellar) (without cellar)	126000	2.99 2.47 6.50	7.41
BLOCK Ground floor First floor Four floor Seven floor Eight floor	224000	1.06 0.79 0.76 0.76 0.76 0.69	2.71

CONCLUSIONS

The measured values of indoor radon concentration have demonstrated that the high concentration levels in similar conditions can be attributed to radon entering the house through the building foundation.

Measurements of the ^{222}Rn concentrations in the indoor air of the same detached building showed values with a great variation. In the rooms with stone and slag basement, the ^{222}Rn concentration exceeded the value recommended by ICRP 65/95 (the recommended action level for the annual average radon concentration in air in dwellings should not exceed 200 Bq/m^3) (7).

In the future the increasing indoor exposure of the population to radon and its daughters might be the consequence of a reduced ventilation by mean of the energy conserving measures and utilizing the new building materials

Determining the radon concentration as well as the origins of the indoor radon contribute to finding out the solutions for reducing at most the risk due to ionizing radiations.

The significance of radon for the human health is well known and thus the measurement of indoor radon is important for the evaluation of its impact in the public health field. A difficulty in evaluating the risk of lung cancer is the evaluation of the cumulative exposure. The relationship between radon, its daughters exposure and other carcinogenic pollutants is still uncertain.

AKNOWLEDGEMENT

We aknowledge and thank to the inhabitants in whose flats these measurements were performed.

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RADON AND REPRODUTIVE HEALTH OF POPULATIONS LIVING IN THE REGIONS OF URANIUM MINERS ON NORD CAUCASUS.

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1. The estimation of the combined influence of technogenically increased natural background made by human activities and radon on the health in current and future generations of people is of great importance. A particular interest is the investigation of population exposure fraught with medical effects in the areas where the constructed houses were build with materials containing the higher doses of natural radionuclides emanating radon and its daughter products of disintegration.

2. One of the areas with the increased irradiation of the population due to Rn and its daughter products is the town of Lermontov. It is situated in the centre of the region of the Caucasian Mineral Waters. It is the territory of the Mineral Waters foothills plain near the mountain Beshtow (the hight is 1200 m).

3. The inhabitancy of this town are workers of the uranium output and processing enterprises. In the past for the house construction especially for the private living blocks, the local raw materials having the higher quantity of radionucleids were widely used as a material for building up houses.

4. From the date of the foundation of the town in 1957 to 1994 the number of its population increased from 11000 to 24300. The growth of the population in average is 300 persons per a year, with the natural growth being 100 - 200 persons a year. The most part of the population deals with the uranium output and processing inductry (within 1952 - 1991).

5. The range of gamma-background in the town of Lermontow is 10 - 120 μR/h, with the mean Russian index being μR/h. In addition, the higher level of emanating Rn from the soil reaching 4,5 Bq/m^2 per a second was observed. The high concentrations of Rn and its daughter products were found out in the living blocks of houses in the town of Lermontov.

6. The content increasing the level of 200 Bq/m^3 in the living flats on the ground floor of the multiflact houses was shown by the long-term control (the total number of analysis is over 2000) to be recorded in more than 50% flats examined, with the five-fold increase being (1000 Bq/m^3 and more) observed in 10% investigated living accommodations. In the air of the living accomodations analysed in 1994 the equivalent equilibrium volume activity of Rn was between 5 - 913 Bq/m^3 and that of toron is within the range of 0,3 - 17,2 Bq/m^3 . The maximum indices increase as high as 25 times the Russian standards for Rn daughter products as tot the toron product standard the index is 5,7 times higher.

7. Thus, in the town of Lermontov where for a very long term (during the three generations) the stable (in the migration aspect) population is living, there is a unique opportunity for making radiological, hygienic, clinical, epidemiological investigation together with scientists of different countries in the great schedule.

8. The results of epidemiological observations of uranium miners living in the houses with the higher Rn concentrations (that is they were under chronic influence of Rn and its daughter products for less then 20 hours

per a day are given. Besides, their familieswere also under the chronic influence (in less doses) of Rn and its daughter products.

9. The oncological mortality in the town of Lermontov for the last years was estimated to be higher in comparison with that of the town population in the Stavropol Kray. Particular attention is drawn to the fact of raising the death rate from pulmonary cancer.

10. It was an attempt to determine the exposure of Rn daughter products on the reproductive health of the population that was first made in studying the Rn influence.

11. The main indices of reproductive health are as follows: health status of the pregnant women, untoward outcomes of pregnancy (miscarriages and spontaneous abortions, premature born children, stillbirths, early neonatal and perinatal death of the newborns), the status of health of the newborns including the estimation of the frequency and character of congenital pathology. On the strength of all the aforementioned evidence it is possible to forecast the consequences of radiation on the health status of the current and future generations.

12. The results of the first step of the investigation made in 1994 - 1995 are full of discrepancy: namely, the rate of stillbirths is less, but there is a growth of newborn with normal adaptive weight (2500 g - 3499 g) and there is an increase of morbidity among the hypoxia and asphyxia (1,49%) and cephalohaematomes (5,24%). The frequency of congenital anomalies in newborns of the town of Lermontov increased 54% in 1993 compared to 1987.

13. The investigation performed between 1982 - 1994 showed the unfavourable medico-demographic evidence in the town of lermontov: the decrease in the birth-rate, the growth of total and oncological mortality, first of all due to lung cancer.

14. For the determination of the extent of correlation of the mentioned tendencies connected with the radioecological surrounding in the town of Lermontov the further investigations both in finding out the houses with Rn higher concentration and in estimating the health status of the population living in those houses remain to be arranged.

Human Exposure Assessment from Radon in Water - Comparison of Experimental and Theoretical Data

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ABSTRACT

In this paper are analysed the results of continuous monitoring of indoor radon in family house with high radon concentration in drinking water (4000 Bq/l). The simultaneous variation of indoor radon in the bathroom and other rooms while taking shower is studied. Human exposure during the stay in the bathroom and other parts of the house is estimated for different types of houses. Finally, assessment of human exposure from radon in drinking water in Eastern Bohemia Region is carried out.

INTRODUCTION

Human exposure caused by water having high radon concentration was studied in last years again[1]. One of the reasons is measured local short-term and extremely high radon concentration in the bathroom during showering or taking bath (water having Rn concentration of 1000 Bq/l , radon emanation during showering around 50 %, can cause in the bathroom volume of 10 m^3 short-term indoor radon concentration of 5000 Bq/m^3 and increase radon concentration in rest of the house). This paper presents estimation of human exposure during stay in the bathroom and during stay in the rest of the house and contribution of this exposure to total exposure from other radon sources.

POTENTIAL ALPHA ENERGY EXPOSURE DURING SHOWERING

Taking shower or bath is a process, that can be described by two steps 1) the water use with rapid radon release into the air, 2) the stay in the bathroom (some minutes to up 1 hour) in approximately constant radon concentration (ventilation rate is usually small) with increasing of radon daughters concentration. It is complex process in which steady state is not reached during normal stay of person in the bathroom (<1 hour) . From this reason human exposure cannot be easily evaluated from radon concentration but directly from potential alpha energy concentration (PAEC). This can be done by evaluation of time integral of potential alpha energy concentration PAEC (theoretically predicted or measured) during stay in the bathroom, or this exposure can be also experimentally measured by continual spectroscopy WL monitors, the response of which is proportional to human exposure in cases described below. This attitude was used in following experiment.

FIELD EXPERIMENT

Set of continuous alpha spectroscopy monitors (with Milipore filters 0.8 m, flow rate of 1 l/min and 3 minutes sample interval) were placed in the bathroom having volume of 11 m^3. During experimental showering 100 l of water was used having radon concentration of 4 100 Bq/l . Indoor radon concentration was measured by set of Lucas cells, maximum value of radon concentration immediately after showering was 20 700 Bq/m3 (ventilation rate was <0.15 h^{-1} · so decrease of Rn concentration during 1.5 hour experiment was <25 %,). After intervals 10,20,30,...90 minutes (simulation of different stay of person in the bathroom) was

the airflow through the monitors stopped in sequence, but monitors continued in alpha counting. Response of monitors is proportional to dose rate in the respiratory tract during and after stay in bathroom. From total registered number of alpha counts (and alpha counts from Po-218) can be calculated potential alpha energy exposure as a function of time spent in the bathroom (and the fraction of this exposure caused by Po-218 itself, this fraction varied from 50 % for 10 minutes stay to 25 % for 1 hour stay in bathroom) . Results of measurements were normalised to initial Rn concentration in bathroom of 5000Bq/m³ (that corresponds to use of 100 l water having of 1000 Bq/l with 50% radon emanation in the room of 10m³) and are summarised in the fig.1.

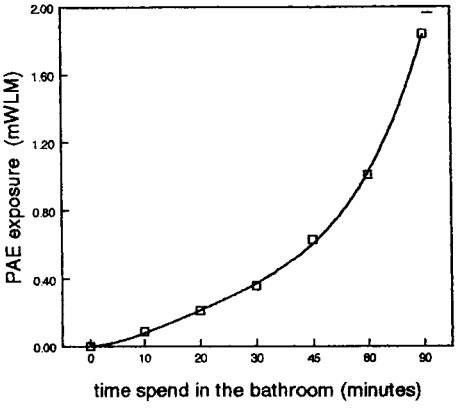

Fig. 1: Potential alpha energy exposure as the function of the time spend in the bathroom (indoor Rn conc.of 5000 Bq/m3)

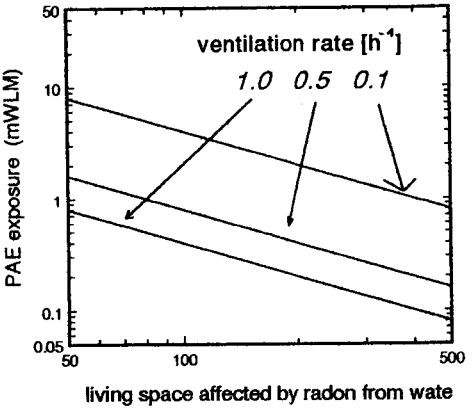

Figure 2: Potential alpha energy exposure in the space adjacent to the bathroom (for radon release of 50000 Bq)

EXPOSURE IN THE REST OF THE HOUSE

During showering usually well closed bathroom is then opened and radon is transferred into the adjacent part of the house and can contribute to elevated average indoor radon concentration. This process is described by transfer coefficient (the ratio of average indoor radon concentration caused by radon release from the water and radon concentration in the water) usually assumed to be 0.0001 (or lower[2]). In fact, there exist a lot of completely different situations depending on the mutual disposition of the bathroom and others rooms and their volumes and also on family habits in ventilation, etc. From these reasons, great variability of human exposures can be expected and therefore lot of scenarios must be considered. There exist two extreme cases: a) after showering the bathroom is immediately opened to the outer atmosphere and there is no radon transfer to the house, b) the bathroom is after showering opened only to the small living part of the house (e.g. one room relatively well closed from other parts of the house) and only this space is affected by radon release from water. Radon is then removed by ventilation Most of real situations are between these two extreme cases. Estimation of radon concentration can be evaluated from elementary consideration: radon release of total radon activity A is transferred from the bathroom to the space of volume V of the house, that is ventilated (average ventilation rate λ). Radon concentration in affected space is

$$a = A/V \cdot e^{-\lambda t} \qquad (1)$$

and PAE exposure can be approximately evaluated (such calculation, of course, overestimates PAE exposure due to non equilibrium state in this space, especially in case of high ventilation rate) from the time integral of radon concentration and equilibrium factor F (assumed to be F=0.5) as

$$PAEE\ [J\ h\ m^{-3}] = 5.6 \cdot 10^{-9} \cdot F. \int a.dt = 2.8 \cdot 10^{-9} \cdot A / (V. \lambda) \qquad (2)$$

$$PAEE\ [mWLM] = 0.00159 \cdot F. \int a.dt = 0.00080 \cdot A / (V. \lambda) \qquad (2)$$

Results of calculated PAE exposure for possible scenarios - different ventilation rate and different affected volume of the house - are on the fig 2. Calculation is done for one release of 50000 Bq of radon in the bathroom, what corresponds to use of 100 l of water having 1000 Bq/l and 50% radon emanation mentioned above. Total exposure is, of course, sum of the individual releases.

OTHER EXPOSURE

In the indoor radon survey carried out in Czech republic in last years by solid track detectors was discovered the average EEC of radon 68 Bq/m3 that correspond PAE exposure of $6.3\ 10^{-6}$ J h m^{-3} (1.8 mWLM) per day (for occupational factor 0.7).

CONCLUSIONS

Comparison of exposures (from one radon release) in the bathroom and in other affected parts of the house is seen from figure 1 and figure 2 (resp.equation 2). From these can be derived, that e.g. 1 hour of stay in the bathroom is equivalent to exposure in affected space of house of volume 100 m^3 and ventilation rate of about 0.5 h^{-1} . What exposure is dominating depends on individual habits and house disposition.

If water having of 1000 Bq/l radon concentration is used for showering (100 l of water in $10m^3$ bathroom) exposure about 1 mWLM can be expected in case of 1 hour stay in bathroom. This is similar to typical exposure from other radon sources in normal house. Beside this water with 1000 Bq/l radon concentration is, in fact, relatively rare case. Distribution of radon in ground-water in Eastern Bohemia (where the problem was studied) is lognormal with geometric mean value 23 Bq/l , that correspond to average PAE exposure during 1 hour showering only about $8.1\ 10^{-10}$ J h m^{-3} (0.023 mWLM.)

As a result, the exposure from showering in case of typical Rn concentration in water was found to be small compared to typical exposures from other radon sources. Even in rare cases of high radon concentration in water (1000 Bq/l) it is less then exposure from other radon sources in typical house.

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INDOOR RADON SURVEY IN JAPAN

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INTRODUCTION

Radon and its progeny are the major source of exposure to the general public. Large scale radon surveys have been performed in many countries with various passive radon detectors. Our group in National Institute of Radiological Sciences has also conducted a nation-wide radon survey in Japan using passive radon detectors which were originally developed at Karlsruhe Nuclear Research Center in Germany. The main purpose of our survey is to obtain annual average indoor radon concentration and to estimate dose due to radon progeny as well as to find out high radon areas in Japan.

METHOD

The survey was planed to cover more than 7,000 houses throughout Japan which corresponds to 0.017% of the total houses in Japan. The number of sampling houses in each prefecture was selected in principle in proportion with the total number of houses in prefecture in order to obtain radon concentration profile of the present stock of houses. The results are also expected to reveal the approximate population averaged radon concentration in each prefecture as well as the whole nation. Each house is measured in two places namely a living room and bed room for two successive periods of six months for the estimation of annual average radon concentrations. After six months of exposure in a house, the dosimeters were returned to our institute, NIRS by mail for the further processing. New detector foils (polycarbonate) were placed in the dosimeter housings and were sent again to the same house for the second six months measurements. These two sets of successive six months measurements provide annual average radon concentration in the selected houses. As a total more than 30,000 detector foils were used for the survey. The detector foils which were retrieved from the selected houses, were subjected to a combined chemical and electrochemical etching processes to develop etch pits due to alpha particles from radon and its progeny. The number of etch pits was then counted either automatically by an image counter or manually depending on the number of etch pits developed in the unit area ($1cm^2$) in the center of the foil. The etch pits were read manually when track density was more than 300 tracks/cm^2 since the machine could not distinguish overlapping etch pits. The overlapping correction for the counting by the machine were executed by an equation determined from the correlation between automatic and manual counting (1).

Self-background of 26 etch pits per cm^2 during the six months exposure as well as inherent background for each batch of foils were then subtracted from the measured track density (2) and the net counts were converted to average radon concentrations during the measurements. The conversion factor of 21.8 tracks/cm^2 per kBq/m^3 day was used to obtain radon concentration. The conversion factor was estimated by a series of calibration exercises which were made available by the Australian Radiation Laboratory, Melbourne, Australia and the Environmental Measurement Laboratory, New York, U.S.A.

The results obtained from more than 7,000 houses were carefully checked to identify abnormal conditions during the measurements. All the data measured not in a living room or bed room were eliminated. The results obtained from the house with no answer to the questionnaire were not included in the final results. Only the results with at least one set of successive six months measurements were used for the estimation of the annual average radon concentration. Total number of houses where the annual radon concentration could be obtained was reduced to 6,645 houses.

Additional measurements were conducted in Hokkaido, Hiroshima and Kochi Prefectures to clarify the thoron contribution to the first measurement undertaken throughout Japan. Two sets of passive detectors were installed in two places (living room and bed room) in each house. One passive detector was the same as the original detector used in the first survey. The other passive detector used plastic foil as a filter to prevent thoron entry into the housing of the detector. The air exchange rates of the original and modified passive detector were estimated to be 27.7, 1.79 h^{-1} by the experiment using SF_6 gas. The original passive detector measures radon

plus thoron and the modified passive detector measures mainly radon since the modified detector is expected to show one tenth of the response of the original detector to thoron due to the slower air exchange rate.

RESULTS AND DISCUSSION

The indoor radon concentrations obtained by the survey shows roughly a log-normal distribution as shown in Fig. 1. The estimated arithmetic mean of the radon concentrations in 6,645 houses is 21.3 Bq/m^3 (S.D. 18.8 Bq/m^3), and the geometric mean and its standard deviation are 17.3 Bq/m^3 and 1.83, respectively. The median of the concentration is 16.4 Bq/m^3. The number of houses having a concentration of ten times higher than the geometric mean is only 17 in 6,645 houses, i.e., 0.26% of the measured houses. Ninety percent of the houses are less than 38 Bq/m^3, 97.5% less than 68 Bq/m^3, 99% less than 96 Bq/m^3. Only 27 houses are over the action level set by EPA in the U.S.A. These high radon concentration houses are distributed in the western part of Japan except one house found in Niigata Prefecture. However, no radon prone area is found, although Hiroshima Prefecture has 6 high radon houses out of total 27 houses. All these high radon concentration houses are wooden houses except two concrete houses in Okinawa Prefecture. The average radon concentrations in each municipality (city, town, village) are shown in Fig. 2. Many higher radon concentrations are found in Chubu, Kinki, Chugoku and Shikoku districts, where winter is relatively mild and the air exchange rate of the houses in these regions is not low, the concentrations seems to have a relationship with the geology as found by the measurements of exposure rates due to terrestrial gamma rays (3). These areas are typical granite region in Japan. The areas covered with volcanic ash, in Kanto District around Tokyo and Kagoshima Prefecture in Kyushu district shows lower concentrations.

The results of the additional measurements carried out in Hokkaido, Hiroshima and Kochi Prefectures shows three different profiles of the correlation between radon and thoron concentrations. In Hokkaido Prefecture no significant thoron contribution was found in the indoor radon measurements. The arithmetic, geometric mean and median for radon plus thoron concentration (27.4, 21.6, 21.4 Bq/m^3) measured by original detectors are very close to the corresponding values for radon (26.3, 20.4, 19.9 Bq/m^3) measured by modified detectors. In Kochi Prefecture the difference is somewhat larger, i.e., the arithmetic, geometric mean and median for radon plus thoron concentrations are 20.4, 16.3, 15.4 Bq/m^3, while the corresponding values for radon are 14.3, 12.9, 11.9 Bq/m^3. The difference is much significant in Hiroshima Prefecture. These values are 49.9, 36.3, 32.0 Bq/m^3 for radon plus thoron, and 27.8, 24.5, 24.3 Bq/m^3 for radon. No indication of large contribution of thoron is available from the values, such as mean, standard deviation, percentile. Most high thoron concentrations were found in traditional wooden houses where they have walls made of clay and straw. The straw is used for adhesive material to keep clay in the wall. Thoron might pass through the air space in the straw and reach indoor air without much delay usually confronted during the diffusion in the normal wall. It infers that thoron concentration may have high values in wooden houses. However, other type of houses may sometimes have high thoron contribution as found in Austria with the same way of measurements by the authors in collaboration with Dr. M. Tschurlovits in Atom Institute of Austrian University.

CONCLUSION

The arithmetic mean radon concentration was estimated to be 21.3 Bq/m^3 (S.D. 18.8 Bq/m^3) by a nation-wide survey in Japan. The geometric mean was found to be 17.3 Bq/m^3. Ninety nine percent of the measured houses are less than 96 Bq/m^3. No radon prone area was found, although a little elevation of radon concentration was found in the western part of Japan due to granite formation. Our detectors have a disadvantage of detection of thoron since the air filtration rate of the detector is very high. The estimated radon concentration based on the first survey of indoor radon might have been overestimated due to thoron contribution to the radon measurements. In the additional survey one prefecture showed no thoron contribution while the other two prefectures showed significant thoron contribution in the radon measurements. No clear-cut indication of high thoron houses has not been obtained.

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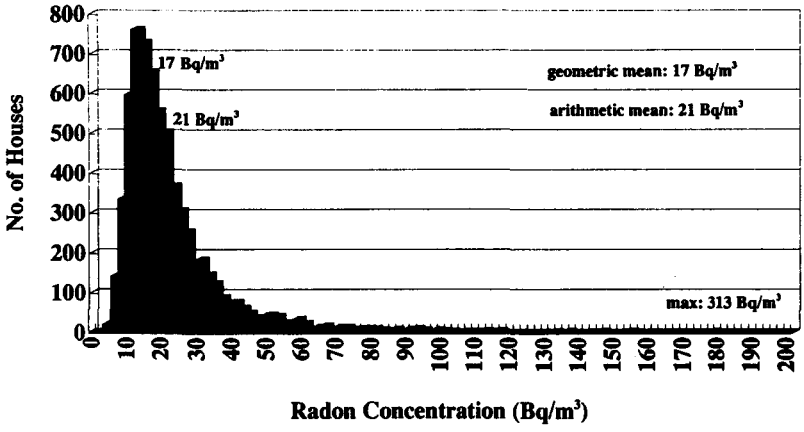

Fig. 1. Histogram of Indoor Radon Concentrations (6645 houses)

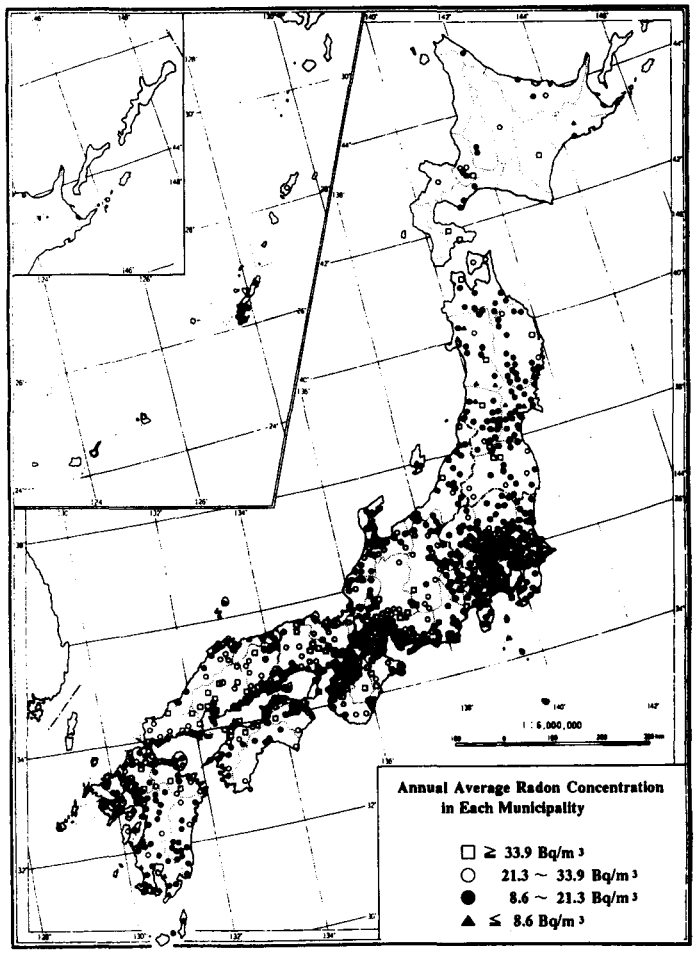

Fig. 2. Distribution of Indoor Radon Concentrations in Japan

RADON DETERMINATION IN GROUNDWATER UNDER CONTROLLED CONDITIONS

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1. INTRODUCTION

Radon is a radioactive gas known to occur in groundwater. During the last years as well public interest in dosimetrical aspects of radon in Germany [1] as in scientific applications using radon as a natural tracer in hydrology and hydrogeology (e.g. [2]) has increased significantly.
Sampling and measurement of radon in water are complicated and often incorrect caused by the volatility of the gas.The aim of this work was to develop and test a low cost radon measurement method including sampling, transportation, and analysis, which allows a quick measurement of many samples with high reproducibility.
The paper describes a modified [3] experimental procedure based on liquid scintillation spectrometry

2. SAMPLING TECHNIQUE

2.1. Sampling devices

Different water sampling devices (submersible pump, membrane pump, bailer) were compared to check their applicability for the sampling method to measure radon concentrations. A submersible pump and a membrane pump were used together in a observation well for a continuous sampling. The period of representative sampling starts after 40 minutes of flushing. The radon concentration of samples collected with the submersible pump is found to be (2690 ± 60) cpm/L and those collected with the membrane pump obtained to (2670 ± 100) cpm/L. Therefore, both pumps lead to the same results. Comparing the two single measurements of the bailer with the pump series means, that all of the used devices are suitable for sampling.

2.2. Flushing time

Every sampling starts with flushing the water located in the observation well and in the screen zone. The time for representative sampling must be determined. A too short initial flushing time represents a major source of error, because the measured radon concentration could be influenced by mixing of the pore water of the filter gravel, stagnant water of the observation well and water influenced by natural emanation rate of the aquifer. Figure 1 shows the behaviour of the radon concentration during continuous sampling in a deep observation well (well screen 44 m below land surface) in which the pump depth is only 9 m below land surface. In the first period of flushing the stagnant water radon concentrations of groundwater samples increased rapidly by 7000 cpm/L. After a time of about 35 minutes (2.6 standpipe volumes of water plus pore water of filter gravel) a representative sampling is possible. The radon concentration on this observation well was (7920 ± 150) cpm/L (29 measurements). Some tests lead to the proposal for practical use to fix the flushing time as a function of the sum of standpipe volume of stagnant water in an observation well plus the pore water of the filter gravel. For qualified measurements of radon concentrations in aquifers flushing with four to five standpipe volumes are necessary. If the pumping depth equals the screening depth the flushing time can be reduced drastically.

The behaviour of radon concentration during flushing gives a new reliable tool to fix the beginning of representative sampling for any chemical parameter especially for older, unknown observation wells. Normally, the constant electrical conductivity and constant water temperature is being accepted to indicate the end of flushing time. However, these are necessary conditions but not sufficient ones. A continuous sampling and measurement of radon concentrations leads to an increasing concentration curve as a function of flushing time or number of removed standpipe volumes starting at a low level and approaching a constant plateau value. The approaching of the plateau is the moment where representative sampling may start for any parameter. It is often necessary to check unknown observation wells for possible use in future measurement campaigns, in these cases radon concentration is the most reliable parameter to do that.

2.3. Filling arrangement

The sample bottle and the filling arrangement must be designed in relation to the mobility of the radon gas. For this reason a glass mensure calibrated to one litre with a narrow neck sealed by a polyethylene stopper with teflon gasket was chosen as sample device. It realises sampling with minimisation of loss of radon. At first

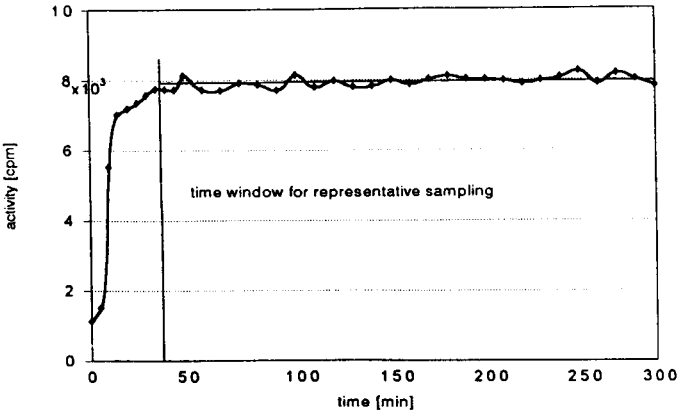

Figure 1: Behaviour of the radon concentration during continuous sampling in a deep observation well

20 mL toluene scintillation cocktail is filled into the mensure. After that the water is filled in through a glass tube specially made for radon measurement by underlayering the scintillation cocktail without air contact. Sampling has to be free of turbulence and air bubbles. After filling the glass tube is removed and the mensure is closed with the polyethylene stopper with teflon gasket for transport. The usefulness of the above described filling arrangement was tested with several measurement series. The results show an improvement of measured radon concentration in groundwater samples between 2 and 8 % depending on the quantity of natural radon concentration in the aquifer. A second reason to use that filling arrangement is the better objectivity of sampling.

3. MEASUREMENT TECHNIQUE

The activity concentration of radon in groundwater is determined as described by HORIUCHI and MURAKAMI [4]. Radon solved in water is extracted by toluene mixed with liquid scintillator cocktail. The toluene phase is filled into a glass vial. The radon activity of this toluene phase is determined by liquid scintillation spectrometry. The calibration of the measurement is carried out by using a liquid solvent of Ra-226, which is in equilibrium with its daughter nuclides especially with Rn-222. The water sample volume amounts to 1 litre. A commercial toluene cocktail with 5 g PPO and 0.1 g POPOP per litre is used as extraction and scintillation cocktail.

The water samples are transported to the laboratory in glass mensures sealed with a polyethylene stopper with teflon gasket. During the transport the water sample is covered by 20 mL toluene cocktail. 10 equivalent water samples were stored after transportation 4.5 hours, 1 day, 2 and 5 days. The measurement results relative to the sampling moment did not change.

The transport mensures with the water samples and the floating scintillation cocktail are shaken in the laboratory to reach a stable distribution of radon between water and toluene. The shaking time amounts to 10 minutes. After shaking a first extraction of 10 mL toluene cocktail is removed and filled into a glass vial, 10 mL fresh cocktail is added to the water sample. Afterwards, the mensure with the water sample and the scintillation cocktail is shaken a second time for 10 minutes. At last, 10 mL of the toluene cocktail is removed again and added finally to the 10 mL in the same glass vial to measure the radon concentration in the next step.

For the measurement of the radon activity high performance glass vials produced from borosilicate glass with cork/aluminium sealed caps and a liquid scintillation spectrometer TRI-CARB 2550 TR/AB (Packard) with a special hard- and software option for α/β discrimination were used. The optimum measuring time is 60 minutes. The measurement error is < 5 % for count rates between 40 and 8000 cpm. To reach a high sensitivity the energy window has a range from 0 to 2000 keV (open channel). Before sampling the background count rates of all used vials are measured with 20 mL scintillation cocktail. After the sampling and extraction procedure the cocktail is filled back into the same vial. The vials are measured about five times in a period of five days. Before the first measurement starts it has to pass a waiting time of at least three hours to reach the equilibrium between radon and its daughter nuclides.

4. EVALUATION AND CALIBRATION

The results of the TRI-CARB measurements are determined in units of counts per minute (cpm). It is essential to control that only the activity of radon is measured. The TRI-CARB 2550 TR/AB has two possibilities to check the correctness of the results:

- Using the software SPECTRAGRAPH it is possible to view the α spectrum (Fig. 2). The peaks of radon-222 (5.49 MeV), polonium-218 (6.00 MeV) and polonium-214 (7.69 MeV) can be distinguished.
- The α/β discrimination allows to calculate the α/β ratio after calibration e.g. by Am-241/C-14. In the case of radon-222 the ratio must be 60/40.

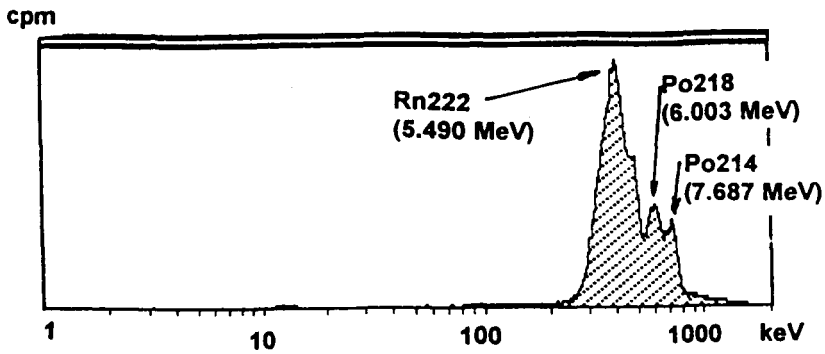

Figure 2: Alpha spectrum of radon-222 and its daughter nuclides represented by TRI-CARB software SPECTRAGRAPH

For the exact measurement of radon activities in water by liquid scintillation spectrometry one of these two possibilities should be integrated in the used equipment.

By means of five measurements in a five days period the radon activity in the moment of sampling can be calculated by regression. Also the possible untightnesses of the vials are visible in this calculation procedure. Then the measured half life times are apparently shorter.

For calibration the radon activity concentration of ground water samples was determined by gamma spectrometry using the daughter nuclides Pb-214 and Bi-214. The calibration factor of $(8.3 \pm 0.7)\ 10^{-3}$ Bq/cpm was calculated related to the mean value of the results of liquid scintillation spectrometry of similar groundwater samples. However it must be taken into account that the chemistry of the water (characterized e.g. by pH and electrical conductivity) can influence the efficiency of the radon extraction.

The presence of radium-226 in the water sample makes the measurement results errorprone because it produces radon-222 continuously. To exclude this possibility the water samples should be stored for about 20 days and then measured again.

5. RESULTS AND CONCLUSION

A modified experimental method for the determination of radon in water including all steps of sampling, sample transportation and analysis has been tested successfully.

The method is characterized by a new onsite technique for sampling water by pumping with minimized aeration and losses of radon during transportation. Submersible pumps as well as other sampling systems like membrane pumps or bailer are equally useful. The waiting time between sampling and measurement can be up to five days. The radon is extracted from the water sample by extraction using toluene scintillator cocktail The radon activity is measured by liquid scintillation spectrometry

The described method is useful for measurement of radon concentration in groundwater samples as well as in other water samples like drinking water, surface water or spring water.

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INTEGRATING MEASUREMENTS OF RADON IN DWELLINGS IN SOME REGIONS FROM ROMANIA

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INTRODUCTION

Many epidemiological studies especially made on different cohorts of uranium miners clearly showed the influence of radon and radon progeny in lung cancer diseases (1).

At present, about ten important epidemiological studies regarding indoor radon concentration and lung cancer risk are in the world under observation (2).

The indoor radon concentrations are strongly dependent on the meteorological factors, the ventilation conditions and on the radon soil potentiality. In some earth areas the radon generated in upper part of the lithosphere or in the wall material may produce high radon concentrations in dwellings (3).

In Romania have been performed measurements in uranium and nonuranium mines (4) also in some country centres (5) by Kusnetz's method. The Kusnetz's method being an instantaneous method for indoor radon concentrations may not provides a correct estimation for average indoor radon content. In some cases, if the measurements are made in determined conditions (after a few hours of the windows and doors closing) it is possible to obtain a certain average value for indoor radon concentration more or less deviated from the real value (6).

Therefore, for a good estimation of the average indoor radon content, the measurements may be made for 3-6 months period, that is integrating determinations. Such measurements usually performed by track method or by electret detectors in the last years (7) are used for the radon indoors in dwellings.

Beginning from 1994 under collaboration with Gent University (Belgium) and NRPB (England) financed by CEC in Romania were started the first integrating measurements of radon in dwellings.

This work presents these measurements in three regions of Romania. In the first two, respectively the Bihor and Cluj districts also an epidemiological pilot study in connection with the lung cancer risk due to radon was started. The third region is the Herculane Spa area where also many radon determinations in different environmental factors exist (8).

EXPERIMENTAL METHOD

The Karlsruhe radon diffusion chambers using makrofol electrochemical etched track detectors were used (9). After a previous chemical etching (0.5 hours) at 25 °C using a 6N KOH solution follows the electrochemical etching performed in the same solution applying an effective voltage of 800 V at 2 kHz during 3 hours (Gent conditions) (10).

The makrofol detectors were exposed for 3 months period in different buildings. The indoor radon concentrations were also performed by the Kusnetz's method using the scintillation flasks of 0.5l which were calibrated with romanian $RaCl_2$ standard solution.

RESULTS AND DISCUSSION

Table 1 presents the average values and the standard deviation for three areas from Romania. For the first two areas (Cluj-Napoca and Oradea cities) the measurements have been made in the cold season

and for the Herculane Spa region the measurements were made measurements both in cold season and warm season. For comparison in the last column of this table we present the measurements for indoor radon obtained by the Kusnetz's method. The values of 76 $Bq.m^{-3}$ and 115 $Bq.m^{-3}$ in each town represent the mean value of 30-35 dwellings and in each dwelling these were measured twice a year in cold and in warm season. These values are smaller than integrating measurements made only in cold season when the ventilation is smaller.

A rather high value was find for ^{14}C-laboratory (1d) and for the cave of one author (1e).

The measurements made in Herculane Spa show an average value for the cold season (185 $Bq.m^{-3}$) of two times greater than the average value for the warm season. Figure 1 also shows the indoor radon distribution for the cold and warm season.

Table 1. Integrating measurements of radon in three areas of Romania.

No.	Place	Number of dwellings	Average indoor radon ($Bq.m^{-3}$)	Standard deviation ($Bq.m^{-3}$)	Time period	Kusnetz's method (5,6) ($Bq.m^{-3}$)
1	Cluj-Napoca city					
	a) houses-downstairs	11	182.6	102	10.12.94 to 10.03.95	76
	b) block flats	8	125.7	20	idem	
	c) University-downstairs	4	127.3	21	idem	(annual
	d) ^{14}C laboratory (ITIM)	1	282	-	idem	average)
	e) author cave	1	1060	-	idem	
	f) Dej (town in Cluj district) houses-downstairs	2	100	15	06.12.94 to 06.03.95	
2	Oradea city houses-downstairs	5	124	15	06.12.94 to 06.03.95	115 (annual average)
3	Herculane Spa					
	a) buildings-downstairs in warm season	15	98.2	82	16.04.94 to 16.07.94	
	b) the same dwellings in the cold season	15	185.1	68	20.11.94 to 20.02.95	-
	TOTAL COULD SEASON AVERAGE (without 1e and 3a) 161 $Bq.m^{-3}$					

The mean value in the cold season (161 $Bq.m^{-3}$) obtained for all measurements from Table 1 is high enough but many integrating determinations are necessary for configuration of this value.

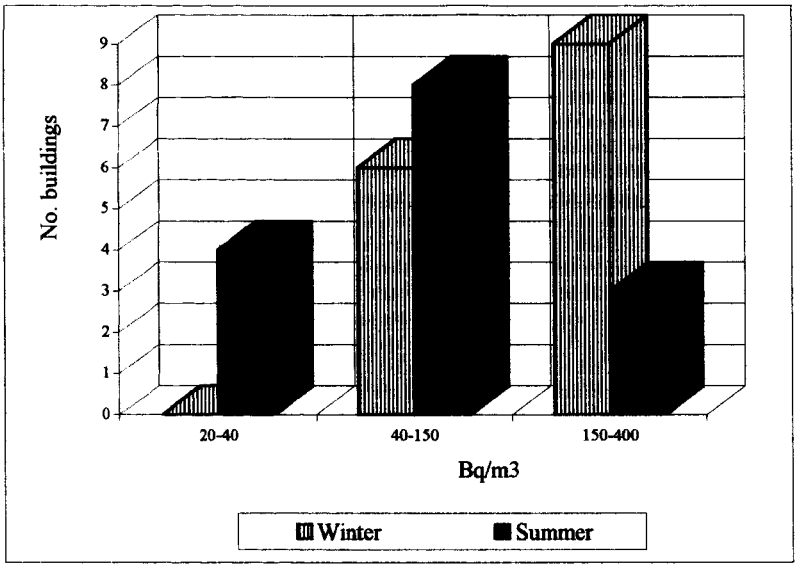

Figure 1. The distribution of indoor radon concentrations in Herculane Spa .

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MEASUREMENT OF ^{222}Rn AND ^{220}Rn EXHALATION RATES BY MEANS OF ALPHA SPECTROSCOPY

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Abstract

The measurement of ^{222}Rn and ^{220}Rn exhalation rates by means of alpha spectroscopy are presented. A metallic hemisphere cup with a hole at the bottom of the cup was used. A surface barrier detector is placed on the top of the hole and sealed with silicone in order to provide good insulation. Radon flux emitted from the surface of the material was measured by placing the inverted cup mentioned above on the top of the material. The increase of activity due to ^{222}Rn and ^{220}Rn inside the inverted cup was continuously measured by electrostatic deposition of ^{218}Po and ^{216}Po onto surface barrier detector. This method can also be applied in the measurement of radon exhalation rate from soil and building materials.

Introduction

The production of ^{222}Rn and ^{220}Rn in terrestrial materials depends on the activity concentration of ^{226}Ra and ^{228}Ra present. The UNSCEAR (1993) report indicates that the precursors of ^{222}Rn and ^{220}Rn have about activity in earth and building materials, and the rates at which the two isotopes are produced are also about equal. It is usually assumed that the emanation fraction is the same for both of them. The ratio for the potential alpha energy concentration of ^{220} Rn to ^{222}Rn is about 0.62, and the dose contribution from α decay of ^{220}Rn and its daughters is about 38.27 % of Rn isotopes[1]. An instrumentation was developed to detect ^{222}Rn and ^{220}Rn by α spectroscopy with electrostatic collector and to measure ^{222}Rn and ^{220}Rn in some of the building materials available in Taiwan.

Experimental

A metallic hemisphere cup with a hole at the bottom of the cup was used. The hole has a diameter of 2.6 cm. A surface barrier detector was placed on the top of the hole and sealed with silicone in order to provide good insulation. The opening part of the cup connects to a fine metal net. The circuit between the metal net and the cup must be open. The whole device is shown in Fig. 1. The detector is connected to preamplifier, amplifier, analogy / digital converter, multiple channel analyzer and personal computer to analyze the data. When ^{222}Rn and ^{220}Rn decay to ^{218}Po and ^{216}Po, the daughters are mostly positively charged. This positive electricity creates an electrostatic field. Then they were expelled to the bottom of the cup and collected on the surface barrier detector by electrifying the cup with high voltage. The α spectrum analysis for the decay daughters can infer the radioactive concentration of ^{222}Rn and ^{220}Rn [2].

The Calibration of Detection Efficiency

The radioactive source used is a Pylon Model-1025, standard radon source. The activity of radium is 1147 k Bq. The equilibrium activity of radon is 1147 k Bq. As shown in Fig. 2 its detection efficiency is 23.2 % for ^{222}Rn and 17.4 % for ^{220}Rn. Because the decay daughters of

^{222}Rn and ^{220}Rn, namely ^{218}Po and ^{216}Po, are positively charged, the detection efficiency of them will be increased with the increasing positive electricity electrified to the metal cup. As shown in Fig. 2 when the voltage is higher than 3600 V, the detection efficiency for ^{218}Po will achieve a steady value, 87%. Meanwhile, when the voltage is higher than 4800 V, the detection efficiency for ^{216}Po will achieve a steady value, 52 %. Therefore, the collector has a higher collection efficiency for ^{218}Po. The half-life of ^{220}Rn and ^{216}Po is very short (55.6 and 0.15 respectively). Therefore, even the detection efficiency can be improved by electrifying, they can not reach the surface of the detector.

The measurement of ^{222}Rn and ^{220}Rn Exhalation rates in Building Materials

The voltage was set at 4000 V with relative humidity (R. H.) between 41.0 ± 0.2 % and temperature was placed 25 ± 2 °C. The building materials sample was placed in the cup. By measuring α-particle decayed from ^{222}Rn and ^{220}Rn and analyzing the α spectra, the concentration of ^{222}Rn and ^{220}Rn can be obtained. There are two exhalation rates: the initial exhalation rate and the equilibrium exhalation rate. The former is the instant exhalation rate after the building materials is airtight. It is influenced by the quantity of radium and the density of cracks in building materials. The latter is the exhalation rate while the production rate and the decay rate of ^{220}Rn and ^{222}Rn achieve equilibrium in the cup. In this experiment 8 different kinds of building materials were selected. The results obtained are listed in Table 1.

Results and Discussion

The detection efficiency can be raised gradually when the applied voltage is over 3600 V. The collection efficiency of ^{218}Po and ^{216}Po decreases as the R. H. increases and the latter has a greater influence on ^{216}Po rather than on ^{218}Po. While the R. H. varies from 10 % to 40 %, the collection efficiency declines from 40 ~ 50 % to 15 ~ 40 %. Table 1 shows the exhalation rate of building materials, and the equilibrium exhalation rate of ^{220}Rn is higher than that of ^{222}Rn. Inside the collector cup, air has the least amount, of ^{220}Rn with a concentration of 0.008 $kBqm^{-3}$. However, marble and granite have the highest amount, 2.78 $kBqm^{-3}$. The average concentration of ^{222}Rn in air tested by 8 different building materials is about two times greater than that of ^{220}Rn. The average concentration ratio of ^{220}Rn to ^{222}Rn is from 0.13 to 0.44 which means the quantity of ^{220}Rn is about one-third of that of ^{222}Rn. This result matches with the published documents in which the dose contribution of ^{220}Rn is about 30 % of radon[1,3,4]. The experimental setup mentioned above can thus improve the detection efficiency, measure the amount of ^{222}Rn and ^{220}Rn, and detect them instantaneously or continuously.

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Table 1 The exhalation rates of ^{222}Rn and ^{220}Rn and radon concentration in the common building materials in Taiwan

Number	Name	Equilibrium exhalation rates				Average concentration		
		^{222}Rn	Average	^{220}Rn	Average	^{222}Rn	^{220}Rn	^{220}Rn / ^{222}Rn
		($\times 10^{-4}$ Bqm^{-2}s^{-1})		($\times 10^{-2}$Bqm^{-2}s^{-1})		kBqm^{-3}	kBqm^{-3}	
1	Marble	0.07~0.13	0.10	0.24~1.32	0.78	0.06	0.008	0.13
2	Quartz brick	0.09~0.31	0.20	0.11~0.67	0.39	0.12	0.04	0.33
3	Floor brick	0.10~0.16	0.13	0.02~0.21	0.11	0.08	0.012	0.15
4	Red floor brick	0.07~0.16	0.12	0.02~0.27	0.14	0.07	0.015	0.21
5	Concrete	0.51~2.11	1.31	0.50~6.53	3.51	0.78	0.36	0.46
6	Terrazzo floor	0.17~3.23	0.27	0.16~0.82	0.49	0.16	0.05	0.31
7	Slate	0.09~.23	0.16	0.08~0.31	0.39	0.09	0.04	0.44
8	Granodiorite	0.37~5.31	2.84	1.16~11.11	6.14	1.73	0.63	0.36

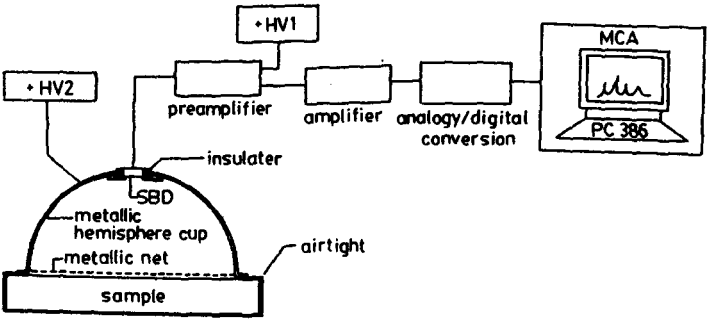

Fig. 1 Block diagram of experimental setup for measuring ^{220}Rn and ^{222}Rn.

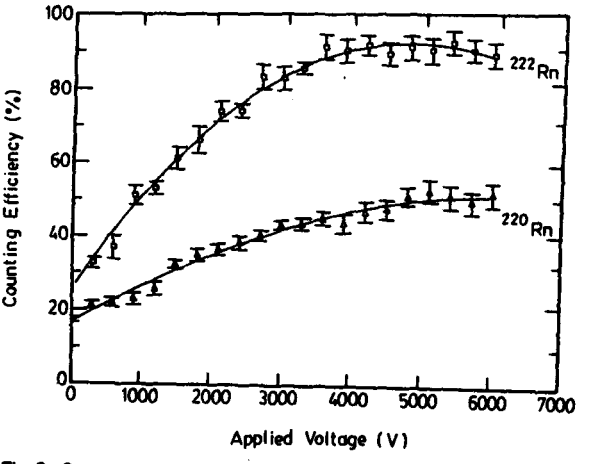

Fig. 2 Counting efficiency vs. applied voltage
(Cup diameter 9.4 cm, Temperature 22± 2 °C,
Relative Humidity 40.0±0.2%)

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PAPER TITLE Estimation of Rn Activity Level and Dose of Resident in Changchun City Area

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 1.1. Radon

ABSTRACT (See instructions overleaf)

This paper reports the monitored results of Radon in air indoor and outdoor in Changchun city area with static accumulation collector for the daughters of Radon from July 1991 to July 1992. The static accumulaton collector is developed, designed and was supplied to us by University of NAGONA, Japan.

Six monitoring points were selected in Changchun city area, the collector is laid horizontally, settled at 1 m above the ground of monitoring point, the sample is collected every two months. Annual average activity of Radon is 9.7~1.8 bq.m^{-3} outdoor, 18.6~6.5 bq.m^{-3} indoor, 35.6~8.6 bq.m^{-3} in underground plaza.

HE/a for lung of residents was estimated according to the parameters published by UNSCEAR, the value of 0.76 msv a^{-1} is obtained

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PAPER TITLE

Estimation of Rn Concentration and Population Dose in Underground Construction of China

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 1.1. Radon

ABSTRACT (See instructions overleaf)

This paper reports the estimation of Rn concentration level, and population HE/a in underground construction in part of cities in China. In this study, the average staying time for working people is 9 hours in underground construction , 5 hours staying outdoor on the ground, 10 hours staying in construction on the ground.

HE/a for people working in underground construction is 0.82 - 1.44 msv a^{-1}. The Rn concentration in underground construction is 1.30 - 7.36 time as large as the value in construction on the ground. That means strenghtening ventilation of underground construction, or other measures to reduc Rn concentration must be taken for people's safety reason.

MANUSCRIPT CUT IN LENGTH !

FOR FURTHER INFORMATION PLEASE CONTACT AUTHOR

EVALUATION OF RADON DOSES OF THE POPULATION OF KARELIA AND COLA PENINSULA

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INTRODUCTION

In opinion of the experts of UNSCEAR, radon and radon irradiation are major components of a natural radiating background. The effective equivalent doze of the planet population irradiation by radon and its affiliated products makes not less than halves of total doze from all natural sources of radiation (up to 70 %).The inhaling of the air, containing 1 Bq m^{-3} of radon, by different literary sources creates an effective doze on lungs from 50 up to 150 μSv y^{-1} (1, 2).

The natural sources of radiation influence the people, both in municipal and industrial spheres.

The heaviest contribution to irradiation of the population introduce radon and its affiliated products, locating in air of premises. The separate groups of the population can receive at the expense of the inhaling of radon extremely high emergency dozes (thousand BER y^{-1} on lungs). On UNSCEAR data , from 20 up to 40 % of all lung cancers are stipulated by radon and its affiliated products.

Total for autumn- winter stages of the work 950 detectors were installed and 1310 instant measurements of radon and its daughters were conducted. It is surveyed 47 towns, from them in details Petrozavodsk, Murmansk, Vyartsila, Snezhnogorsk.

ANALYSIS OF DATA ON KARELIA

The geometrical means of radon concentration in Karelia for separate towns by results of integrated measurements of a "pure" radon(i.e. radon without its daughters products) are represented for 42 towns in a diagram on fig. 1. The analysis of all set of available materials permits to reveal some general laws, describing the contents of a radon in houses of Karelia: for the majority of towns a rather high similar average level of concentration of a pure radon is characteristic. Geometric mean (GM) on whole Karelia is 72 Bq m^{-3}, and expectation value for lognormal distribution is equal to Bq m^{-3}. The difference between maximum and minimum values does not exceed 2 - 4 times. Lower GM of radon concentration was found in Belomorsk and Kemm (up to 50 Bq m^{-3}). Probability of excess of allowable values of radon concentration, determined from parameters of lognormal distribution, makes for whole Karelia about 1 %. Thus, on a background relatively high stable year concentration of radon in air of premises for towns of surveyed regions as a whole, significant percent of buildings with extremely high its levels emerges. The analysis shows, that the highest levels are registered in small settlements, i.e. the design of buildings plays here the significant role. The analysis of the relationship of geology and radon levels with using geological data and data on Finland and Cola peninsula shows the following: - correlation of high radon levels with intrusion of rapakivy granite is doubtless.

In Finland the average concentration of radon in region with such geology lies within the limits of 200 - 300 Bq m^{-3}. The similar geological structure is present in Karelia on north-east coast of Ladoga lake. The least concentration of a radon (30 - 50 Bq m^{-3}) is marked in gneiss zones and in amphibolites of belomorsk series. The large part of Karelia, as a geological structure, belongs to regions with most ancient gneisses and oligoclases granitoids. For them it is possible to expect wide spreading of radon concentration, from 70 up to 400 Bq m^{-3}.

ANALYSIS OF DATA ON COLA PENINSULA

Results of integrated radon measurements are represented for 6 towns in the diagram on fig. 2. The analysis of set of available materials does not allowed to reveal general laws, describing the contents of a radon in houses of Murmansk area: in surveyed towns average levels of radon concentration (GM on all Cola peninsula is equal 29.6 Bq m^{-3}, and expectation value for lognormal distribution is equal 40.7 Bq m^{-3}) lie in a wide range from 20 up to 93 Bq m^{-3}. Lower GM of radon concentration was found in Murmansk and Kovdor (up to 25 Bq m^{-3}).

The analysis of measurements distributions shows, that, as well as in Karelia, wide spreading of radon levels from several ten up to thousand Bq m-3 is observed. Probabilities of excess of allowable levels of radon concentration, determined from parameters of lognormal distributions, equal for Murmansk - 0.001 %, for Kovdor - 0.01 %, for Kirovsk and Revda - 0.1 %

DETAILED INSPECTION OF PETROZAVODSK AND VYARTSILA

Measurements of radon were executed within the limits of 12 sites, located in regular intervals within the limits of Petrozavodsk. Total in Petrozavodsk more than 600 measurements were conducted. By results of

conducted research is established, that the background values of equivalent equilibrium volumetric activity of radon (EEVA) in buildings of various purpose change from the first units up to first dozens Bq m^{-3}.
On the majority of surveyed sites abnormal objects with volumetric activity of radon and its daughters products from the first hundreds up to thousands Bq m^{-3} (Center, Kukkovka, Perevalovka) are found out. Number of buildings with the abnormal content of radon, more than 100 Bq m^{-3}, is about 9 %.
However, using the complex approach to valuation radon danger of buildings (volumetric activity of radon and its daughters, specific flow of a radon), the total number of potentially radon dangerous buildings reaches 34 %. The accumulation EEVA in a premise and determination of radon flow, used in this work, are described in (3).

Attracts attention the fact, that objects with a high radon EEVA level were poorly found out at the survey. At the same time high levels of a "pure" radon are found out at low EEVA values, that specifies high receipt of radon and potential opportunity under other conditions of reception of very high EEVA values, down to dangerous levels.

The valuation additional deaths at the expense of irradiation by radon only at Petrozavodsk (calculations are made in conformity with ICRP 50 publication) makes of 60 cases per year.

In Vyartsila settlment, located directly on border with Finland, active construction both of industrial premises and residential fund was assumed within the framework of economic commonwealth. Provided emanation survey of the settlement territory differs from all spent early by direct measuring of radon flow from a soil surface, instead of measuring volumetric activity of radon in soil air . The measured flow of radon from a soil surface on different sites was in a range from 10-15 up to 300 Bq m^{-2} h^{-1} with average level equal to 40 Bq m^{-2} h^{-1}. The volumetric activity of a radon in soil air was from 20 up to 100 kBq m^{-3} at average 60 kBq m^{-3}. The maximum registered value was equal to 150 Bq m^{-3}, at average on surveyed premises equal to 30 Bq m^{-3}. Survey in Vyartsila settlement has clearly shown, that it is not enough to have knowledge only about the radon content in soil air. Accounts of the radon contents in the basements and radon flow very well correlate by average values.

THE DOSE LOADS ON THE POPULATION OF KARELIA

The average level of irradiation by the expense of availability of radon in air of premises (on average EER) makes for Karelia 4,5 mSv in a year. In separate settlements range of fluctuations from this average value is about 4 times decrease or increase (from 0.8 up to 14.3 mSv in a year). Annual ED of irradiation, were designed for the adult . With the children in age till 10 years they can be at 1.52 time above (1,2).These dozes make up in average about 0.3 from the accepted normative level. However with separate inhabitants, living in houses with high radon levels in air, the last value can be exceeded 23 times (as it is, for example, in Pudozhgorsky settlement). According to the available demographic data, there were 700000 persons in the Karelian republic in 1968 year, then the number of cases of the lung cancer (Qp) can make 60.0 cases in a year.

DOSE LOADS ON THE POPULATION OF COLA PENINSULA

The Murmansk region annual doze from a radon in air of premises is about 2,2 mSv. It should clearly see, that this value is enough rough valuation, owing to limited number of surveyed towns. On separate settlements a range of fluctuations from this average value is about 2 times decrease or increase (from 1.2 up to 3.7 mSv in a year). The excess number of the lung cancers for Murmansk region can be Qp= 27.8 cases per year.

CONCLUSIONS

1. The radiation conditions, formed by the expense of receipt of radon and its daughters products in the air of living accommodations and buildings of various purpose in settlements of Karelia and Cola peninsula, should be regarded as intense and creating for the population and its critical groups (in particular, the children) potential opportunity of irradiation higher then established normative limits.
2. Average annual ED of irradiation of population, stipulated by receipt from air of radon and its daughters, is for the inhabitants of various settlements 0.6 - 25 mSv in a year. These levels of irradiation change within the limits of 6- 120 % from value, considered by international organizations as not requiring fulfillment's of protective measures. At the same time, in the process of settlements survey separate houses are revealed, where according to results of measurements, radon concentrations 2 - 3 times exceeding national normative values are registered. The number of such houses in different settlements makes up 13 % from total number of buildings. In these cases protective measures should be accepted.
3. Annual mean ED value in surveyed towns of Karelia is 4.5 mSv, and for Cola peninsula - 2.2 mSv.
4. According to prognostic account, the possible lung cancer number, caused by the inhaling radon daughters, for the inhabitants of surveyed regions of Karelia, , makes up 9 persons on 100 000 of the population in a year, and for the inhabitants of Cola peninsula - 4 persons.

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RADIATION CONDITIONS FROM RADON DWELLINGS OF VARIOUS PURPOSE

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INTRODUCTION

A specific feature of the mechanism of pollution of air environment is continuous flow of a radon in volume of a dwelling and, owing to it decay , accumulation of affiliated products, mainly determining a dose load on the person.

The intensity of radon allocation in volume of a dwelling depends basically on geophysical properties of spread under rocks, building material and time of air exchange .This determines specific character of formation of a radiating conditions in residential and industrial dwellings.

SOURCES AND FACTORS, INFLUENCING TO INTENSITY OF ALLOCATION OF A RADON IN VOLUME OF DWELLINGS

Main sources of allocation of a radon in volume of dwellings are:

1. Massif of rocks under buildings;
2. Building materials;
3. Technical and drinking water, natural gas.

The quantity of a radon, arriving from these sources, depends on a number of the reasons, determined by geological and geophysical properties of spread under rocks and by construction and architect features of dwellings and buildings.

The allocation of a radon from a massif of spread under rocks.

Process of receipt of a radon from a massif of rocks consists of two consecutive steps: the first step is allocation of a radon in emptiness of rocks (pores, cracks) thanking the phenomenon of effect Ra-226 decay, second - distribution of allocated in thepores radon under the laws of gas diffusion or convection.

The theory of distribution of a radon in pores was in detail developed in (1,2), where for various cases of radon distribution the theoretical formulas were deduced.

In general the differential equation of distribution of volumetric activity of radon at steady conditions will have a form:

$$K\frac{\partial^2 C}{\partial^2 n} - V\frac{\partial C}{\partial n} + q\rho k - \lambda C\varepsilon = 0 \qquad (1)$$

where: K - coefficient of diffusion;
C - quantity of a radon in 1 cm³ of pore air;
V - convection velocity;
q - quantity of a radon, formed for 1 s in 1 g of rocks;
ρ - volumetric weight of rocks;
k - emanation coefficient;
λ- constant of radon decay;
ε -porosity of rocks;
t - time.

As an example we shall consider the elementary case and, in particular, diffusion distribution of a radon in infinite, homogenious layer, leaving on a surface. The change of volumetric activity of radon with depth is described by the equation:

$$C_{Rn} = \frac{q\rho k}{\lambda} * \left(1 - H\sqrt{\frac{\lambda}{K}}\right) \qquad (2)$$

And flow of radon from 1 cm² of a rock surface (exhalation)

$$E = K\frac{\partial C}{\partial x} = \frac{q\rho k}{\lambda}\sqrt{\lambda K} \qquad (3)$$

THE ACCUMULATION OF SHORT LIVED AFFILIATED PRODUCTS OF RADON IN AIR OF DWELLINGS OF VARIOUS PURPOSE

Accumulation of products of radon decay in a general case can be expressed by the following equation:

$$N_n = N_{Rn} * \lambda_1\lambda_2....\lambda_{n-1} * (\frac{\exp(-\lambda_1 t)}{(\lambda_2-\lambda_1)(\lambda_3-\lambda_1)*...*(\lambda_{n-1}-\lambda_1)} + \tag{4}$$

$$+....+\frac{\exp(-\lambda_n t)}{(\lambda_1-\lambda_n)(\lambda_2-\lambda_n)*...*(\lambda_{n-1}-\lambda_n)}\}$$

As the initial volumetric activity of decay products , and also the correlations between them can be different, we shall consider character of accumulation of volumetric activity of radon decay products and their potential energy directly from a radon, i.e. when the initial volumetric activity RaA, RaB and RaC is equal to zero. In this case the equation (4) became:

$$N_a = \lambda_{Rn}\frac{N_{Rn} * (1-\exp(\lambda_a t)}{\lambda_a} \tag{5}$$

$$N_b = \frac{\lambda_{Rn} N_{Rn} * \{\lambda_a(1-\exp(\lambda_b t) - \lambda_b(1-\exp(\lambda_a t)\}}{\lambda_b(\lambda_a-\lambda_b)} \tag{6}$$

$$N_c = \lambda_{Rn} N_{Rn}(\frac{1}{\lambda_c} - \frac{\lambda_b \exp(-\lambda_a t)}{(\lambda_a-\lambda_c)(\lambda_a-\lambda_b)} - \frac{\lambda_a \exp(-\lambda_b t)}{(\lambda_c-\lambda_b)(\lambda_a-\lambda_b)} + \frac{\lambda_a\lambda_b \exp(-\lambda_c t)}{\lambda_c(\lambda_a-\lambda_c)(\lambda_c-\lambda_b)} \tag{7}$$

The quantity of accumulated radon decay products , expressed in potential energy units of alpha-decay, is equal in MeV l^{-1}

$$E = E_a + E_b + E_c = 13{,}68 * N_a + 7{,}68 * N_b + 7{,}68 * N_c \tag{8}$$

where N_a, N_b and N_c are calculated under the formulas (5, 6, 7).
The change of potential energy in time in relative units will be equal

$$\phi = \frac{E}{E_{eqv}} = \frac{13{,}68 * N_a + 7{,}68 * N_b + 7{,}68 N_c}{E_{eqv}} \tag{9}$$

The analysis of decay curve "ϕ" shows, that with rather good approximation the change in ϕ can be expressed by the following dependence:

$$\phi = 1 - \exp(-\lambda_e * t) \tag{10}$$

Where λ_e - empirical constant, s^{-1};
t - time of a radon presence of in volume of a dwelling, s.
Then the equation of accumulation of potential energy or equivalent equilibrium volumetric activity (EEVA) of a radon in a dwelling will have a form:

$$E = C_{Rn} * [1 - \exp(-\lambda_e * t)] \tag{11}$$

In dwellings, in which radon continually arrives, the change of volumetric activity is recorded by an equation:

$$v\frac{\partial C_{Rn}}{\partial w} + \lambda_{Rn} C_{Rn} = Q_w \qquad (12)$$

where C_{Rn} - volumetric activity of a radon in a point "w" of a dwelling, Bq m^{-3};
v - speed of air movement in a dwelling, m / s;
λ_{Rn} - constant of radon decay , s^{-1};
Qw - volumetric density of radon flow, Bq $m^{-3} s^{-1}$.

Using a boundary condition $C_{Rn} = C_{Rn}o$ at x=0, the solution of equation (12) at constant ventilation mode can be recorded in the following form

$$C_{Rn} = C_{Rn}o * \exp(-\lambda_{Rn} * t) + (Qw / \lambda_{Rn}) * [1 - \exp(-\lambda_{Rn} * t) \qquad (13)$$

Taking into account, that the time of air exchange in a dwelling cannot be more than two hours, $\lambda_{Rn} * t << 1$, the equation 13 can be recorded with sufficient for practical purposes accuracy:

$$C_{Rn} = C_{Rn}o + Qw * t \qquad (14)$$

Thus, the equation of accumulation of potential energy can be submitted as

$$E = C_{Rn}o * [1 - \exp(-\lambda_{Rn} * t)] + Qw * t * [1 - \exp(-0,5 * \lambda_{Rn} * t)] \qquad (15)$$

PARAMETERS, DESCRIBING A RADIATION CONDITIONS UNDER THE FACTOR "RADON" IN A DWELLING

In the basis of acceptance of the decisions is incorporated criterion not excess mean year EER of a radon equal 100 Bq m^{-3} for new housing accommodation and 200 Bq m^{-3} for existing. For valuation of these values, long-term measurements of a radon by extent till 6 months (measurements in current two seasons) and subsequent recalculation on EEVA of radon with accepted shift of balance equal 0,5 are recommended. I shall not be go in critics of this rule, but I offer, as alternate, following approach to valuation and forecast of a radiating conditions in dwellings of various purpose at different stages of its development from designing before delivery it in operation and while in service.

CRITERION of DECISIONS ACCEPTANCE

Stage of designing and organization of the use of land.

Parameter	Radon safe	Additional researches needed	Radon remedial measures needed
1. Content Ra-226 in soil, Bq kg^{-1}	< 20	$20 < A_{Rn} < 50$	>50
2. Diffusion coefficient, $cm^2 s^{-1}$	> 0,01	> 0,002	> 0,002
3. Volumetric Rn activity kBq m^{-3}	< 10	$10 < C_{Rn} < 40$	> 40
4. The relation C_{Rn} (1 m)/C_{Rn} (0,1 m)	1,2	1,1-1,0	>1,05
5. Exhalation, Bq $m^{-2} h^{-1}$	< 75	75 - 150	> 150

The exhalation of radon from a surface can be determined experimentally or by calculation using gradient of volumetric activity in soil pore air. Unfortunately, the estimates are rather complex and possible only on computers. The account of an exhalation of a radon is carried out under the formula (3), and we receive factor of diffusion from the solution of equation:

$$K = \frac{H1 * H2 * \lambda_{Rn}}{2 * \ln(1 - B(1 - \exp(-H2 * \sqrt{\frac{\lambda_{Rn}}{K}})))} \qquad (16)$$

Stage of a reception of a new living accommodation and valuation of existing.

The accumulation EEVA of radon in a dwelling is well described by equation (15), from which it is clear, that determining parameter is specific volumetric flow of a radon in a dwelling. Taking into

MEASUREMENTS OF Rn - 222 CONCENTRATIONS IN DWELLINGS IN AN AREA OF HIGH NATURAL RADIOACTIVITY

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ABSTRACT

An investigation was conducted on the levels of Rn- 222 concentrations in dwellings in Köprübaşı where there exits large uranium ore deposits. Considering the importance of the region in radioactive environmental programme of our Institute the radon concentrations of dwellings were measured by using LR-115 cellulose nitrate films as nuclear track detectors. The results have indicated that the range of radon levels in this area is 20-340 Bq/m^3 . The highest radon concentrations are observed in the stone buildings which were constructed with the material obtained from that area.

INTRODUCTION

Long term measurements by using LR-115 nuclear track detectors have been carried out to investigate the distribution of radon concentrations in dwellings in Köprübaşı located in vicinity of uranium ore deposits (1). One can think it might be a possibility of healt risk even for long exposure to low irradiation doses. This is fundemental when some population group are living in an area of high level of natural radioactivity. It follows that more information about radon concentration level in the atmosphere of the dwellings in Köprübaşı seems to be necassary for considering the importance of the region. Taking this point into account the radon concentration has been measured in about hundred houses of the region in radioactive environmental programme of our Institute.

EXPERIMENTAL AND RESULTS

Kodak- Pathe LR -115 type 2 cellulose nitrate film is used as SSNTD for the measurements of radon in dwellings in the vicinity of uranium deposits. These small passive and inexpensive detectors are convenient for making the long term measurements of alpha activity of indoor radon. LR- 115 type 2 film in this study was cut into 1.2* 1.2 cm pieces, numbered and attached with the tape on the bottom of a numbered plastic cup. These sampling cups were hung to living quarters of dwelling. After exposing for a month the films were collected and processed chemically. Etching conditions in this process are % 10 NaOH at 60 °C for 2 hours (2) using 25 ml etchant for each film (3) in a thermo-controlled water bath. The number of registered alpha tracks per film area is manually counted using an optical microscope. To avoid poor counting statistics on each film normally an area of 1 cm^2 is counted. After counting, the alpha track density was converted to units of Bq / m^3 radon. In our laboratory calibration 1 track/ mm^2 corresponds to an exposure of about 31.25 Bq/m^3

radon and its daughters during a month (4). The distribution of radon levels in this region is shown in figure 1.

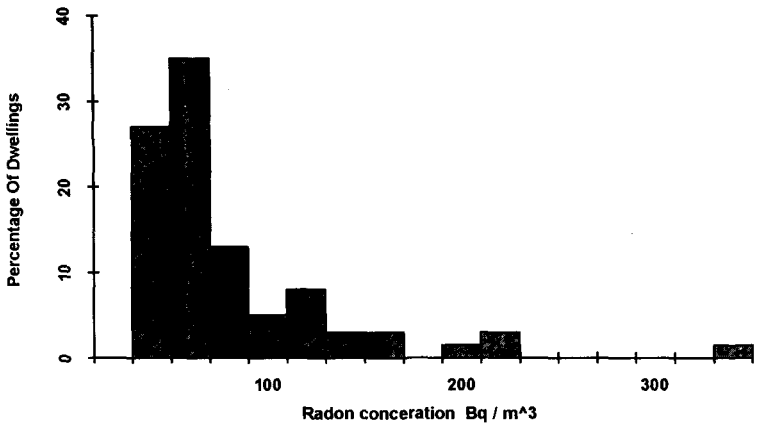

Figure 1. Radon consentration in Bq/m^3 according to persentage of dwellings

The results have indicated that the range of radon levels in this area is 20-340 Bq/m^3. The highest radon concentrations are observed in the stone buildings which were constructed with the material obtained from that area.

CONCLUSION

The survey of radon levels in Köprübaşı located in vicinity of uranium ore deposits indicated that a large persentage of the houses has Rn-222 concentrations below 100 Bq/m^3 as a small percentage of the dwellings is above 200 Bq/m^3. But these values are higher the correspoinding values for the dwellings in İzmir (4) in radioactive environmental programme of our Institute.

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RADON CONCENTRATIONS IN TAIPEI METROPOLITAN RAILWAY STATION

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INTRODUCTION

For routine airborne radon monitoring, including use in field conditions, the technique based on electret ion chamber technology seems to be the most suitable choice in many applications (1). However, this simple and relatively inexpensive method has some specific drawbacks: poorer reproducibility at lower radon concentrations, some uncertainty in the use of manufacturer suggested gamma correction factors, and limited reusability (2).

A modified electret ion chamber method has been proposed (2), but it is mainly for water borne radon measurement. Therefore, we still applied the simple method recommended by the manufacturer to survey radon concentrations in Taipei Railway Station.

MATERIAL AND METHOD

Taipei City is the capital of Taiwan. Its metropolitan railway station was built with concrete elements, two stories underground, and equipped with mechanical ventilation system. The daily number of passengers is about 100,000, and during long weekend it may reach 300,000 per day.

Three types of instrument were used for a period of one-year survey between June 1994 and June 1995. For the radon survey the electret ion chamber was used with a commercial name of E-PERM. For the survey of radon progeny the electret ion chamber equipped with an air pump and filter system was used with a commercial name of E-RPISU. Since both instruments mentioned above cannot detect the instantaneous fluctuation in radon concentrations, a third instrument was added to the one-year monitoring program. It is a pulse type ion chamber equipped with temperature, barometric and humidity gauges (3). The ambient gamma exposure was measured with thermoluminescent dosimeters $CaSO_4$:Dy. All calibrations were performed inside a radon chamber installed at the Taiwan Radiation Monitoring Center.

RESULTS AND DISCUSSION

Radon concentrations at the platform which is located at second basement below the ground floor are shown in Fig.1 while those at the passengers' waiting area which is located at the first basement below ground floor are shown in Fig.2.

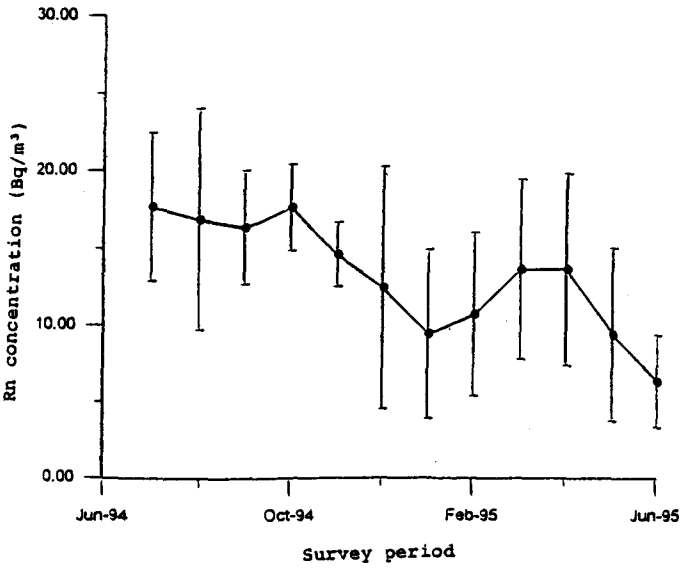

Fig.1 Yearly survey of ^{222}Rn concentrations at the platform.

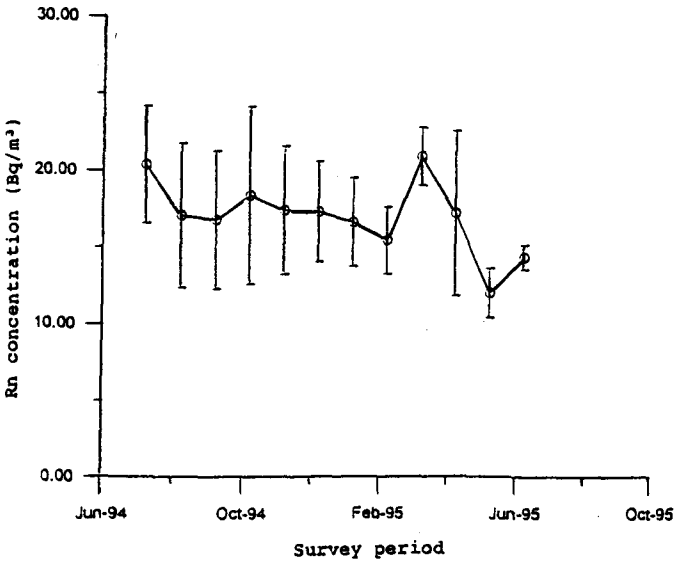

Fig.2 Yearly survey of ^{222}Rn concentrations at the waiting area.

The decrease in radon concentrations as a result of ventilation improvement can be seen in both Figs. 1 and 2. The equilibrium factors varied from 0.29 in hot and humid summer to 0.39 in cold winter with a yearly average of 0.35. The gamma background was rather low with an average energy of about 0.23 MeV.

The annual effective dose received by the Taipei Railway Station personnel is estimated to be about 87 uSv while the annual equivalent dose to lung is about 71 uSv.

ACKNOWLEDGMENTS

Financial support from the Atomic Energy Council of the Executive Yuan (Cabinet) is much appreciated.

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BUILDING AND ENVIRONMENTAL FACTORS ASSOCIATED WITH ELEVATED RADON LEVELS IN KINDERGARTENS IN SLOVENIA

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INTRODUCTION

In the winter periods of 1990/91 and 1991/92, the 730 Slovenian kindergartens and play schools that together care for more than 66,000 children, instantaneous indoor radon concentrations were measured using alpha scintillation cells. In 528 (72%) of the kindergartens and play schools radon concentrations below 100 Bqm^{-3} were found. In 16 buildings (2.2%) the concentrations exceeded 800 Bqm^{-3} (1).

Parameters that can affect radon concentrations are geological location, structural characteristics, climate, building material, and the occupancy patterns within these buildings. However, the most important parameter affecting radon concentrations is source. Without source material, there can be no radon. In certain areas of Slovenia, uranium, radon's source material, is abundant (2).

Each building possesses its own dominant parameters for elevated radon concentrations. Sampling and evaluation provide the best opportunity for advising mitigation measures in buildings. New buildings can be constructed utilising radon resistant techniques. Possible building sites can be evaluated with soil testing for radon, and evaluation the porosity of the soil and drainage from the site.

METHODOLOGY

For radon measurements alpha scintillation technique (3) was applied. Grab samples were collected from one groundfloor playroom, closed overnight, at each building. At the same time a short questionnaire concerning construction characteristics and working regime was filled.

In an effort to further define the extent and possible centers of elevated radon concentrations Slovenia was divided into nine units (regions) on the basis of telephone codes and four geographic units for evaluation. The nine regions are: Ljubljana, Maribor, Celje, Kranj, Nova Gorica, Koper, Postojna, Novo mesto and Murska Sobota (1). The four geographic units are: Alps, Fore-Alps, Dinarides and Pannonian basin (4). For each unit the mean and median value was calculated. All analyses were performed using SAS (5) and SigmaStat (6) software packages.

RESULTS AND DISCUSSION

The distribution of radon concentrations in Slovene kindergartens is lognormal with a geometric mean of 58 Bqm^{-3} and geometric standard deviation of 2.5. Data from previous studies indicate that grab samples collected with alpha scintillation cells may over estimate continuos radon levels by as much as a factor of two (7), therefore the information presented here should be considered as the "worst case scenario".

The mean and median radon concentrations for four geographic units are plotted in Figure 1. The Alps (the Julian and Savinja Alps, Karavanke and Pohorje) occupy the northern part of Slovenia. The Julian and Savinja Alps are built predominantly of carbonate rocks, the Karavanken and Pohorje from metamorphic and volcanic rocks. The Fore-Alps occupy the

central part of Slovenia and are mostly built of clastic sediments. The Dinarides, a well known as "dinaric karst" region, represent southern parts of the country and are predominantly built of carbonates and flysch deposites. The Pannonian basin occupies the flat-lands and hilly regions in eastern Slovenia, built of alluvial deposit as sand, gravel and clay (4). Statistically significant differences were found between groups. The Dinaric unit was statistically different from both, the Pannonian basin and the Alpine unit, but did not vary statistically from Fore-Alps. A statistically significant difference existed between the Fore-Alps and the Pannonian basin, but no statistical difference existed between the Fore-Alps and Alps. There was no statistical difference between the Alps and the Pannonian basin.

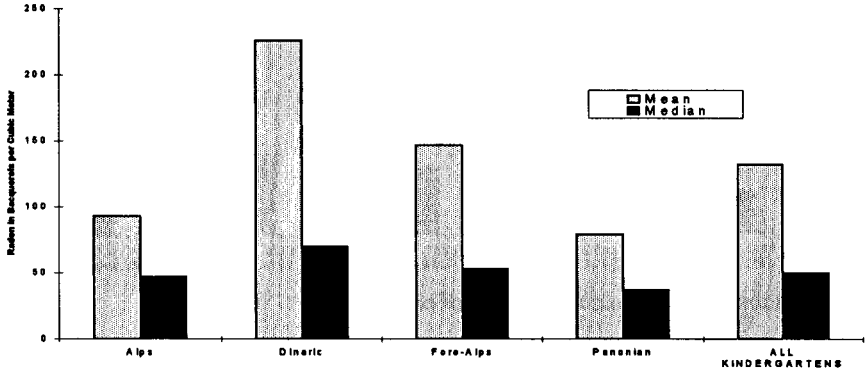

Figure 1. Mean and median radon levels in Slovene kindergartens by geographical regions

Radon concentrations in kindergartens within eight regions on the basis of telephone codes were below 200 Bqm^{-3} with an exception in Postojna region (the Dinarides), with the highest radon exposure risk.

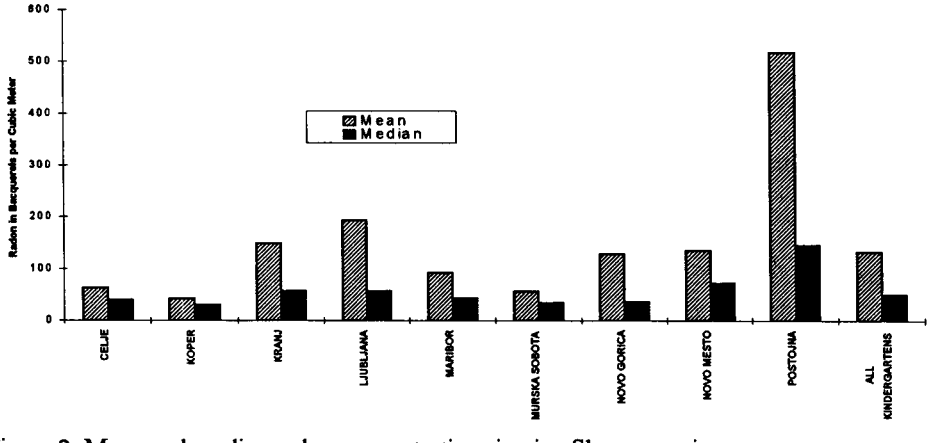

Figure 2. Mean and median radon concentrations in nine Slovene regions

The relationship between radon concentration and age of the building was looked but not found (Figure 3). The majority of the kindergartens are 20-30 years old single story buildings, built of bricks. In some older buildings, built of stone, higher radon levels were found and the mean reason was bad building construction. In years around 1960 fly ash bricks

were used as building material, but not extremely high radon levels in these buildings were found.

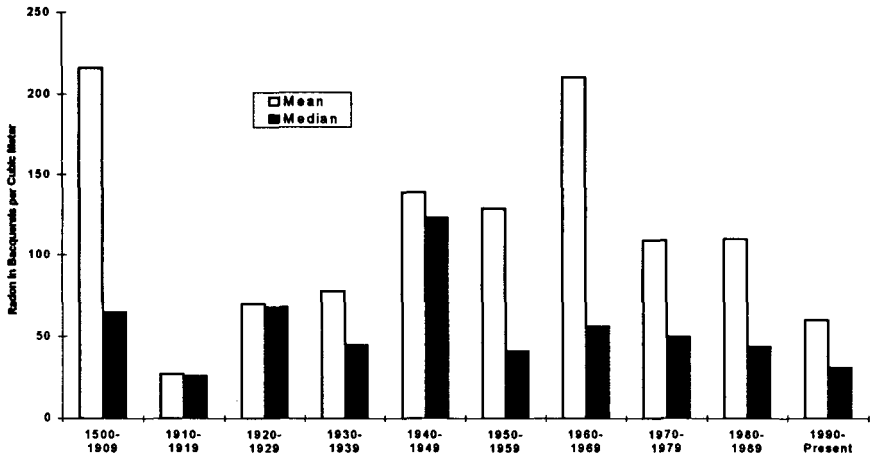

Figure 3. Mean radon levels in buildings by year of construction

CONCLUSIONS

What is important is that radon levels in Slovene kindergartens and play schools were evaluated using consistent techniques and conducted with trained personnel, so that proper evaluations could be made. Most important, all kindergartens and play schools were sampled for radon. Although geographic region, age, construction type and construction material can affect radon levels, sampling is the only technique which will determine actual radon levels within a particular building. Data from this study indicates that 5085 (7.68%) children attend a kindergartens and play school with radon levels which exceeded 400 Bqm^{-3}, the proposed Slovene Radon Action Level. 12,719 (19.2%) of kindergarten and play school children attended a kindergarten in which the radon level exceeded 150 Bqm^{-3}, the current US-EPA radon action level. In 16 buildings radon concentrations exceeded 800 Bqm^{-3}, affecting 1719 (2.59%) children (1).

Data from this study has been used to determine both need and method of mitigation in the buildings affected most by elevated radon levels. Mitigation has begun or been completed on five of the kindergartens effectively lowering radon concentrations to more acceptable levels.

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MODELIZATION OF THE RADON EMANATION FROM NATURAL SOURCES IN THE SOIL

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Radon is a natural radioactive gas from the uranium family which is the main component of the annual natural radiation exposure of the population. It has been shown in several epidemiological studies that this gas, when inhaled, provokes lung cancer.
Part of the radon formed in the earth's crust can migrate to the atmosphere through cracks and fissures by transport mechanisms (diffusion and fluid convection). Different parameters such as the localisation of radon sources, the porosity, the humidity of the soil can influence this migration and therefore the radon emanation in the atmosphere.
To evaluate the radon emanation and hence the risk to populations, we have adapted an original mathematical model based on the method of distribute parcels (L.Hlou, These d'état, Faculté des Sciences, Kénitra, MAROC, 1994). This allows us to follow the migration, in time and space, of a quantity of radon produced in a unit volume as a function of the geological, morphological and structural characteristics of the site studied. Knowing the petrographic and pedologic parameters enables us to calculate the radon concentration in all points inside the soil of the site as well as the radon emanation in the atmosphere. It is therefore possible to calculate the radiological risk for populations brought to live on the site studied.
Different applications of this model have been realised in Morocco and in France to demonstrate its efficiency.

SOME INFLUENCING PARAMETERS OF THE RADON AND THORON DAUGHTERS CONCENTRATIONS IN DWELLINGS

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INTRODUCTION

Man spents at least 60% of his life indoor and this fact imposes a careful investigation of his habitual environment. Indoor exposure rises up to 50% of the total radiation exposure of the public to natural sources (1).

It is also well known that inhalation of radon is the principal mechanism of human exposure to ionising radiation and that the exposure to high levels of radon short-lived decay products (radon daughters) is undesirable since it has been shown to cause lung cancer.

In Romania, during the 80's, several industrial wastes with high natural radioactivity have been used as building materials. At the same time, the energetic problems of those years obliged the population to a very drastically dwelling isolation in order to preserve the heat. These factors, in addition with other influencing parameters have been investigated.

METHOD

Measurements of indoor and outdoor radon and thoron daughters concentrations were performed in 119 Romanian dwellings, by air samping on a membrane filter, followed by repeated gross-alpha countings. Several constructive and physical parameters have been recorded: construction type (detached houses, block of flats), constructive solution design (brick work, prefabricated reinforced concrete), floor and room position, building materials used, heating systems, season, indoor and outdoor temperature, pressure and relative humidity, direction and speed of wind. Knowing the radon and thoron daughters concentrations, the Potential Alpha Energy Concentration (PAEC), the Equilibrium Equivalent Concentration (EEC), the equilibrium factor (F) and the effective dose for adults (ED) have been calculated.

RESULTS

The results obtained are presented in Tables 1,2 and 3.

Table 1. Dependence of the indoor Radon and Thoron daughter concentrations on constructive type, solutions and building materials

Constructive type	Constructive solution	Building material	EEC (Rn) Bq/mc		F	EEC (Tn) Bq/mc	
			g	s		g	s
Detached houses	Brick work	Brick	29.0	4.1	0.68	1.17	0.30
Block of flats	P.R.C.	Concrete	21.0	4.6	0.64	1.40	0.30
	C.R.C.	Concrete	6.3	3.9	0.50	0.80	0.31
	Brick work	Red brick	2.1	3.9	0.46	0.93	0.30

P.R.C.: Prefabricated Reinforced Concrete; C.R.C.: Casted Reinforced Concrete;
g: geometric standard deviation

Table 2. Dependence on the season; gearing of the the flat: SW

Month	Wind speed	Wind dir.	Temp. (°C)		Press (mm Hg)		Humidity%		Indoor Conc.(Bq/mc)			F
			ind	out	ind	out	ind	out	Ra A	EEC (Rn)	Th B	ind
July	0.54	NE--SW	25	25	751	752	80	85	159,6	125,9	4,3	0,72
Nov.	0.30	N-S	15	7	750	752	80	77	50,4	22,6	1,4	0,52

Table 3. Dependence on floors and indoor humidity

Floor	Wind speed	dir.	Temp (C) ind	out	Press. (mmHg) ind	out	Hum (%) ind	out	Flat gea-ring	Indoor Conc. (Bq/mc) Ra A	EEC (Rn)	ThB	F ind
Ground	1.01	SE-NW	14	7	755	760	93	53	NE	104.4	62.2	1.4	0.54
Ground	0.30	NE-SW	14	6	759	760	60	53	SW	18.7	15.5	1.3	0.76
Ground	0.30	N-S	15	7	750	754	80	77	SW	50.4	29.8	1.4	0.52
2nd floor	0.30	SE-NW	14	9	760	759	85	82	SW	60.3	31.6	0.6	0.48
5th floor	0.67	SW-NE	14	12	760	749	60	56	NE	49.9	43.7	1.1	0.81
6th floor	0.20	NW-SE	16	12	760	760	80	77	SW	43.0	37.8	1.4	0.80

CONCLUSIONS

EEC (Rn) experimental values ranged from 3 to 130 Bq.m^3. The main determined influencing parameters were: ventilation rate, type of building material and indoor relative humidity. It was also pointed out that the contribution of the thoron daughters to the total ED is about 20% (2).

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Radon measurements in the cave Vilenica P11-29

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Abstract
The cave Vilenica is about 1300 m long and 180 m deep. One third of the cave is open for the visitors. Radon and its daughter concentrations and also temperature and pressure have been measured at different times of the year and at different locations in the cave in order to estimate doses to visitors and guides. Radon concentrations ranging from few 100 Bqm^{-3} in the winter up to few 1000 Bqm^{-3} in the summer months. The dose rate for visitors and guides is 10 to 40 μSv/h, depends from the radon concentration.

1. Introduction

The Vilenica cave is situated near the town Sežana in the middle of the typical karst and it's probably the oldest show cave in Europe. First tourists visited the cave in the year 1633. Total length of the galleries is 1300 m. The part arranged for tourists is one gallery 450 m long and 94 m deep. Other part of the cave is accessible only to cave explorers with special equipment.

The Vilenica cave is one of caves, where a speleotherapy (1,2) will start in near future. Speleotherapy is a special kind of climatotherapy and is being used as an additional treatment for curing bronchial and asthmatic diseases of children and adults in last 40 years in many countries in Europe.

The objectives of this measurements are (i) to determine the physical parameters of the cave microclimate, important for the speleotherapy and (ii) to determine radon and progeny concentrations for dose estimation.

2. Methods

In the period from January to December 1995 monthly measurements of different parameters on the route for visitors have been investigated. Relative humidity, temperature and air flow were measured by standard digital devices. Concentrations of positive and negative ions were measured by ionometer Shomandl with the accuracy of about 10 %. Concentrations of different gases were measured by Dröger tubes. Radon and progeny concentrations were performed by radon gas monitor RGA-40 and working level monitor WLM-30 from Scintrex, Canada. Radon concentrations were measured also by track each detectors and charcoal canisters. Duration of measurements was three days for radon and progeny, measured by devices or charcoal canisters and one month or more by track detectors. All measurements were made in the middle of the cave at one place, "near the water", except radon concentrations, which were measured at 9 different places.

The visiting part of the cave has two different temperature regimes, first part about 200 m long with stronger influence of meteorological parameters from outside, where temperature changes throughout the year for few degree Celsius and second part about 250 m long with nearly the same temperature of 10 °C in all periods of the year. From September to December air velocities and temperature at 5 different places were measured, two in first and three in second part of the cave. The volume of the cave is about 200000 m^3, of the second part about 120000 m^3.

3. Results

Measurements of microclimatic parameters showed us no content of ozone, sulphur dioxide, nitrate, fluoride and carbon monoxide in cave. Only carbon dioxide was detected, its concentrations were in the range of 0.05 vol % to 0.4 vol %. Temperature in the first part of the cave changed from 2 °C to 10 °C, at the point, where second part started (temperature point) was between 7 °C and 10 °C, and at the lowest part, accessible for visitors, varied from 10,5 °C to 12 °C. Air pressure was changing in a measuring period for few millibar (3-7 millibar), only in December first decreased for 6 millibar and then increased for 15 millibar. Temperature on all places slowly increased with hight. On the bottom of the Hall of the Faires is a small entrance (1m^2) in a Fabris cave. Air flow at this place is obout 30 m^3/h. Only in December changes in air velocities from 0.01- 0.08 ms^{-1} were found. It was like a breathing with a period of 30 s.

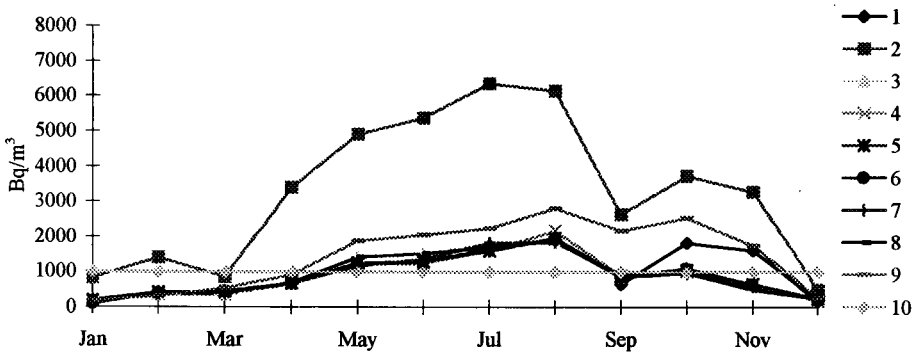
In summer months we can suppose air velocities of 0.01 ms^{-1} or less on all places in the cave. In December was air velocity at the place "temperature point" in the range of 0.03-0.1 ms^{-1} and at the entrance between 0.2-0.3 ms^{-1}. If we assume effective cross section at the temperature point of 50 m^2, air flow in summer was 2000 m^3h^{-1} and ventilation rate was less then 0.01 h^{-1}. Calculated air flow and ventilation rate in December were between 6000-10000 m^3h^{-1} and 0.03-0.1 h^{-1}, respectively.

The effective cross section at the entrance is about 20 m^2. Air flow and ventilation rate in summer were the same like at the temperature point, in December air flow was between 10000-15000 m^3h^{-1} and ventilation rate in range form 0.05-0.1 h^{-1}. If we take into account equation from Wilkening and Watkins (3), $Q = \lambda V(C_{max}-C)/C$, and insert values

for radon concentrations, we obtain for air flow in December at the temperature point and at the entrance the value of 9000 m^3h^{-1} and 11000 m^3h^{-1}, respectively. Results are in good agreement with measured values of air flow.

Concentrations of positive ions were changing from 3000-11000 per cm^3 and negative ions from 3000-9000 per cm^3. The difference between concentrations of positive and negative ions was higher by higher radon concentrations.

Radon concentrations at different places in the cave were nearly the same in different measuring periods (Fig. 1), except at the place The dance Hall. This is a very big hall on the entrance in the cave. We didn't measure radon concentration in the hall but in small crack (about 30 cm^2). We suppose the crack lids to a big unknown hall with higher radon concentrations.



Legend:

1 The entrance
2 The dance hall
3 Under the Red hall
4 Red hall
5 Near the water
6 Above the hall of the Faires
7 Before the hall of the Faires
8 The hall of the Faires
9 On the bottom of the hall of the Faires
10 Annual average

Fig. 1. Radon concentrations in the Vilenica cave in the period Jan - Dec 1995

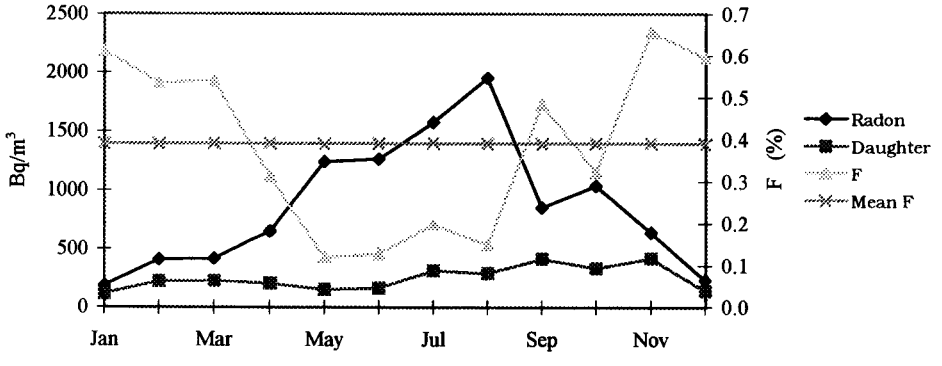


Fig. 2. Monthly radon and daughter concentrations and equilibrium factor in Vilenica cave at the place near the water

Higher values were measured on the bottom of the Hall of the Faires, where radon cames through the small entrance from the Fabris cave. This is a big cave with higher content of clay and higher temperature. This are main reasons for higher radon concentrations on the bottom of the cave, accessible for visitors, which has no clay only washed limestone. Radon concentrations at these place were 20-50 % higher then on other measuring places, depends from the period of the year. We suppose this is a main radon source.

Radon concentrations in cave were higher in summer then in winter, mean value was 980 Bqm^{-3}. The lowest values were in December and January. In first and last months of the year the wetter was changing, measured radon concentrations at different times could be different form presented values. In September it was very cold wetter in the time of measurements (less than 10 °C), what is a reason for lower radon concentrations. Radon concentrations in cave increased from January to August, and then decreased from September to December, like mean outdoor temperature. Different temperature gradient between the cave air and outdoor air produced higher or lower ventilation rate.

On the Figure 2 radon and daughter concentrations at the place "near the water" are presented. Radon progeny concentrations were nearly the same in the first half of the year, then increased about two times from July to November and then sharply decreased in December. Equilibrium factor between radon and daughter was about 4 times lower in summer then in winter months.

The measurements which have been made in Postojna cave (4) showed very low aerosol concentrations, between 2000-4000 per cm^3. The activity median aerodynamic diameter AMAD was 250 nm (σ_g=1.6). A quite high activity fraction (30%) was found in the nucleation size range between 4 and 80 nm. Unattached fraction f_p was between 10 and 15 %.

We didn't measure aerosol size distribution and unattached fraction in different periods of the year, but we suppose that in winter time fresh air brings more aerosols from outside into the cave. Because of higher ventilation rate in winter time the air in the cave is mixing faster and more radon decay products can claimed with aerosols what produces higher equilibrium between radon and progeny. This explains differences in equilibrium between radon and daughters; we will prove it with measurements next year.

4. Dose estimates

Our dose calculation based on three different models, ICRP 50 (5,6), Jacobi-Eisfeld (J-E) and James-Birchall (J-B). For AMAD and unattached fraction we take into account values from Postojna cave, fp=0.1 and AMAD 0.2 μm. The dose rate is 10 to 40 μSv/h, depends from the radon concentration. If we suppose 1000 to 2000 working hours per year and mean progeny concentration of 250 mWL, annual dose is about 20 mSv from J-E and 30 mSv from J-B model. If we suppose new dose convention 5 mSv/WLM, annual dose lies between 10-15 mSv.

In future we will measure unattached fraction and aerosol size distribution of radon daughters, for more precisely dose estimation and for understanding the differences in equilibrium factor.

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RADON SURVEY IN KUWAIT HOUSES

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INTRODUCTION

Measurement of indoor radon levels in Kuwait houses is part of a project undertaken to estimate the average annual effective doses from different sources for Kuwait population. It is known that airborne ^{222}Rn in houses is a major contributor to this value (1). Among common sources of radon in dwellings, geological factors may play important role. Kuwait is situated on the northern coastal area of the Arabian Gulf. From geological point of view its surface soil is relatively uniform - desert type with prevailing smooth sand sheets with relatively low uranium and radium concentrations (2). The first pilot study of indoor Rn concentration showed surprisingly relative high concentrations with mean values above 40 $Bq \cdot m^{-3}$ (3). In this study results of a survey of indoor radon in 150 houses in different parts of Kuwait over summer period are presented. Among many methods developed for indoor radon measurement, the method using vials with activated charcoal adsorbant and liquid scintillation counting proved to be very reliable and especially useful for large scale surveillance (4) and therefore this method has been applied in this survey.

METHODS

Commercially available plastic liquid scintillation vials with charcoal PICORAD were purchased from Packard. In each household two vials were exposed in the slepping rooms for a standard time of 48 hr. The returned vials (within 1 to 7 days after the end of exposition) were eluted with 10 ml of liquid scintillation cocktail (5 g/1 PPO; 0.3 g/l POPOP in xylene plus 10 % of methanol) and after at least 8 hrs (to reach the maximal elution of Rn from charcoal to a scintillator) were measured in liquid scintillation counter Packard 2550 TR/AB for one hour in the channel 25-900 keV. Concentrations of Rn were calculated from the following formula:

$$C_{Rn} = K_1 \, I \, [1-\exp(-k_2 \, t_e)]^{-1} \, [\exp(-\lambda \, t_d)]^{-1}$$

where: K_1 - is the normalization constant that converts net counts per minute I to $Bq \cdot m^{-3}$,
K_2 - absorption constant,
λ - radioactive decay constant for ^{222}Rn,
t_e - exposure time in hrs,
t_d - delay time, which elapsed from the end of exposition to the start of counting.

The K_1 and K_2 constants were experimentally determined by exposure of vials for different times in the radon experimental chamber with constant radon concentrations, determined by scintillation detector REX-013. The: Lower Limit of Detection" (LLD), i.e. the lowest measurable concentration for used method with the confidence level α=0.95 has been achieved as 1.4 $Bq \cdot m^{-3}$ for the standard duration of exposure 48 hours and delay period 24 hours.

The arithmetic mean from two vials was taken for calculation of Rn concentrations. Usually deviation between two vials was below 30 % of the mean value, showing good reproducibility.

All experimental results for 150 houses were divided to the three groups: those for the ground and first floors, for higher floors (above first floor) and total results without distinction of the floors.

It has been found that obtained results very well correspond to lognormal distribution with parameters given in the Table 1.

Table 1. Parameters of lognormal distribution of ^{222}Rn concentrations in Kuwait houses.

Parameter of Study	Value of Parameter		
	Ground Floor & First Floor Above Ground	Higher Floors	All Flats Together
Number of Flats	89	61	150
Arithmetic Mean ($Bq \cdot m^{-3}$)	13.9	14.2	14.0
Arithmetic Standard Deviation ($Bq \cdot m^{-3}$)	16.7	9.0	14.0
Median ($Bq \cdot m^{-3}$)	10.1	11.4	10.6
Mode ($Bq \cdot m^{-3}$)	5.8	6.6	6.1
Geometric Standard Deviation (Dimentionless)	0.74	0.74	0.74
Minimum value of Concentration ($Bq \cdot m^{-3}$)	1.95	3.95	0.67
Maximum value of Concentration ($Bq \cdot m^{-3}$)	119.2	45.4	119.2

DISCUSSION

It is evident from the Table 1 that the average values (arithmetic and geometric) are well below the previously published values of 40 $Bq \cdot m^{-3}$ (3) and below the word average value also being equal to 40 $Bq \cdot m^{-3}$. However, this data well correspond to those for similar climatic and geological conditions, for example, for Cyprus (6). No significant differences have found for the two lowest floor flats and the flats from higher levels. It confirms the fact that radon entry from subjacent earth with low ^{222}Ra concentration is not a limiting factor.

The average concentration which has been found (14.0 $Bq \cdot m^{-3}$) corresponds to the average annual effective dose equal to 0.246 mSv if the average effective equilibrium factor (Rn daughters/Rn) F_{eff}=0.4 would be assumed. If the real effective equilibrium in Kuwaiti houses would be different, e.g.

F_{eff}=0.3 or 0.6, then the average annual effective dose should be respectively modified, i.e. proportionally changed. Therefore, the 'radon component' of the average annual effective dose for population of Kuwait, mentioned above, should be recognised as the provisional, temporary value until the study of average effective annual equilibrium factor would be conduced.

However, it should be recognised that the average radon concentration and the average annual effective dose relevant to that concentration is smaller than the analogical components evaluated for European countries where they lie between 1.0 mSv (for UK) and 6.4 mSv (for Finland).

Even if the effective equilibrium in Kuwaiti houses would be much higher than it has been assumed, i.e. F=0.8 against F=0.4 (assumed), then the relevant average annual effective dose would be higher, i.e. ~0.500 mSv instead 0.246. However, even that value (0.500 mSv) still remains much lower than mentioned values found for European countries.

Nevertheless, the separate regular study of reasons of relatively low concentrations of radon in Kuwaiti houses as well as study of effective equilibrium factor seems to very desirable.

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INDOOR RADON LEVELS IN A PUBLIC SCHOOL AND SOME DWELLINGS FROM THE VILLAGE OF TEIÀ, CATALONIA (SPAIN).

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INTRODUCTION

The radiation dose from inhaled decay products of radon (^{222}Rn) is the dominant component of radiation exposure of the general population (1). Due to the fact that children are more sensitive to radiation than adults, it is of interest to determine the indoor radon levels at which they are exposed. The determination of radon levels in schools is important because children spend there an important fraction of their time. In this paper we present the indoor radon levels measured in a public school and in some of the houses where pupils inhabit. This study was carried out during winter in the village of Teià,, situated at 20 km far from Barcelona, which has only one public school and where the soil is composed mainly of granite rock. Moreover, an estimation of the upper limit of the annual effective radiation dose received by pupils is given.

EXPERIMENTAL METHOD

Indoor radon levels were measured with closed type dosimeters based on Makrofol ED etched track detector. The dosimeter consists of a hemispherical cup (internal radius r = 1.5 cm) of electrically conductive material as a diffusion chamber with a fibreglass filter, where all aerosol and radon daughters solid products are deposited. Therefore, only radon gas can diffuse through the filter. Makrofol foils are covered with aluminised Mylar. The dosimeters were exposed for 3 months in the period November 94 - February 95 and after collecting them, the detectors were etched electrochemically. The etching conditions for the Makrofol ED foils were obtained in a previous study which accounted for the diffusion chamber size (2). The optimal etching conditions found were: (a) chemical etching for 4 h, and (b) electrochemical etching (frequency: 3kHz, voltage: 1000 V_{rms}) for 1.5 h, at 40°C, using a mixture of 50% 6N KOH and 50% ethanol as etching solution. Tracks registered in exposed foils were counted in a 0.97 cm^2 surface area using a semiautomatic track density counting system set up in our laboratory (3). The sensitivity of the dosimeters was obtained by exposing several sets of dosimeters in the Radon Environmental Chamber at the National Radiological Protection Board (NRPB) Chilton Laboratory. The value of the sensitivity corresponding to the optimal etching conditions is (0.89±0.08) tracks.cm^{-2}/kBqm^{-3}h (4). The reliability of the dosimeter calibration was checked in an European Intercomparison Exercise (5) . Background track density of Makrofol ED plates is 7 ± 2 cm^{-2} . The Radon Minimum Detectable Concentration (MDC) was calculated using the Currie (6) relations for well known background (7) and its value is 4 Bqm^{-3} for a 3 month exposure.

DISTRIBUTION OF DOSIMETERS

In the village of Teià, there are two types of houses: those situated in the village centre and those placed in the residential area, where the school is built. Multiapartment dwellings almost do not exist. The school staff placed 10 dosimeters in the school building, distributed in ten classrooms, five from the ground floor and five from the first floor. In addition, 15 pupils were selected to collaborate in the study by placing two dosimeters in their homes: one in the living area and the other one in their bedroom. A questionnaire was handed out together with the dosimeters in order to collect various physical features of the dwelling and of the life style of the occupants. As a result of the small size of the study, there were no dosimeters lost and only 2 questionnaires were missing.

The distribution of the dosimeters is shown in table 1.

Table 1. Distribution of the dosimeters

	Houses	Dosimeters			
		Total	Living area	Bedroom	Other
Residential area	7	14	6(1)	6(1)	
Village	8	16	7(1)	7(1)	
School		10			10 (classrooms)

(1) We do not know the placement of 4 dosimeters due to the two questionnaires lost.

RESULTS

Table 2 shows the results obtained in the residential area, in the village and in the school. These results indicate that radon levels in the school are similar to those corresponding to the houses where pupils inhabit and thus, the fact of going to school does not affect the radon levels at which pupils are exposed to. Even though the number of data is small, the indoor radon levels obtained are similar to those measured in a previous winter survey carried out in the Barcelona area (50 Bqm^{-3}) (4), and also to the world arithmetic mean value (40 Bqm^{-3}) given in the UNSCEAR 1993 report (1).

Table 2 Results of indoor radon measurements in the period November 94 - February 95

	Range	**AM**	**GM**	**GSD**	**n**
Residential area	19 - 94	46	41	1.68	14
Village	8 - 126	52	37	2.53	16
School	20 - 95	48	43	1.66	10

Radon concentration values are given in Bqm^{-3}

AM: arithmetic mean; **GM**: geometric mean; **GSD**: geometric standard deviation; **n**: number of dosimeters

In addition to the dosimeters distributed in living areas and bedrooms, one dosimeter was placed in the basement of a village house. The radon concentration obtained there was 198 Bqm^{-3}, higher than all the other values found in living areas and bedrooms. This result can be a consequence of the lack of ventilation of the basement and also of the fact that the room is in direct contact with the granite soil, typical of this region. Moreover, the radon concentration values obtained in the ground floor of the

school building are higher than those obtained in the first floor (arithmetic means 61 and 46 Bqm^{-3}, respectively) confirming the soil as the most important radon source.

The results obtained constitute an upper limit to the indoor radon concentration annual average because radon measurements were performed in winter season when windows are kept mostly closed. Previous studies carried out in Barcelona confirm that winter concentrations are higher than summer concentrations (4). Thus, the estimation of the radiation dose from our results leads to an upper value for the annual dose that pupils receive. The effective dose was estimated for each pupil using the conversion factor given in the last UNSCEAR report (1) (3.6 nSv per $Bqm^{-3}h$ of radon exposure) considering an indoor occupational factor of 0.8 and assuming that pupils spend 5 hours at school , 10 in the bedroom and 4.2 in the living area. The mean value obtained was 1.2 mSv and the range 0.5 - 2.2 mSv. These values constitute an upper limit of the annual effective dose and are similar to the mean world values (1).

CONCLUSIONS

The indoor radon concentration values obtained in the public school of Teià and in 15 houses where pupils inhabit are similar and comparable to the mean world values.

An upper value of annual effective dose was estimated for each pupil taking into account the school, bedroom and living area radon levels and occupational pattern. The arithmetic mean was 1.2 mSv and the range 0.5 - 2.2 mSv.

ACKNOWLEDGEMENTS

This study was supported in part by the Spanish Consejo de Seguridad Nuclear.

The authors gratefully acknowledge the school management and teaching staff and the pupil families for their collaboration in the study.

Special thanks are given to Ms. Fermina Pérez for her help to carry out this work.

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A PORTABLE APPARATUS FOR CONTINUOUSLY MEASURING Rn-222 EXHALATION FROM GROUND

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INTRODUCTION

For the purpose of continuously measuring the rate of ^{222}Rn exhalation from the ground surface, an apparatus has been developed that consists of a radon collector, a 60-*l* cylindrical buffer tank, three kinds of filter and an ionization chamber of flow-through type.

The apparatus is capable of providing an evaluation of the ^{222}Rn exhalation rate that is sufficiently accurate for all practical purpose.

MEATHOD OF MEASUREMENT AND STRUCTURE OF APPARATUS

The means adopted for measuring the exhalation rate of ^{222}Rn, is an improved version of Wilkining's method.[1][2] The ^{222}Rn collector is of structure and dimensions as shown in Fig.1.

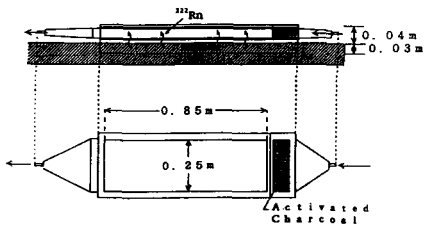
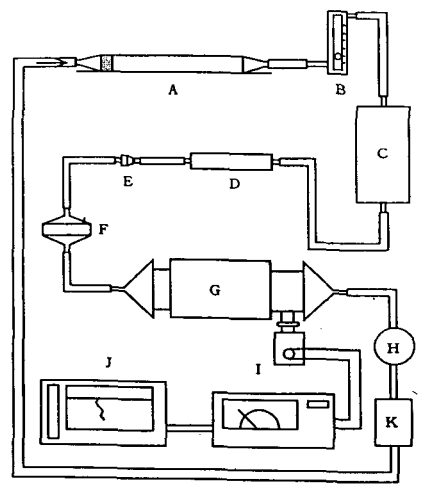


Fig. 1 Device for collecting ^{222}Rn exhalation

The measuring system is arranged as schematized in Fig.2. The ionization chamber(G in Fig.2) is cylindrical aspiration condenser with inner and outer cylinders measuring respectively 0.05 m and 0.2 m in diameter and 0.45 m in common length. The outer cylinder is maintained at a potential of -1 080 volts. The vibrating reed electrometer (I) is used to measure the ionization current.

The measurement proceeds with the respective components functioning in the following manner: The ^{222}Rn collector is placed on the ground to be measured: the sampling air is adjusted to flow at 6 *l*/min, which is a rate that will not let the radon exhaled from ground be forcibly entrained; this sam pling air contained ^{222}Rn and ^{220}Rn is then passed into the 60-*l* buffer tank(C) to eliminate ^{220}Rn, whose half-life is only 54.5 s and thus decays away during its stay in the buffer tank. Further downstream, before attaining the ionization chamber(G), the sampling airpasses through the silica gel desicator(D), the glass fiber filter(E) for eliminating the daughter nuclides of ^{222}Rn and ^{220}Rn, and the ion trap(F) for removing ions generated in the channels of the system by the radiations from ^{222}Rn and ^{220}Rn, as well from their daughter nuclides. The ionization current measured by the electrometer(I) is continuously recorded in analogical form, to be converted off-line into ^{222}Rn concentration byconsulting a calibration table, and the resulting data are then used for calculating the ^{222}Rn exhalation rate, asdescribed in the next section.



A: Radon collector, B: Flow meter, C: Buffer tank (60-*l* capacity), D: Silica gel desicator, E: Glass fiber filter, F: Ion trap, G: Ionization chamber, H: Pump, I: Electrometer, J: Recorder, K: Charcoal bed

Fig. 2 Arrangement of apparatus for measuring ionization current generated by ^{222}Rn exhalation from ground

DERIVING THE ^{222}Rn EXHALATION RATE FROM THE MEASURED IONIZATION CURRENT

For determining the relation between the measured ionization current and the concentration of entrained ^{222}Rn flowing through the ionization chamber, calibration measurements were made on radon gas exhaled from samples of sandy rock used for extracting titanium, containing in average 4 900 Bq/kg of ^{226}Ra.

Before proceeding on a calibration measurement, the initial zero point of ionization current was determined using Miranda's charcoal trap method.[3] Thereafter, the calibration proceeded as follows: Place a suitable quantity of the sandy rock sample in a spare 60-*l* tank; connect this tank and the ^{222}Rn collector(A in Fig.2) to the system; close off the system and leave it standing until establishment of radioactive equilibrium in the tank between radon and its daughter nuclides, with ^{220}Rn decay away; start up the pump(H) to circulate the sampling air through the system, and let system as a whole; adjust the pump speed to obtain a flow rate of 6 *l*/min; while continuing to circulate the sampling at this rate through the system, measure the ^{222}Rn concentration in the tank, applying Thomas' two-filter method;[4] plot the measured ^{222}Rn concentration against the corresponding reading of ionization current on the electrometer(I) to obtain a calibration plot; repeat this calibration measurement with the quantity of sandy rock parametrically varied, to gen-erate the calibration curve.

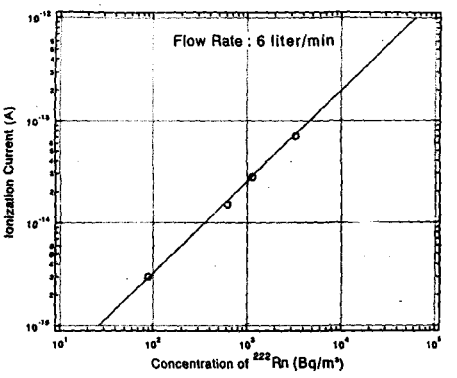


Fig. 3 Calibration plots relating measured ionization current to ^{222}Rn concentration

The calibration curve thus obtained is shown in Fig.3, relating the ionization current to the ^{222}Rn in the sampling air flowing through the ionization chamber at 6 *l*/min. Using the values of ^{222}Rn concentration determined by means of this calibration curve, the exhalation rate is calculated with the equation

$$E = C_{Rn} V_s / S$$

where E : Exhalation rate (Bq/m^{2}s)
C_{Rn}: Radon concentration (Bq/m^3)
V_s : Flow rate of sample air (m^3/s)
S : Covering area by collector (m^2).

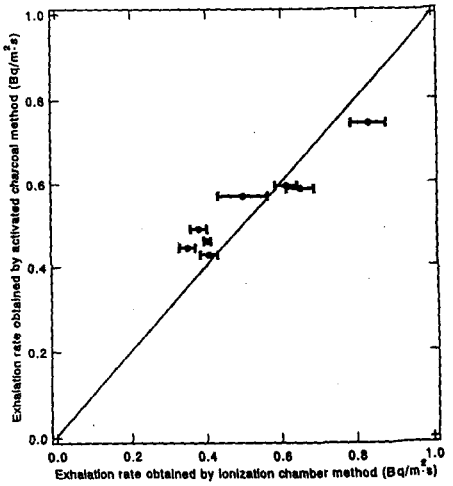


Fig. 4 Measured values of ^{222}Rn exhalation compared between present method and that using activated charcoal

COMPARISON BETWEEN VALUES OBTAINED WITH THE PRESENT AND WITH OTHER METHODS

The accuracy of measured values with the present method was evaluated through comparative measurements with corresponding values obtained using Megumi's activated charcoal method,[5] adopting as common sample parametrically varied quantities of river sand (for its uniformity of grain size) placed in a wooden box measuring 2 m x 2 m x 2 m. The measurements were performed inside a shed, to eliminate wind and other environmental effects.

The results obtained from the comparative measurements are plotted in Fig.4, from which a correlation coefficient of 0.91 has been derived. The plots of Fig.4 indicate a higher range of exhalation rate given by the ionization chamber compared with those by activated charcoal method.

OPERATING CHARACTERISTICS

The operating characteristics of this apparatus were determined from long-duration observations performed on the apparatus at different sites.

An example of the continuous record of exhalation rate is shown with atmospheric pressure and weather con-

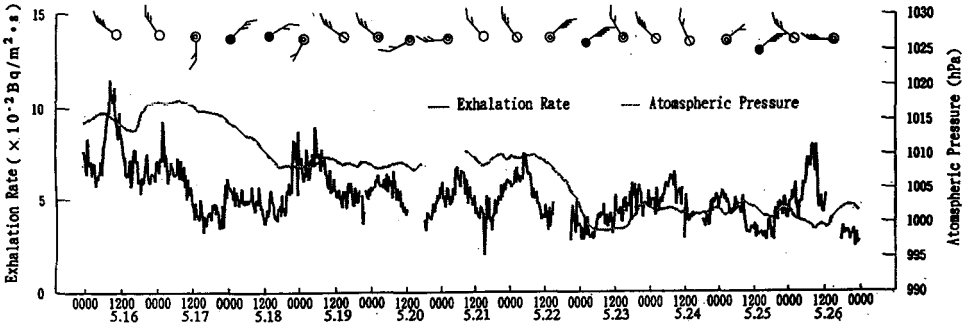


Fig. 5 An example of continuous record of exhalation rate.

ditions in Fig.5. From the observations it was confirmed that the apparatus operates stably in a period of long-duration and even in stormy weather.

CONCLUDING REMARKS

The basic experiments described above, substantiated by performance obtained in practical application in different sites, have yielded the following informations:

(1) The apparatus is capable of continuously recording measured data that provide an evaluation of ^{222}Rn exhalation rate with sufficient accuracy for all practical purposes.

(2) The apparatus operates stably even in stormy weather, and impairment of evaluation accuracy can be expected to be minimized, once a suitable formula -now being sought- is found for taking account of the difference between collector interior and ambient exterior brought by the storm in respect of such factors as the ground conditions, wind and air flow speed.

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RADON EXPOSURE OF PASSENGERS IN THE PRAGUE METRO

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INTRODUCTION

Although the world-wide average annual effective dose from natural sources is 2.4 mSv, in some locations and under some conditions elevated values up to 10 or 20 mSv may be encountered.[1] The additional contributions depend strongly on human activities and practices and show therefore enormous fluctuations. This is especially true with regards to the doses from indoor inhalation of the short-lived decay products of radon gas which are influenced by local geology, building design, and by building or surrounding materials as well as air ventilation. Generally, closed or poorly ventilated enclosures, such as cellars, tunnels, mines and other subterranean areas, are characterised by increased radon concentrations.

Of the three naturally occurring radioactive radon isotopes, namely ^{219}Rn (actinon), ^{220}Rn (thoron) and ^{222}Rn (commonly referred to simply as radon), the last one, owing to its relatively long half-life (3.824 d), is the most significant for human exposure.[2,3] The production of ^{222}Rn in the terrestrial materials depends on the activity concentration of its parent nuclide ^{226}Ra, a member of the ^{238}U series.

It is well known that radon-related exposures are actually almost entirely caused by the inhalation of their decay products rather than by the inhalation of radon gas itself. Consequently, from the radiological point of view, information concerning the concentration of radon decay products is of primary importance because it can be interpreted in terms of the appropriate radiation protection quantity, i.e., the effective dose. Since the quantity usually monitored is radon gas concentration (C_{Rn}), the parameter *EECR* (equilibrium equivalent concentration of radon) is commonly obtained using the equilibrium factor (F) which is defined as the ratio of *EECR* and C_{Rn}. Generally, the equilibrium factor is taken to be 0.4 for indoor exposure and 0.8 for outdoor exposure. For more accurate evaluation the factor F has to be derived from simultaneous measurements of both radon concentration and concentrations of its decay products. In intensively ventilated objects, the equilibrium factor may often be below 0.4.

CHARACTERISTICS OF THE PRAGUE METRO

The Prague Metro[4], whose first section between the *Florenc* and *Kačerov* stations was opened in 1974, is the most important public transport system in the capital of the Czech Republic now serving more than 1.2 million commuters each day. This share represents about 44% of all passengers using all Prague mass transport systems. At present, the Metro network consists of three lines A, B and C, with a total rail length of 45.1 km and 46 stations altogether. The busiest line is line C (42%), followed by line A (38%), while line B transports about 20% of all persons using the Metro which operates daily from 5:00 AM to midnight.

Some further characteristics of the Prague Metro network are illustrated in Fig. 1, which shows the depth profiles of individual lines and positions of the stations. The deepest station is station *Nám. míru*, 52 m below ground level. On the other hand, the deepest location in the tunnels is between the stations *Hradčanská* and *Malostranská*, about 68 m below the ground. As to the absolute value related to sea level, the lowest place is in the tunnel of line A beneath the River Vltava (about 19 m below its bed).

The inner spaces of tunnels, stations and other Metro areas are built using reinforced concrete, prefabricates and steel, and they are tiled with natural materials and, in some places, with marble or aluminium plates. These materials provide essentially a water-tight encasement for all inside areas. The Metro is intensively ventilated by means of powerful blowers and fans; the piston effect of the moving trains also contributes to the air exchange. The ventilation rate is typically 3-4 h^{-1}; the highest rate is in line C where the air in all underground areas is completely exchanged 6 times each hour.

INSTRUMENTATION

Radon measurements were carried out using AlphaGUARD[5], a portable radon monitor using an ionisation chamber with an active detection volume of 0.56 L. In order to overcome problems with long-tail output pulses as well as microphonic artefacts, this instrument uses a special digital signal processing technique. The pulses are counted, stored and converted into the corresponding radon concentration whereas 1 cpm corresponds to 20 Bq m^{-3}. One of two sampling intervals, 10 or 60 minute, can be selected. The data capacity of a non-volatile memory is 20 days at 10 min sampling or 4 months at 60 minute measuring intervals. The instrument can work

continuously for at least 10 days with an internal rechargeable battery. The monitor is suitable for continuous monitoring of ^{222}Rn concentrations covering the range from 2 Bq m^{-3} up to 2 MBq m^{-3}. The sensitivity for ^{220}Rn is only about 5% of that for ^{222}Rn.

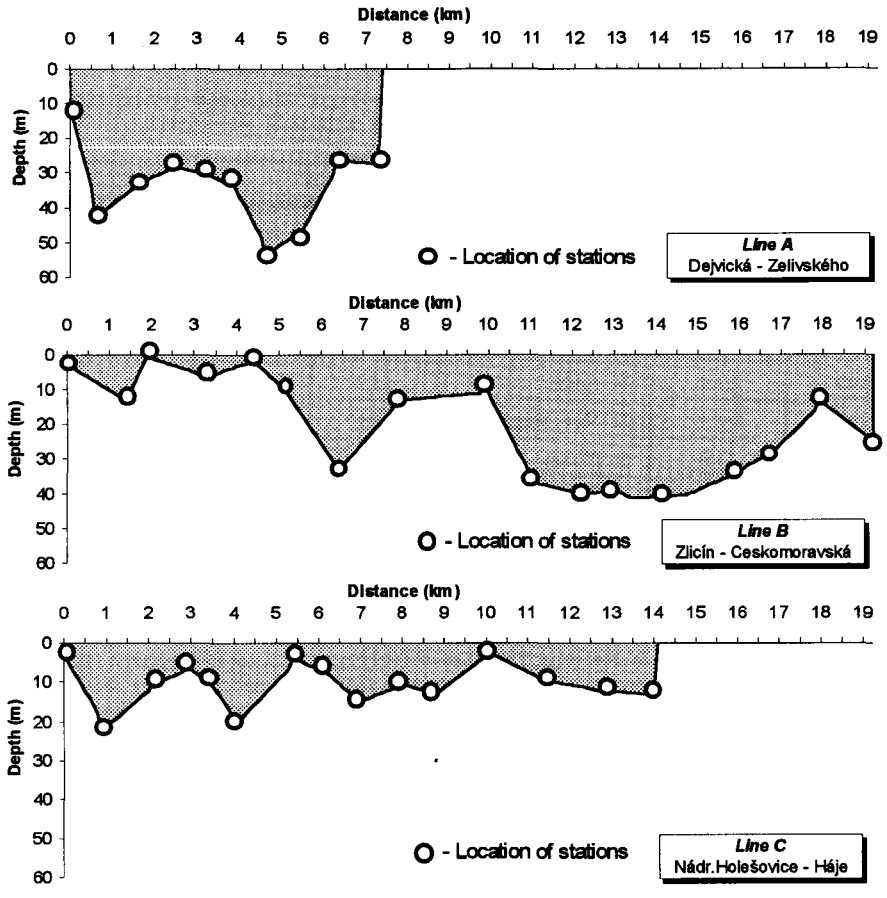

Fig. 1. Some characteristics of the Prague Metro (the distance, depth and position of the stations).

In addition to radon concentration in the air, this radon monitor at the same time also measures and records ambient temperature, relative humidity and atmospheric pressure. By combining simultaneous monitoring of the radon level and these associated environmental parameters, some interesting correlations can be observed..

The measuring results may be seen on a 2 line x 20 character alphanumeric display indicating the current values of all parameters. The data from the monitor can be transferred through the RS-232 output to a PC, where they are presented in the form of diagrams and can be evaluated further. These operations are performed by means of the available company software for the analysis of multiple parameters.

RESULTS AND DISCUSSION

The measurements of both airborne radon concentrations and gamma dose rate were carried out in the period of January - May 1995. The results of monitoring in the trains and the selected stations are summarised in Table 1 showing radon and gamma radiation levels given in appropriate units. Special attention has been

paid to line A which has several stations and tunnel sections at the greatest depth positions below the ground surface (Fig. 1).

Most of the measurements presented in this survey took 2-7 days and were carried out in winter 1994/95. The results given in Table 1 may be considered representative although some small variations may be expected in comparison to the average for the whole year.

Table 1. Results of radon and gamma measurements at some Prague Metro underground stations and in trains (St. and Train results refer to the values at the station and in the train, respectively).

Place	C_{Rn} (Bq m^{-3})	Place	C_{Rn} (Bq m^{-3})	Place	C_{Rn} (Bq m^{-3})
Line A		Line B		Line C	
St. Dejvická	15.5	St. Zličín	13.2	St. Nádr. Holešovice	12.7
St. Hradčanská	11.3	St. Lužiny	12.0	St. Florenc	9.2
St. Staroměstská	11.5	St. Nádraží Smíchov	10.8	St. Muzeum	13.0
St. Muzeum	17.4	St. Anděl	12.9	St. Vyšehrad	8.9
St. Nám. Míru	12.4	St. Nám. Republiky	14.1	St. Pankrác	11.2
St. Želivského	13.0	St. Florenc	9.9	St. Kačerov	12.1
St. Skalka	19.8	St. Českomoravská	11.1	St. Háje	9.7
Train	12.6	Train	11.7	Train	10.8

CONCLUSION

Due to intensive ventilation, average radon concentrations in the Prague Metro are generally less than 30% of the concentrations found in Czech dwellings. The effect of shielding against cosmic radiation (soil and rock layers, lining and construction materials) results in a slight reduction of external penetrating radiation within the stations. In addition, the robust construction of the Metro trains also contributes to the decrease of dose rates found in the trains.

This paper summarises the results of radon concentration monitoring in the carriages and at some stations of the Prague Metro network. The measurements revealed that radon levels in the Metro are relatively low in comparison to those normally encountered in dwellings in the Prague region. On average, the radon concentrations in the air inside the carriages have been found to be about 11 - 12 Bq m^{-3} while the levels at most stations reached values between 10 and 15 Bq m^{-3}. The Metro is intensively ventilated by means of powerful blowers and fans; the piston effect of the moving trains also contributes to air exchange. The ventilation rate is typically 3-4 h^{-1}. The highest rate is in line C, where the air in all underground areas is completely exchanged 6 times within each hour.

These results demonstrate that Metro passengers receive about the same effective dose as passengers using surface transport. The doses from radon in the metro are only slightly higher than radon-related doses in the open air, while exposure due to external photon radiation seems to be a few percent lower than dose rates common in typical Czech houses.

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RADON-222 CONCENTRATION IN WATER AND THE EXPOSURE OF THE PUBLIC,

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INTRODUCTION

Radon and its short-lived decay products are greatly concerned by the world scientific community since such exposure constitutes nearly 50% of the effective dose received by the public in countries with temperate climates(UN93). Vigorous researches have been conducted to investigate many aspects of the indoor ^{222}Rn problem. The predominant source of indoor ^{222}Rn in China genera lly appears to be the soil adjacent to the foundation. The other major sources, potable water and building materials, occasionally contribute significantly to indoor radon concentrations, and may be the dominant source in some cases. Several investigations in the past decade were conducted to measure ^{222}Rn concentrations in public water supplies, in wells, thermal springs and underground thermal water in China. Radon concentrations in drinking water have been observed to range over an extremely large range. The highest ^{222}Rn concentration recorded so far is 11 MBq m^{-3} in a private well of Guo Village, Jiangxi province(Li81). The ^{222}Rn concentrations in domestic water from public water supplies of the Chinese major cities are depends on the types of water supplies. The purpose of this paper is to summaries the results of ^{222}Rn concentrations in public water supplies of Chinese major cities, each with population larger than one million, and to estimate the contributions of waterborne radon to indoor air radon concentration as well as the related doses received by the residents from waterborne radon.

METHODOLOGY

The investigation was conducted in nearly 100 Chinese cities, which either are the capital cities of each province, municipal city directly under the Central Government, or are cities with population larger than 0.4 million. Several sites were selected in each city, and at each site, two representative tap water samples were sampled from water tap with radon bubblers after allowing about 10 minutes water flow. Each sample, with a volume of 50 cm^3 water, was collected into 100 cm^3 radon bubbler. High pure nitrogen gas was used to degassing Rn dissolved in water and drove it from bubbler into scintillation flask for measurement in laboratories.

RESULTS

1. Radon Concentrations in Different Types of Water

Radon concentrations in different types of water have been observed to range over an extremely large range, as listed in Table 1. The average concentrations in 138 thermal springs, most of them are famous in China, and 93 underground thermal water wells are 118.2 and 36.3 kBq m^{-3}, respectively, The highest concentration found is 11 MBq m^{-3}, in a private well, Guo Village, Jiangxi Province. The ^{222}Rn concentrations in domestic water from public water supplies of about 100 Chinese cities are ranging from 0.035 kBq m^{-3}, in

Fuzhou, to 101.3 kBq m^{-3}, in Shenyang, with average of 7.82 kBq m^{-3}.

Table 1. Radon concentrations in different types of water (KBq m^{-3})*

Water Type	No. of samples	range	Mean+-1SD
Surface	6	0.035- 2.98	0.27+- 0.28
Wells	9	0.68 -11000	20.5 +- 14.8
Thermal spring	138	0.3 -2720	118.2 +-266.3
Underground thermal water	93	0.3 - 259	36.3 +- 29.0
Tap water		0.053-101.3	9.5 +- 13.1

* Data from Ch93, Ch84, Ch87, Fa93, Ji89, Li81, Wa91,Zh87, Zh89.

2. Radon Concentration in Public Water Supplies

Table 2 shows the ^{222}Rn concentrations measured in the public water supplies of 24 Chinese major cities, each with population larger than one million. The average ^{222}Rn concentrations in domestic water from public water supplies of the Chinese major cities, are in the range from average of the 24 major cities is 8.61 $\pm$ 13.3 kBq m^{-3}, For those cities with Surface public water supplies systems, for example, Shanghai, Tianjin, Guangzhou et al., the average radon concentration is 0.27 $\pm$ 0.28 kBq m^{-3}, ranging from 0.05 to 7 kBq m^{-3}. In some cities, the main sources of thier public water supplies are from well or underground water, the waterborne ^{222}Rn concentration are much higher with average value of 20.5 $\pm$ 14.8 kBq m^{-3}, ranging from 0.68 to 101.3 kBq m^{-3}.

Table 2 Radon concentration in public water supplies(kBq m^{-3})*

City	Mean+-1SD	City	Mean+-1SD
Beijing	14.8 +- 2.2	Zhengzhou	1.22+- 0.09
Haerbing	5.02+- 0.26	Xian	7.98+- 0.53
Changchun	0.96+- 0.18	Nanjing	2.13+- 0.05
Wulumuqi	21.2 +-22	Wuhan	0.33+- 0.06
Anshan	34.8 +- 3.6	Shanghai	2.46+- 1.87
Shenyang	54.7	Chengdu	9.6 +-10.5
Sijiazhuang	18.0 +- 2.1	Hangzhou	1.39
Tianjin	0.27+- 0.11	Chongqing	0.44+- 0.17
Lanzhou	26.4 +- 3.2	Changsha	0.45+- 0.04
Dalian	0.73+- 0.09	Kungming	0.98+- 0.20
Taiyuan	12.3 +- 0.7	Nanchang	6.44+- 0.01
Jinan	5.26+- 0.34	Guangzhou	0.85+- 0.43
Aggregate water supplies			9.5+-13.1
Surface water supplies			1.0+-0.7
Underground water supplies			20.5+-14.2

* Some data from Reference Ch93.

3. Transfer Factor of ^{222}Rn from water to Indoor Air
An important question in assessing the impact of waterborne ^{222}Rn on indoor exposure is to know the transfer factor, f, of ^{222}Rn from water to indoor air. Transfer factor can be expressed as f= Ca/Cw, Ca and Cw are ^{222}Rn concentrations in air and in water, respectively. Several measurements on the values of f were made. One study measured indoor air ^{222}Rn concentrations in real conditions in bathrooms using underground thermal water by dual filter grab sampling method. The values of f were in the range of 2 10^{-3} to 26 10^{-3} and average and standard deviation were 7.2 10^{-3} +- 7.1 10^{-3}(Ch84). In another study, ^{222}Rn concentrations were measured in a room with volume of 15 m^3 by grab sampling. The average f values were 2.4 10^{-4} and 3.2 10^{-3} under window open and close conditions(Ch93).

4. Increment of ^{222}Rn Concentrations in Indoor Air Due to Water Use
We take 2 10^{-4} as typical value of the ratio of airborne to waterborne ^{222}Rn concentrations. The increments of radon concentrations in indoor air are 0.2, 4.1 and 1.9 Bq m^{-3} for the three different types of public water supplies listed in Table 2. These increments equivalent to 0.8%, 17% and 8% of the nation-wide average of indoor radon concentration, respectively. This suggest that the reference concentration, 10 kBq m^{-3}, adopted by UNSCEAR in its 1993 Report is reasonable. The estimated annual effective doses to the sesidents, are 2.3,177 and 734.3 10-6 Sv, respectively, by inhalation pathway, including doses from inhaled radon that becomes dissolved in tissues.

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RADON MEASUREMENTS IN THE SHENZHEN REGION

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INTRODUCTION

In recent years there has been growing concern about radon levels indoors in China. Shenzhen is a coastal, subtropical city having a typical maritime climate with average annual rainfall of 1948 mm. The city is famous for its very fast economical and social development speed. These features make an opportunity to study ^{222}Rn variation and the controlling factors. Measurements of indoor ^{222}Rn concentrations in Shenzhen were carried out in two phases, 1986 and 1992.

METHODOLOGY

Many measurements were made in different types and different usage of buildings at widely distributed locations in both phases. Compared with the first phase, much more buildings were well fit up inside with plain or color painting or paper or marble stone and air-conditioned in the 1992 investigation. In the first phase, most of the measurements were conducted with a dual-filter environmental radon monitor having a decay volume of 15l, and the lower limit of detection(LLD) was 2.3 $Bq.m^{-3}$ for ^{222}Rn for 30-min counting. Another grab sampling technique used was a scintillation flask radon monitor with volume of 0.7l and the LLD was 3.6 $Bq.m^{-3}$, using mainly for checking or for rural area measurements. Integrated activated carbon detectors were also employed for comparison. All these instruments were calibrated with liquid radium sources. The in situ intercomparison showed that the results of these three methods agreed within +-24%[1,2] .

In the second phase of investigation, solid state nuclear track detectors were employed for measuring radon concentration in air. CR-39 chips supplied by Pershore Mouldings LTD, England, were placed in Karlsruhe diffusion chamber, which were covered with polyethylene filters. After exposure, chips were etched under the condition: 6.25N NaHO at 80^0 C for 4 hours. The LLD calculated is 5554 Bq h m^{-3} , equivalent to radon concentration of 7.0 Bq m^{-3} for one month exposure. Intercomparison test was made in a radon chamber and the results of CR-39, scintillation flask and electret passive environmental radon monitor(E-PERM) all agreed within +-13%[3].

RESULTS AND DISCUSSIONS

1. Outdoor and indoor radon concentration

In the first phase 222 dwellings were measured at sites selected[2]. The total sites were 176 in the second phase, having same site distribution with the first phase of investigation[4]. The arithmetic mean of outdoor radon concentration obtained in

second phase is 13.0 Bq m^{-3} , which is in good agreement with 1986's value of 13.7 Bq m^{-3}. The city-wide indoor radon average is 33.7 Bq m^{-3} . These values are higher compared with the province-averaged indoor and outdoor radon concentrations in China, being 23.6 and 10.7 Bq m^{-3}, respectively. Shenzhen is geographically located in a uranium-rich granite rock region, its soil contains higher radium, average being 84.7 Bqkg^{-1} , about 2.2 and 3.4 times of China's and world average, respectively. The facts may explain the higher radon value.

2. Radon in different types of building

The measured ^{222}Rn concentrations, extremes, arithmetic and geometric means, in several types of buildings and outdoors are given in Table 1. In the first phase, ^{222}Rn concentrations in detached houses and multi-family apartments are similar, however, the average in high-rise buildings is significantly higher than in other two types of dwellings. One of the reasons is that the air exchange rate is much lower in high-rise buildings, being central or separate mechanically air-conditioned, than in other types of dwellings. The same explanation can applied to 1992's results, in which arithmetic mean of radon concentration in multi-family apartments is raised to a level of 31.0 Bq m^{-3}, because more and more apartment, about 83% of the total dwellings investigated, set up air-conditioner in recent years. In the opposition, the 1992's mean value of radon in basement is significantly lower than that of 1986', because more basement space is used as cafeteria or department store, in which ventilation condition was improved.Radon concentrations in naturally ventilated detached houses and 1986's multi-family apartments, whose air exchange rates were about 2 h^{-1} and 15 h^{-1} for closed and open conditions, respectively, are about the same in the two phases and only slightly higher than outdoor radon values. Their values of ratio of indoor to outdoor radon are between 1.1-1.3, in the contrast with values between about 2.3-7.9 for other types of dwelling. This supports the UNSCEAR's prediction: indoor and outdoor radon concentrations for well-ventilated buildings should be essentially equal. It indicates that social and economic development, through indoor air ventilation, is one of the controlling factors of indoor radon concentration.

3. Dose estimation

Exposure to radon and its decay progeny comes mainly from the inhalation of the products of radon. Table 2 lists the annual effective doses in ventilated and unventilated dwellings, In the calculation the conversion coefficients relating average annual concentrations to effective dose and occupancy factors for indoor and outdoor given by UNSCEAR[1] are used.

Table 1. Measured ^{222}Rn concentration in different types of dwellings in Shenzhen (Bq m^{-3})*

Types of dwelling	Number of dwellings	Range	Arith. mean	Standard deviation	Geom. mean	Geometric standard deviation
High-rise buildings	147 (65)	1.5-14.2 (13.3-87.7)	31.1 (39.7)	22.9 (17.4)	25.2 (36.3)	1.96 (1.5)
Multi-family dwellings	19 (80)	3.3-52.5 (12.1-98.1)	15.5 (31.0)	11.8 (18.9)	11.8 (26.2)	2.16 (1.8)
Detached houses	50 (24)	4.1-54.0 (6.7-53.0)	17.8 (19.5)	11.8 (12.7)	14.4 (16.3)	1.93 (1.8)
Basement	6 (7)	77.7-142 (35.5-86.2)	108 (62.5)	26.3 (22.4)	106 (58.1)	1.27 (1.5)

* The values in the parentheses are results of 1992 investigation.

Table 2. Estimation of annual effective dose caused by ^{222}Rn and its progeny*

types of dwelling	Typical concentration (Bq m^{-3})	Effective dose rate (nSv h^{-1})	Annual effective dose (mSv) (I)	(I+O)
Ventilated	36	142	0.97	1.08
Unventilated	16	63	0.43	0.54
Basement	65	257	1.8	1.91
Outdoor	13	62		

* "I" and "O" stand for "Indoor" and "Outdoor", respectively. Outdoor contribution to annual effective dose is 0.11 mSv a^{-1}

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INDOOR RADON MEASUREMENTS IN DWELLINGS OF GARHWAL HIMALAYA, NORTHERN INDIA

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Measurement of indoor radon and daughters concentration were performed in several houses in Garhwal Himalaya during 1993-95 with solid state nuclear track detector films (LR-115 Type II). The detector films were exposed for a period of three months to one year. The films basically measured total airborne alpha activity but may be calibrated in unite of EEC_{RN} (equilibrium equivalent concentration of radon with equilibrium factor F=0.45) in an environment with known radon and daughters concentrations. A numbers of dwelling in the area exhibited radon daughters concentrations (EEC_{RN}) exceeding the recommended level. The abnormal values are due to typical house construction (mud house) in the area. The houses are constructed with soil and local stone with a thin paste of mud. Behaviour and abnormality of radon in mud houses are discussed in details the corresponding annual effective dose has been calculated.

INTRODUCTION

Nearly one half of the total natural radiation exposure is due to the inhalation of radon and its progeny in air(1) which can result in a significant risk to the general public. The health effect of radon is well known in mine workers (2,3) but recent survey carried out all over the world (4-10) shows high concentration of radon in some dwellings could entail significant health risk. The daughter products of radon, which are solid under ordinary condition, attach themselves to atmospheric dust. During inhalation these particles may deposit in the lung and damage the tissue. It is therefore desirable and possibly necessary to monitor levels of radon in places where people are exposed to radon, particularly at sites where the geological formations are enriched with uranium. This paper presents the results of radon survey carried out in the houses of Garhwal Himalayas using LR-115 type II, plastic track detector.

CALIBRATION OF DETECTOR

The experiment on the calibration of LR-115 type II, plastic track detector was performed at Environmental Assessment Division, Bhabha Atomic Research Centre, Bombay. The detector films (2.5 cm X 2.5 cm) were exposed in bare mode in a calibration chamber of known radon concentration. The condition was kept similar to that in the field measurements. In all 12 samples were exposed and an average value was obtained as 3.12×10^{-2} tracks/cm^{2}d = 1 Bq/m^3.

EXPERIMENTAL TECHNIQUE

Track etch technique has been used for the measurement of radon activity in the dwellings. Small pieces (2.5 cm X 2.5 cm) the LR-115 were fixed on a glass slide which acts as a supporting material and these slides were suspended inside the houses at a height of about 2 meters from ground floor. The detectors are then removed after three months, etched in 2.5 N NaOH solution at 60 °C for two hours and were scanned under an optical microscope for track density measurements. The resulting values of track densities are converted in Bq/m^3 by using the calibration factor as discussed above.

RESULTS AND CONCLUSIONS

The results of indoor radon concentration in Garhwal Himalaya are given in Table 1. The choice of house was random and one room of each house selected for radon measurement. Most of the houses in the area are mud houses, constructed with local stone and soil.

Table 1. Radon concentration and equivalent dose in the dwellings of Garhwal Himalayas.

Place	Average Radon Concentration (Bq/m^3)	EEC_{RN} (Bq/m^3)	Dose (mSv/y)
Padiyargaon	199	90	7.9
KotiColony	122	55	4.8
Tehri	110	50	4.3
New Tehri	119	54	4.7
Nail	119	54	4.7
Chamma	109	49	4.3
Dikholgaon	76	34	3.0
Thanegaon	64	29	2.5
Sabligaon	72	32	2.8
Khanda Srikot	178	80	7.0
Srinagar	75	34	2.9
Chauras	69	31	2.7
Kotdwar	33	15	1.3
Malideval	133	60	5.2
Serain	82	37	3.2
Rajgaon	118	53	4.7
Uppu	48	22	1.9
Dang	54	24	2.1

The results reveal that radon concentration in mud houses are relatively higher than that in the normal buildings. This variation may be due to the building material and mode of construction of the mud houses. The ground floor of such houses allows more radon to diffuse inside the room because of higher porosity of the material used. The emanation of radon from building material (stone and soil) is also higher than the normal building material and may contribute additional radon inside the room. As such high radon concentration is recorded in mud houses.

The radon concentration is expressed in terms of equilibrium-equivalent radon concentration (EEC_{RN}) by using the following relation:

$$EEC_{RN} = F.A_{RN}$$

Where F is the equilibrium factor (F=0.45) and A_{RN} is the measured radon activity. The dose equivalent received by bronchial and pulmonary regions of human lungs have been calculated using a conversion factor 1.0×10^{-5} mSv per $Bq.h/m^3$ (4).

The calculated dose for indoor radon values were found to be four to six times higher than the recommended value (4). The total relative life time risk of lung cancer from inhaled radon and daughter concentration in the study area was calculated as 0.52 per Bq/m^3. This indicates that the lung cancer risk for a population exposed throughout an average life time to the calculated mean EEC_{RN} is more than half of normal lung cancer. These abnormal recorded values in the area finds a scope for the further study in the areas with identical geographical and geological conditions. A detailed study is already in progress in other areas of Garhwal Himalayas.

Acknowledgements - Author is grateful to BRNS, DAE, Govt of India for providing financial assistance under the research scheme No. 27/4/93-G.

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INDOOR RADON CONCENTRATION AND GAMMA DOSE IN HUNGARIAN DWELLINGS

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INTRODUCTION

The first measurement of indoor dose in Hungarian dwellings were made in 1979. From this date the system is working as a passive dosimetry network. The indoor radon concentrations were measured during 1985-1986 at the sites of the network. The number of the sites are 123 which is rather low.

The aim of the nation-wide survey during 1993-1994 was to obtain results of indoor dose and radon concentration in represantative dwellings and in thi base to determine the average dose equivalent to the population. The effect of wall materials and geographical situation were also examined.

One thousand of dwellings were selected as a representative sample which means 1 in 3.800 of the housing stock. The sites cover uniformly the territory of the country (1 site/km^2).

Method used are $CaSO_4$:Tm TLD's for the measurement of dose and electret ionization chambers for the measurement of radon activity concentration. Exposure duration is one year. The detectors were distributed personally and were collected through postal service. The lost of detectors is 8 per cent.

RESULTS

1) Radon concentration and exposure

The arithmetic mean of indoor radon concentration is 128 $Bq.m^{-3}$, the median value is 81 $Bq.m^{-3}$. The concentration of radon in groun floor dwellings are higher than in dwellings above it. The lowest radon concentration is in prefabricated multi-family houses.

The frequency distribution of indoor radon concentration is nearly log-normal. 17 per cent of the dwellings have radon concentrations above 200 $Bq.m^{-3}$ and 2 per cent of the houses exceed 600 $Bq.m^{-3}$.

Taking into account the arithmetic means of the radon concentration in dwellings with different wall materials, their position to the soil (ground floor and above it) and the population density the weighted mean is 107 $Bq.m^{-3}$ radon. The estimated annual effective dose is 1,3 mSv, supposing that the occopancy in dwellings is 5.000 hours/year, F factor is 0,4 and the other parameters are the same as in the Recommendation of ICRP 65 (3).

2) Indoor dose rate and exposure

The arithmetic mean of indoor dose rates is 127 $nGy.h^{-1}$, the median value is 133 $nGy.h^{-1}$, including the secunder cosmic rays and terrestrial radiations. The arithmetic mean of the dose is the lowest in dwellings in prefabricated multifamily houses compare with other buildings. The weighted mean is 109 $nGy.h^{-1}$. Based on this result the estimated annual effective dose indoors from terrestrial sources and cosmic rays is 0,43 mSv using the same parameters as for radon and Sv/Gy factors according to the UNSCEAR 1993 Report (4).

3) Annual effective dose to adults from terrestrial and cosmic rays and from radon inhalation in dwellings

The sum of the estimated effective dose based on the weighted means is 1,73 mSv for 5.000 hours yearly occupancy and is 2,4 mSv for 7.000 hours. It can be seen that about 75 per cnt of the indoor exposure is because of radon inhalation.

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Table 1.

Radon activity concentration in Hungarian dwellings

Number of sites	Arithmetic mean	Standard deviation	Extreme values	Median value	Geomtric mean
		$Bq.m^{-3}$			
998	128	163	1990-10	81,4	81,0
weighted mean	107				

Table 2.

Indoor dose rates in Hungarian dwellings+

Number of sites	Arithmetic mean	Standard deviation	Extreme values	Median value
		$nGy.h^{-1}$		
1092	127	30	268-43	133
weighted mean	109			

+ including secunder cosmic rays

SEASONAL VARIATION OF RADON GAS CONCENTRATIONS IN KINDERGARDENS OF SERPUKHOV MOSCOW REGION

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The problem of radiation exposure of population from radon gas in indoor air is well-known now and the national regulations on this factor are adopted in many countries [1]. But from our point of view existing regulations don't take into account the special group of population - childrens and tine adgers whose organizms are significally more sensitive to this factor. In this report we present some preliminary data on radiation exposure of childrens in schools and kindergardens from natural radon gas in t.Serpukhov(Moscow region) during 1993-1994 y.y.. Presenting this report we hope to attract attention of IRPA-96 Congress scientists to discuss the question about the necessity of new regulation on radon gas factor for childrens and tineadgers. The 5-th International Symposium in Salzburg, 1991, Natural Radiation Environment, tooke more attention for radon gas concentration in dweling, but only one publication concerned the above mentioned problem [2]. Me have measured the concentration of radon gas and it's DDP (daughter disintegrated products) in 36 kindergardens and schools in July 1993 practically simultaneously with the goal to choose objects for the following investigations. The results of DDP concentration (Crd) measurements are shown in Table 1.

Table 1. The number of rooms due to Crd, %.

Crd, Bq/cub.m	**0...40**	**40...100**	**100...150**	**150...200**
Number,%	**67**	**28**	**4**	**1**

For the more detail observation we choose three buildings with different radon gas concentrations. The procedure of measurement consists of determination of momentary radon gas concentration, momentary DDP concentration Crd and two - days integrated radon gas concentrations one time per month during one calendar year. The obtained data are given in Table 2.

Table 2. Momentary daughter concentration Crd, Bq/cub.m.

(The upper line represents month of measurements)

Dat	De	Fe	Fe	Ap	Apr	May	Jun	Jl	Au	Au	Oc	Nov
Crd	25	19	40	98	112	125	123	53	58	34	37	102

The mean Crd values per year are: for building 1 - 65 Bq/cub.m, for building 2 - 31 Bq/cub.m, for building 3 (background) - 11 Bk/cub.m. The mean equlibrium coefficient per year between radon and it's DDP for all investigated buildings was equal 0.6. But for the definition of the effective total dose it's more important to know the Crd daily variations. These data are shown in Tabl.3 for building 1.

Table 2. The daily variation of momentary DDP concentration Crd,

Bq/cub.m.(The upper line represents the hours of measurements)

H	11	12	13	14	15	16	17	18	19	20	21	22
Crd	92	73	62	79	78	65	58	38	40	43	38	56
H	23	24	1	2	3	4	5	6	7	8	9	10
Crd	53	88	77	53	35	39	52	50	50	67	124	131

Now there is the possibility to take into account the real children exposure during the maximum daily time from 8 a.m to 6 p.m.(Tabl.4).

Tabl.4. Exposure of children in daily time (1840 hours), mSv/y

Buildings	Dose from DDP	External gamma dose	Summary dose
N 1	1.35	0.26	1.61
N 2	0.65	0.24	0.89
N 3	0.32	0.22	0.54

These data must be compaired to "normal" effective population dose per year (8500 hours) from radon gas 1.26 mSv/y [3].

Summary:

- the using of only integral methods for the determination of mean value C per year is not enough. It's nesessary to have the additional data on the daily varyations of it to correct the estimation of effective dose.

- from our point of view it is nesessary to have the special regulation on children exposure from the radon factor.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
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PAPER TITLE COMPARATIVE MEASUREMENTS OF HIGH OUTDOOR RADON LEVELS IN DIFFERENT MINE ENVIRONMENTS IN SLOVENIA

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Beside well known sources of radon at the Žirovski vrh Uranium mine there are also other disposal sites with technologically enhanced natural radioactivity, distributed in several non-uranium mining districts of Slovenia. The deposited material, such as uranium waste rock and mill tailings, roasted mercury ore residues, waste coal residues, fly-ash, bauxite waste, could be classified according to the levels of uranium content as low activity radioactive wastes or at least as materials with restricted use.

Environmental concentrations of radon and radon daughters were measured in the nearby surroundings of disposal sites, simultaneously at different mine districts of the country, mainly by continuous alpha spectrometry. It was found that the highest radon levels - far exceeding a value of 100 Bq/m^3 - are not related to uranium production but rather to coal and mercury mining. The results were evaluated taking into account site specific features, including size and position of radon sources, and topographic and climatic characteristics of environment. Typical alpha spectra were also recorded and values of equilibrium factor at sampling points near ^{222}Rn sources were estimated. The data obtained are important base for future restoration of all contaminated sites.

RADIATION EXPOSURE OF THE MOLDAVIAN POPULATION FROM RADON AND THORON PROGENY

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INTRODUCTION

The aim of this paper is to provide un updated, completive review of the data available for assessing the Moldavian population exposure from radon and thoron progeny inhalation and the potential risk of radiation-induced lung cancer. According to our last estimate (1), the short - lived decay products of radon and thoron make the most significant contribution to exposures from natural radiation background (2.5 $mSva^{-1}$).

METHODS

The radon and thoron daughters concentrations have been measured in 348 typical Moldavian dwellings from urban and rural areas using the active method by sucking air through filter and counting the deposited activity by means of a ZnS alpha scintillation counter. Measurement and calculation procedures have been described previously (2).

The exposure of individuals resulting from radon and thoron progeny inhalation has been expressed in terms of effective dose (E), the new indicator of the total detriment due to the stochastic effects in the exposed individual and his descendents, as defined by ICRP in Publication 60 (3).

The annual effective doses have been derived from the average Equilibrium Equivalent Concentration (EEC) of radon and thoron using the dose coefficients adopted in the UNSCEAR 1993 Report (4) and assuming the average occupancy factors of 0.75 indoors and 0.25 outdoors.

The radiation-induced risk estimates have been performed from the annual collective effective doses by applying the risk factor of $5x10^{-2}$ per Sv for fatal cancers for a population of all ages and both sexes,recommended by ICRP Publication 65 (5).

RESULTS AND DISCUSSIONS

Table 1 shows the arithmetic means and the range of values for both EEC of radon and thoron and the resulting annual effective doses. The average value of the equilibrium equivalent concentrations of radon have been estimated at 8.7 Bqm^{-3} in block of flats and at 28.0 Bqm^{-3} in detached houses. The highest value found has been of 564 Bqm^{-3}, in a rural house. It is evident from estimates given in table 1 that EEC of radon levels are about three times higher in detached house than in block of flats. The explanation is that in detached houses the major source of indoor radon is the subjacent ground but, in the upper storeys of tall block of flats, the building materials might be expected to be the main source.

The resulting annual effective dose of an individual living in a detached house is 1656 µSv on average while in a flat, un individual receives only 514 µSv.

Table 1. Equilibrium Equivalent Concentrations (EEC) of radon and thoron in dwellings and the resulting annual effective doses

Source of exposure	Location	EEC (Bqm^{-3}) Average	EEC (Bqm^{-3}) Range	Annual Effective Dose (μSv) Average	Annual Effective Dose (μSv) Range
^{222}Rn daughters	Detached house	28.0	10.2-564	1656	603 -33340
	Block of flats	8.7	3.8- 19.4	514	225 - 1147
^{220}Rn daughters	Detached house	1.3	0.2- 6.4	273	42 - 1345
	Block of flats	0.6	0.1- 2.8	126	21 - 589

Regarding thoron progeny inhalation, the annual effective doses are of 273 μSv and 126 μSv for detached house and block of flats, respectively.

The population-weighted averages of EEC of radon and thoron, the corresponding annual effective doses as well as collective doses and risk estimates, are given in Table 2.

Table 2. Radiation exposure of the Moldovian population from radon and thoron progeny inhalation

Source of exposre	Location	EEC* (Bqm^{-3})	Annual effective doses per capita (μSv)	Annual effective doses collective (man Sv)	Potential annual no of lung cancer
^{222}Rn daughters	indoors	20.2	1195	6214	311
	outdoors	4.6	90	468	23
Total radon			1285	6682	334
^{220}Rn daughters	indoors	1.0	200	1040	52
	outdoors	0.2	5	26	1
Total thoron			205	1066	53
TOTAL radon+thoron			1490	7750	387

*) population - weighted average

The values are of 20.2 Bqm^{-3} indoors and 4.6 Bqm^{-3} outdoors for EEC of radon and of 1.0 Bqm^{-3} and 0.2 Bqm^{-3} for EEC of thoron indoors and outdoors, respectively.

From these values and applying the dose coefficients, the annual effective dose per capita from inhalation of radon progeny was estimated at 1285 μSv and at 205 μSv from thoron progeny inhalation. From the total of 1490 μSv, more than 90 per cent are received indoors.

Taking into account the annual collective effective dose of about 7750 manxSv it is possible to estimate that 387 lung cancer

each year in Moldavian population (5.2 mill. inhabitants) may be attributable to radon and thoron short-lived decay products inhalation.

Radon and thoron progeny are the dominant source of human exposure to ionising radiation contributing about 60 per cent of the total dose from natural background of Moldavia, as Fig.1 illustrates.

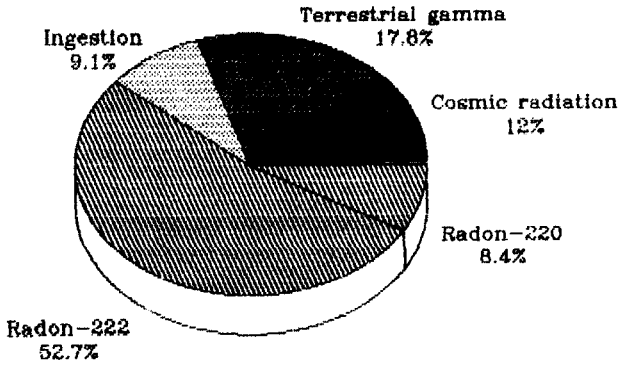

Fig.1 Contribution of natural radiation sources to annual effective dose

CONCLUSIONS

1. The population-weighted averages of EEC of radon and thoron are of 20.2 Bqm^{-3} and 1.0 Bqm^{-3} indoors and of 4.6 Bqm^{-3} and 0.2 Bqm^{-3} outdoors.

2. The annual effective dose per capita due to radon and thoron progeny inhalation are of 1285 μSv and 205 μSv, respectively.

3. The resulting annual collective dose of about 7750 manSv may be responsable for 387 potential lung cancers induced annually in Moldavian population.

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PAPER TITLE MEASUREMENTS OF RN-220 DECAY PRODUCT CONCENTRATIONS IN GERMAN DWELLINGS

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MAJOR SCIENTIFIC TOPIC NUMBER 1.1 (see page 7)

ABSTRACT (See instructions overleaf)

Continuous measurements of the potential alpha-energy of thoron progeny have been carried out in 23 dwellings in the south-east of Germany over time periods from two month to over one year. Mean values of the equilibrium equivalent concentration of thoron decay products were found to range from 0.1 to 1.0 Bq/m^3 with an overall average of 0.5 Bq/m^3.

Results show that there is a seasonal variation of the concentration with a minimum in summer. This can be attributed to increased ventilation in this time period.

Exposure from thoron progeny was compared to the exposure from radon progeny which was measured simultaneously. The contribution to the annual effective dose equivalent of the inhabitants by inhalation of radon progeny is about 5%.

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STUDIES OF ^{212}Pb STORM

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1. Introduction

Naturally occurring radon is estimated to be one of the most significant environmental radioactive substances(1). The main sources of natural radioactive radon are ^{222}Rn and ^{220}Rn. This paper presents investigations of ^{220}Rn daughters in the environment. The earth's crust contains trace amounts of ^{232}Th which decay to ^{220}Rn. ^{220}Rn is an alpha particle emitter. Its half-life is 56 s; and upon decay it emits a single alpha particle of energy 6.29 Mev. ^{220}Rn gas molecules leave the soil and enter the atmosphere. The concentrations of ^{220}Rn and their decay products vary according to the time because their information is due to different decay constant. It is set free in the atmosphere where it brings about a series of radioactive products which are rapidly fixed on the aerosol of the air. Although ^{220}Rn levels can be inferred from ^{220}Rn daughter measurement, measuring ^{220} directly avoids the problem of uncertainty in the extent of equilibrium between ^{220}Rn and its daughters.
^{212}Pb which reached its equilibrium state with its daughters in the air was measured around small uranium mines in Japan. Environmental. ^{212}Pb concentrations rose suddenly and reached a value ten times as high as usual values. These phenomena were observed many times during the past six years. We called these phenomena ^{212}Pb storms. Meteorological conditions lead to the variations of ^{220}Rn progeny concentrations(2-5). These phenomena have been studied in the point of meteorology.

2. Method

Kamisaibara village was selected as the investigation site. This village is situated in the central sector of the Chugoku mountains in Okayama Prefecture, Japan(35° 18′ N, 113° 35′E. about 710 m above sea level). This district is studded with small uranium mines which are of the sedimentary type. The geological structure of this area is comprised of quaternary gravel and the surface layer of this area is composed of granite. The location of the Kamisaibara region is shown in Fig. 1. Three monitoring stations were selected in this area. The monitoring stations were set up in the Toge, Akawase and Tennoh. Atmospheric ^{212}Pb and meteorological parameters were measured from January 1990 to December 1994. The meteorological parameters which were measured :atmospheric temperature, atmospheric pressure, precipitation, relative humidity, wind velocity, net exchange radiation. ^{212}Pb concentrations were determined by obtaining

an aerosol sample on an air filter, followed by measurement of the alpha activity on the filter after the end of sampling. To collect atmospheric aerosols, a high volume air sampler was used for a period of over 3 hours at suction speed 250l/min. After standing 7 hours at the end of air sampling, ^{212}Pb exists in equilibrium with ^{212}Bi and ^{212}Po. ^{212}Bi half-life is 60.6 min; and upon decay it emits an alpha particle of energy 6.05Mev. ^{212}Po half-life is 0.30μs; and upon decay it emits an alpha particle of energy 8.78 Mev. These alpha particles were detected by the use of a ZnS(Ag) detector(150 mm φ). The counting of alpha particles on the filter was carried out a period of over one hour. Eight samples were collected in one day.

3. Results and discussion

The yearly variations of atmospheric ^{212}Pb were presented in Fig. 2. The yearly mean values and the standard deviations are shown. In 1994, the yearly mean value was slightly high. The yearly fluctuations were small. The seasonal variations of atmospheric ^{212}Pb are presented in Fig. 3. The maximum mean value was shown in autumn and the minimum value was shown in winter. One reason for this may be that usually this district was covered with snow in winter. The mean values and the standard deviations at every station for five years are shown in Fig. 4. The mean values were slightly different.It may be presumed that these differences were caused by the environment of the setting point.The yearly maximum values for every station are presented in Table 1. Environmental ^{212}Pb concentrations rose suddenly and it reached a value ten times as high as the usual value. The maximum concentrations of ^{212}Pb for five years were shown at 594 mBq/m^3 on 28 Apr. 1992 in Akawase, 608 mBq/m^3 on 3 Nov. 1994 in Tennoh.The relationships of the maximum values for two cases and meteorological conditions at that time were investigated. In the case of the maximum values of ^{212}Pb concentrations at Akawase and Tennoh, atmospheric pressure was about 940 hPa/h. These values were equal to the approximate monthly mean value . Atmospheric pressure was not large affected to the maximum ^{212}Pb concentrations. As the maximum values of ^{212}Pb show, the relative humidities range from 90 to 95%. These values are the highest values in the month. These facts shows that the relative humidity strongly affected ^{212}Pb concentration. The relative humidity was positively correlated to ^{212}Pb concentration. Although the relative humidity reached above 90% in the case of precipitation, ^{212}Pb concentration was low. In daily fluctuation, the relative humidity was similar to that of ^{212}Pb concentrations. In general wind speed strongly affected ^{212}Pb concentration. As ^{212}Pb reached maximum value, wind velocity ranged from 0.3 m/s to 0.5 m/s. As the ^{212}Pb concentration reached maximum value, net exchange radiation ranged from -0.20 to -0.25 MJ/m^2/h. Net exchange radiation was inversely proportional to ^{212}Pb concentration.

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Table 1 Yearly maximum value of ^{212}Pb at every station (mBq/m^3)

Tohge	Time	Akawase	Time	Tennoh	Time
164	Jun.11,'90	573	Sep.10,'90	409	Jun.12,'90
130	Sep.21,'91	399	Apr.23,'91	538	Nov. 7,'91
169	Sep. 7,'92	594	Apr.28,'92	528	Sep. 3,'92
77	Sep.25,'93	483	Apr.28,'93	475	Apr.20,'93
127	Aug. 7,'94	494	Aug. 7,'94	608	Nov. 3,'94

IRPA9
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PAPER TITLE Ventilation Model for Elevated Radon Decay Product Levels Indoors

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Indoor concentrations of short-lived radon decay products (Rn-d) can reach levels over 1000 times that of outdoor levels and such elevated levels pose a significant risk of fatal lung cancer. High Rn-d levels indoors are generally dominated by ingress of soil gas of high Rn concentration and by ventilation characterized by poor outdoor air exchange - building materials and water making only minor contributions (with recognizable exceptions).

When considering the buildup of Rn and Rn-d in a building the source is Ra-226 in a soil-gas compartment which is successively (serially) connected to compartments of the building. There are losses from each compartment to the outdoor air which acts as a sink and to surfaces which act as sinks for the Rn-d; in the soil-gas compartment surfaces remove the Rn-d quantitatively.

The ventilation fluctuates and can coarsely be categorized into 'stagnant' situations and into wind activated situations. Modelling these situations, and changeovers thereof, provides an insight into the relative importance of the various parameters, such as the effect of the (?poorly known) soil-gas compartment.

Stagnant situations generally provide the higher concentrations whereas changeover situations can produce peak concentrations of short duration. Accurate short-term measurement of the activity ratios of Rn and individual Rn-d can indicate what ventilation situation prevails.

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Abstract No. PC 8770
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Author 20749
Acceptance
Mini-Presentation

PAPER TITLE CALIBRATION AND QUALITY ASSURANCE FOR THE MEASUREMENT OF RADON

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ABSTRACT (See instructions overleaf)

CALIBRATION AND QUALITY ASSURANCE FOR THE MEASUREMENT OF RADON

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In many countries measurement of 222-Rn in air is performed as a part of governmental radiation protection activities. In such screening programs, there is need for exact national primary standards. Additionally reliable secondary references should be available as regular quality assurance for radon measurement services. The author reports on his experiments in

- ☐ qualifying a new type of radon monitor, to be used as a reference machine for the calibration of other active and passive radon measurement systems
- ☐ cross-checking a set of five monitors in 15 different radon calibration facilities around the globe (International Calibration Experiment ICE'94)
- ☐ qualifying the instrument under US-EPA's RMP program and for the Federal Russian GOSSTANDARD as radon reference equipment
- ☐ establishing a "quick" field calibration procedure for routine QA, based on the NIST radon standard SRM 4968 and a portable calibration chamber

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PAPER TITLE

Radon measurements in the environment

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Radon is a natural radioactive gas. It is a chemically inert, highly radioactive daughter of uranium, thorium and actinium decay. A part of the radon formed in earth's crust (natural source) and in radioactive wastes storage sites (artificial source) can migrate to the atmosphere through cracks, fissures by transport mechanisms (diffusion, fluid convection). Thus, radon occurs naturally in every place and not in uranium mines only. It may pose a health risk for the population, because it is the first component of the average annual radiation exposure to the population.

For the past 10 years, we have been developing complementary techniques in the detection of radon and its decay products (proportional counter, silicon detector and solid state nuclear track detectors). A methodology has been developed to analyze the radon concentration in large and varied areas in using. This analysis is effected in time or in space. Thus, the measurements reveal the distribution of radon emanation and the range of variation. The method is used routinely for the following purposes: in homes in the aim of radioprotection, in geological applications for prospecting, studying the movement of the earth crust and analyzing the effect of alpha pollution in the environment.

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STOCHASTIC MODEL OF RADON DAUGHTER DEPOSITION AND CLEARANCE IN HUMAN BRONCHIAL AIRWAYS

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INTRODUCTION

Morphometric measurements of human airway casts have revealed that the human bronchial tree is an asymmetrically dividing network, exhibiting individual variations of airway dimensions within a given airway generation (1). Thus, a statistical correlation exists between the linear dimensions of a given parent airway and those of the asymmetrically dividing daughter branches (2). This statistical relationship, however, is constrained by correlations among various geometrical parameters, i.e., the human lung is not a fully stochastic system. From a statistical analysis of these morphometric data, the following geometrical parameters could be obtained for each airway generation or bifurcation: (i) probability distributions for diameters and lengths, ratios of parent cross section to the combined cross section of both daughters, ratios of major to minor daughter diameters, and branching angles for major and minor branches; (ii) correlations of diameters and lengths; and (iii) the probability of terminating the bronchial region as a function of both diameter and generation number (2). Since the original lung morphometry refers to total lung capacity (1), airway diameters and lengths were scaled down to functional residual capacity (3).

STOCHASTIC BRONCHIAL DEPOSITION AND CLEARANCE MODEL

The stochastic deposition model used in the present study refers to the transport of inhaled particles through this stochastic lung structure by selecting randomly the sequence of airways for each individual particle, applying Monte Carlo techniques (4). For the computation of particle deposition within a selected airway bifurcation, however, the average deposition behavior of many particles is assumed, i.e., deposition probabilities are calculated by the commonly used deposition equations (4). In other words, the stochastic deposition model used here reflects the effects of a stochastic airway system on particle transport and deposition. In this model, deposition is computed for individual airway bifurcations (consisting of half of the parent and the major and minor daughter lengths) in a given lung lobe, considering the filtering effect of nose or mouth (5). The stochastic deposition model has been validated by comparison with total and regional deposition data in human test subjects for a wide range of particle sizes (6).

In the present study, a bronchial mucociliary clearance model has been implemented into the above described stochastic deposition model, based on the following assumptions: (i) the average mucus velocity in the trachea is assumed to be 5.5 mm min^{-1} (7); (ii) at a given airway bifurcation, average mucus velocities in the asymmetrically branching daughter airways are proportional to their respective diameters (7); (iii) mucus transport is delayed at central bifurcation zones (delay time Δt is uniformly distributed between 0 and 30 min) (8); and, (iv) the slow bronchial clearance fraction, f_s, with a half-life of about 10 days, decreases in a linear fashion with the particle's diameter, d, according to the equation: $f_s = 0.77 - 0.12\ d\ (\mu m)$ (9).

RESULTS

Experimental data on particle retention in the lungs after shallow bolus inhalation for 1.7, 3.2. and 6.7 μm geometric diameter particles are plotted in Figure 1 (10) (note: the volumetric front depth of 55 cm^3 refers to a bolus inhalation time of 3.78 s during a 4 s inspiration time, with an initial dispersion of 0.1 s). Except for the first four hours, in which the stochastic clearance simulations predict a higher retention than has been observed in the experiments, the agreement between experiment and model predictions is quite satisfactory, considering potential effects of inter-subject variability in lung dimensions and structure.

In the simulations presented here, bronchial deposition patterns, normalized to the number of particles entering the trachea are computed for 0.2 μm unit density particles under resting breathing conditions, i.e., for a tidal volume of 1000 cm^3, a cycle time of 4 s (2 s inspiration, 2 s expiration, no breath-hold period), and uniform inhalation during the 2 s inspiration time. The initial activity of the inhaled radon decay products is 9 Bq (^{218}Po), 6 Bq (^{214}Pb), and 4 Bq (^{214}Bi/^{214}Po), representing typical decay product ratios in domestic environments (8).

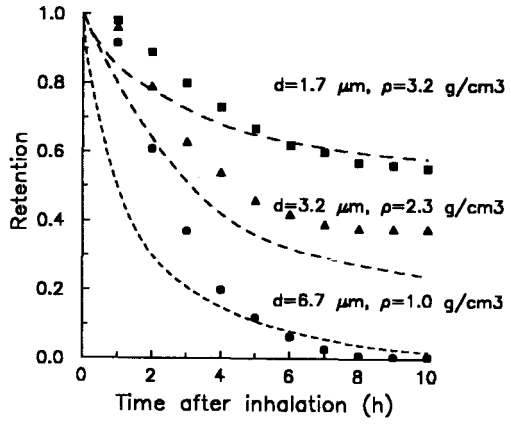

Figure 1. Particle retention as a function of time after shallow bolus inhalation for three different particle sizes. The symbols denote the theoretical predictions, while the fits to the measured data (10) are represented by the broken lines.

The subsequent upstream transport of the deposited radon progeny by mucociliary clearance is simulated for the same pathway which was selected upon inspiration, considering their radioactive decay. The transit times of the deposited particles in different bronchial airway generations, illustrating the effect of mucucs clearance, together with the initial deposition pattern are displayed in Figure 2. Because of the presumed relationship between mucus velocity and airway diameter, the decrease of diameters in peripheral bronchial airway bifurcations leads to increasing transit times. In the most distal genenerations, however, transit times decrease again due to the decreasing number of bronchial airways in these generations (note: in contrast to the commonly used symmetrical models, still 0.6 % of all airways in bifurcation 20 are terminal bronchioles).

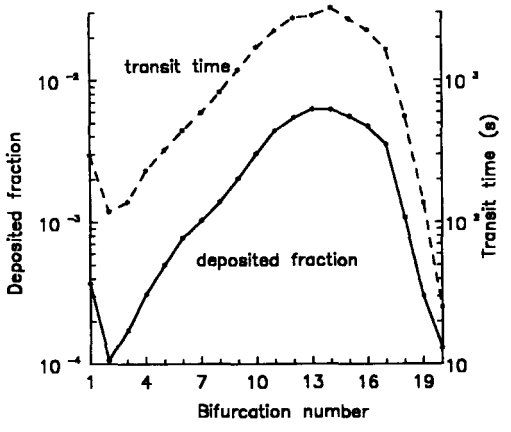

Figure 2. Initial deposition pattern for 0.2 μm unit density particles under resting breathing conditions, and the transit times of these particles in individual bronchial airway bifurcations due to mucociliary clearance.

The total number of radioactive decays, illustrating the combined effect of deposition, clearance and radioactive decay, and the number of radioactive decays per unit surface area, representing the radiation dose distribution, are plotted in Figure 3 as functions of the bifurcation number for both alpha-emitting radon progeny (only attached ^{218}Po nuclides have been considered here). Because of their relatively short half-lives,

the distributions of both nuclide transformations reflect the transit time distribution. On the other hand, the distribution of the number of nuclear transformations per unit surface hardly changes with penetration into the bronchial tree due to the larger total surface area in peripheral airway generations. Above about bifurcation 13, there is no further increase in total surface area due to a decreasing fraction of bronchial airways and the surface activities, therefore, drop parallelly to the decay pattern.

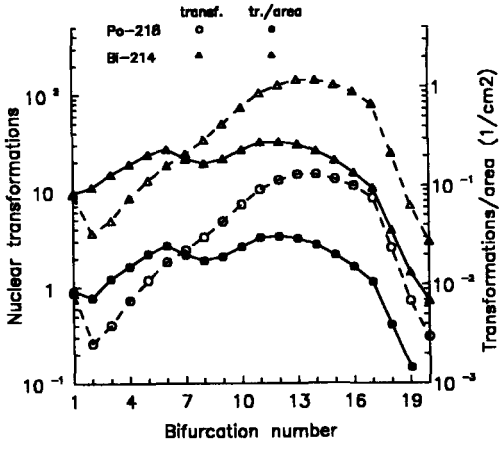

Figure 3. Number of nuclear transformations and number of decays per unit surface area in individual bronchial airway bifurcations for inhaled radon progeny with initial activities of 9 Bq (^{218}Po), 6 Bq (^{214}Pb), and 4 Bq (^{214}Bi/^{214}Po).

CONCLUSIONS

The results of the stochastic deposition and clearance calculations presented here reflect primarily the intra-subject variability of the human lung morphometry. In other words, the random walk of individual particles within a given bronchial bifurcation during inspiration and, if deposited, during mucociliary clearance has not yet been considered due to the lack of pertinent experimental information. Future incorporation of the random location of target cells in bronchial epithelium into the present stochastic deposition and clearance model will allow us to quantitate the effect of individual biological variability in cellular dose.

ACKNOWLEDGEMENTS

This research was supported by the Jubiläumsfonds of the Austrian Nationalbank (Project No. 4626) and by the Commission of the European Communities (Contract F13P-CT92-0064).

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THE NATIONAL STANDARD FOR ^{222}Rn ACTIVITY MEASUREMENTS DEVELOPED AT ENEA IN ITALY

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INTRODUCTION

The growth of interest in radon measurements and the increasing importance of laboratory quality assurance programmes have caused a growing request of calibration of radon measuring instruments. To this purpose radon calibration equipment and testing facilities have been developed in many countries and recommended sampling and measurement protocols are being studied by national and international standardisation organisations.
In Italy, the Istituto Nazionale di Metrologia delle Radiazioni Ionizzanti of ENEA, the Italian Primary Standard Institute for ionising radiation quantities, has developed a set of equipment to satisfy the national request of radon calibrations. These equipment are traceable to the ENEA national standard of ^{222}Rn through an interrupted chain of transfer instruments. The ^{222}Rn national standard is derived from the national standard of ^{226}Ra and it is made by a ^{222}Rn Reference Measurement System (RMS) and a ^{222}Rn Transfer System (TS). The ENEA ^{222}Rn standard has been compared with other ^{222}Rn national standards in the frame of two EUROMET (European collaboration in measurement standards) intercomparisons carried out in 1992 and in 1994 (1). The system of experimental equipment developed at ENEA and the relation to the national standard are reported in figure 1. This paper briefly describes this system of experimental equipment.

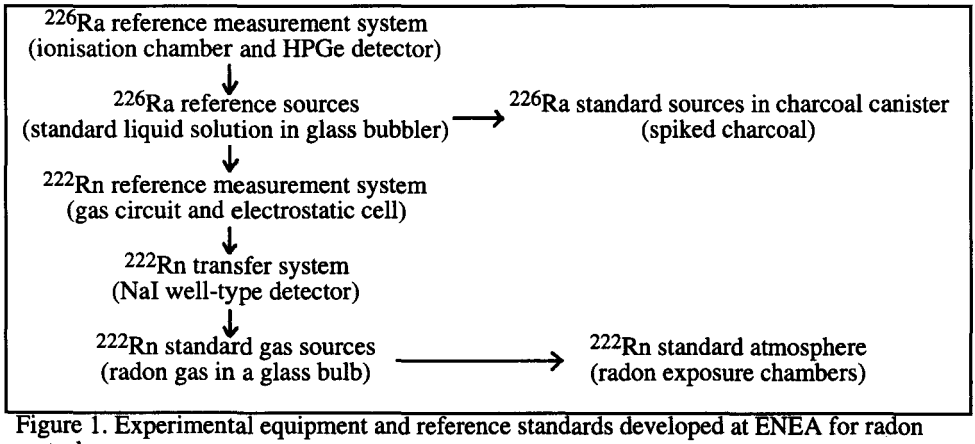

Figure 1. Experimental equipment and reference standards developed at ENEA for radon metrology.

THE ^{222}Rn REFERENCE MEASUREMENT SYSTEM

The ^{222}Rn RMS is used to relate the activity of ^{222}Rn gas samples to the activity of a ^{226}Ra reference source by the physical separation of the ^{222}Rn produced in the ^{226}Ra decay and its transfer into a counting chamber.
The ^{226}Ra reference source consists of a 5 kBq liquid standard solution (0.8 M HCl, 3.8 mg of BaCl2 per gram of solution) with volume of about 40 cm^3 sealed in a glass bubbler. The reference source was calibrated at ENEA by the ^{226}Ra reference measurement system. This

system is used to maintain the national ^{226}Ra standard and it is based on a well-type ionisation chamber used for activity measurement of gamma-ray emitting nuclides (2). The ^{226}Ra calibration factor of the ionisation chamber was determined by a ^{226}Ra standard source, traceable to the 1934 Hönigschmid international mass standard of radium and supplied by the National Institute of Standards and Technology. The ionisation current was corrected for the contribution of ^{210}Po the activity of which was determined by HPGe X-ray spectrometry. The combined standard uncertainty of the activity of the ^{226}Ra reference source was 0.63% (1s).

The ^{222}Rn RMS is a gas transfer and circulation system made of a gas input/output unit, a radioactive source section, a circulation and dehumidification unit and an electrostatic cell with a Si detector. The overall volume of the circuit is 3.88 dm^3, the volume of the electrostatic cell alone being 3.72 dm^3. The activity of ^{222}Rn gas sources in spherical glass bulbs (35 cm^3) is measured in the RMS by a particular procedure that does not require volume determinations or any assumption of total radon removal from the liquid ^{226}Ra solution (3). The radon gas, either from gas samples or from the ^{226}Ra reference source, with ^{222}Rn in equilibrium with ^{226}Ra, is transferred from the source section into the whole circuit by slowly flushing and continuously circulating a carrier gas through the circuit. The ^{218}Po count rate of the electrostatic cell is used for monitoring the radon activity in the circuit. Under reproducible extraction conditions the cell response is proportional to the overall radon activity in the circuit and hence it is used to relate the activity of radon gas samples in glass bulbs to the activity of the ^{226}Ra reference source. The conbined standard uncertainty of the efficiency value of the RMS is 0,98% with the main contribution coming from the calibration of the ^{226}Ra solution (0.63%) and the reproducibility of the transfer procedure (0.75%).
Unfortunately the direct calibration of radon gas samples with the RMS is based on a destructive procedure for the gas sources itself.

THE ^{222}Rn TRANSFER SYSTEM

The ^{222}Rn Transfer System (TS) was developed at ENEA for the non-destructive calibration of ^{222}Rn gas sources.

The ^{222}Rn activity of gas sample bulbs (1 to 10 kBq) is determined by measuring the source photon emission rate by a NaI(Tl) well-type detector. The NaI(Tl) crystal has an external diameter of 10.5 cm and a thickness of 14.5 cm. The well diameter is 7.5 cm and the well height is 10.5 cm. The well is suitable for containing the 35 cm^3 glass bulb. Similar systems have been developed and described by other authors (1-4). The count rate in the gamma energy window from 148 to about 1360 keV is measured to determine the ^{222}Rn activity. This energy window includes the photon emissions produced in the decay of ^{222}Rn daughters and their sum peaks. The NaI(Tl) detector was calibrated by using a set of gas sources that, after counting on the TS, were calibrated by the RMS.

The additional component of uncertainty introduced by the TS on the calibration of ^{222}Rn gas sources in glass bulbs is just due to the TS-response reproducibility (0.2%). The combined standard uncertainty in the calibration of these sources is then about 1.2% (1s).

As the TS is used at ENEA for routine calibration of radon gas sources, including the sources used for filling the radon chambers, the stability and the reproducibility of the system response is carefully checked by use of a solid ^{226}Ra sealed source. The TS has been also characterised in terms of several additional parameters such as the working range and the linearity of the system response (5).

STANDARD SOURCES FOR INSTRUMENT CALIBRATION

The calibration service provided by ENEA include the preparation of standard sources for radon measurements. In particular the following sources are available (figure 1):

- ^{226}Ra liquid solutions in glass ampoules or glass bubblers;
- ^{226}Ra standard sources in charcoal canisters;
- ^{222}Rn gas sources in 35 cm^3 glass ampoules.

The sources in charcoal canisters are prepared by spiking a charcoal matrix with a ^{226}Ra standard solution. The spiking procedure was specifically developed at ENEA for the preparation of reference materials for gamma-ray spectrometry (6). The ^{222}Rn gas sources in glass bulbs are utilised for the calibration of several radon measuring instrument equipped with a counting chamber, such as scintillation cells or ionisation chambers, by directly transferring the radon gas from the bulb into the counting chamber. In other cases the instrument calibration is carried out in standard atmospheres created in suitable exposure chambers.

THE ENEA RADON CHAMBERS

In order to allow the calibration of the different measuring equipment utilised in the environmental radon surveillance, two radon chambers were developed at ENEA (7-8). These radon chambers have different volumes (0.1 m^3 and 1 m^3, respectively) and different possibilities of climatic parameters control. Both the exposure chambers are equipped with suitable instruments for monitoring the radon concentration and the climatic parameters. The calibration of the ^{222}Rn monitor instruments has been carried out by transferring known amount of radon gas from glass bulbs into the radon chambers.

The 1 m^3 chamber is working at ENEA since 1985. Standard atmospheres are created in this chamber with control of the following parameters: temperature, relative humidity, activity concentration of ^{222}Rn and its decay products, aerosol size distribution and concentration. The facility can operate both in static and dynamic condition. Under static condition a known radon activity is transferred into the chamber from a standard gas source. At dynamic condition the radon concentration is maintained constant at a given level by continuously flushing the carrier gas through an external ^{226}Ra source. This radon chamber is utilised for the calibration of radon measurement systems that require the control of climatic parameters and for the calibration of devices, such as continuous and integrating passive monitors, for which the simple transfer procedure described above is not applicable.

The 0.1 m^3 chamber is working since 1992 and it is used for the calibration of nuclear track detectors and other small volume devices. In this chamber ^{222}Rn activity concentrations higher by an order of magnitude, with respect to those routinely used in the 1 m^3 chamber, can be obtained.

CONCLUSIONS

A complete system of experimental equipment utilised for calibration of radon monitoring devices has been developed at ENEA. The system is based on the ^{222}Rn national standard developed and maintained at ENEA. The work in progress concerns the completion of the calibration of the radon chambers and the development of new national standards related to radon monitoring such as the radon-in-water and the radon flux-density standards.

A new project consists in the development of a radon chamber with volume of about 50 m^3 for calibration of radon daughters measuring instruments and for intercomparison exercises to be organised by ENEA for the quality assurance of the laboratories belonging to the national radioactivity surveillance network.

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TIME INTEGRATED MEASUREMENT OF POTENTIAL ALPHA ENERGY (PAE) DUE TO SHORT LIVED DECAY PRODUCTS (SLDP) OF RADON 220 and 222 IN THE ENVIRONMENT

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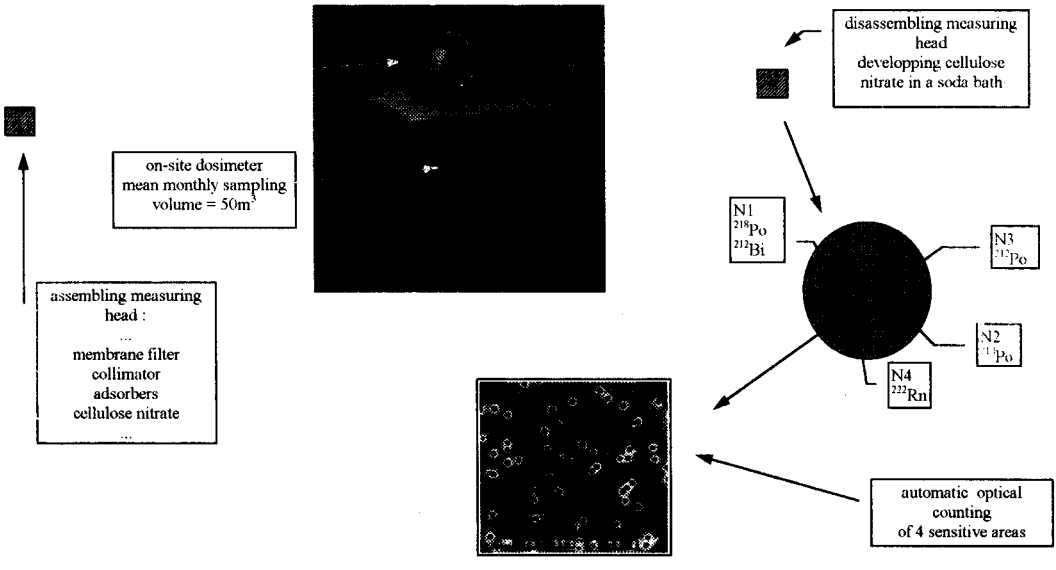
INTRODUCTION

The on-site alpha dosimeter, which provides time-integrated measurement of PAE due to SLDP of ^{220}Rn and ^{222}Rn in the environment is a derivative of the well-known individual alpha dosimeter used since 1983 for the monitoring of workers in french uranium mines It includes a sampling unit and an interchangeable measuring head, and while sampling on the incorporated membrane filter, the detection of alpha particles from SLDP of ^{220}Rn and ^{222}Rn is carried out following energy discrimination using a solid state nuclear track detector, of the type KODAK LR115 .

In practical terms, we can give as an « use range » of measurement for the on-site dosimeter an interval between 5 nJm^{-3} and 5.10^4 nJm^{-3} ($PAE^{222}Rn$) for a monthly sampling of 50 m^3 . The setting-up of calculation for the uncertainty estimation for one single measurement result, including the whole measurement chain, shows that for this interval the total relative standard deviation is less than 40% .

It's good metrological performance, the great density of informations it can provide, and the simplicity of use make this device a usefool tool : not only to make small investigation of PAE monitoring in houses, in radium waste storage environment or to study specically night/day PAE ratio nearby uranium site for example, but also to follow up with a dosimeter network the evolution of the radiological impact of uranium or thorium facilities . At present, more than 150 on-site dosimeters are in use in France, Africa and others .

SIMPLIFY DIAGRAM OF USING PRINCIPLES



PAE CALCULATION FROM RAW OPTICAL DATA

Knowing the physical constant ratio of ^{212}Bi desintegration (α:36% and β/α:64%)

$$PAE_{(thoron)} = K\,(8.7\,N3 + 6.08\,N3/2) / \rho V \quad in\ nJm^{-3}$$

$$PAE_{(radon)} = K\,(7.7\,(N2 - a\,N4) + 5.99(N1 - N3/2 - b\,N4)) / \rho V \quad in\ nJm^{-3}$$

with ρ : ***combination of collection and detection efficiency***
V : ***sampling volume***
a, b : weighting factors of radon contribution
K : International System unit factor

METROLOGICAL FEATURES OF ON-SITE DOSIMETERS

Range measurement

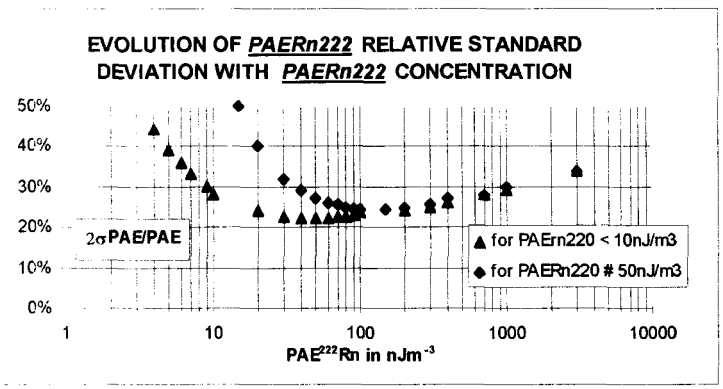
Theoretical aproach use to estimate the uncertainty of one single measurement results based on the simple propagation of statistical errors is a bit complicated by the fact that we have got some linked parameters . Nevertheless this calculation is essential to have got a good estimation of the lower end range measurement of the system .

The global formulae presented below represents the basis of this calculation and has been used to obtain the simplified curves which show the expexted evolution of relative standard deviation with PAE results .

$$\sigma_f^2 = \left(f_x' \sigma_x\right)^2 + \left(f_y' \sigma_y\right)^2 + \left(f_z' \sigma_z\right)^2 + \Lambda + 2 f_x' f_y' \operatorname{cov}(x,y) + 2 f_x' f_z' \operatorname{cov}(x,z) + 2 f_y' f_z' \operatorname{cov}(y,z) + \Lambda$$

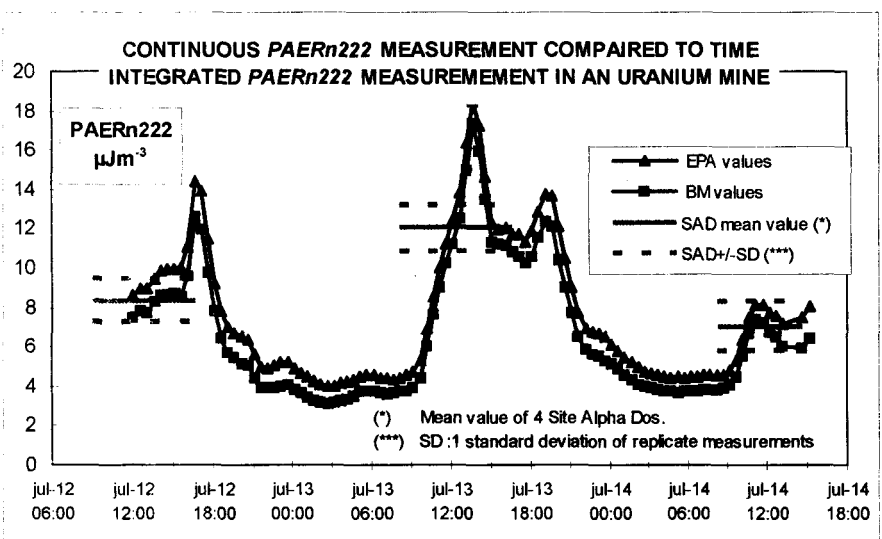
where the variable parameters will be in our case :

- the number of tracks N1, N2, N3, N4 counted on each sensitive areas
- the experimental collection efficiency
- the calculated by monte carlo algorithme detection efficiency
- the real-time measured sampling volume



Calibration

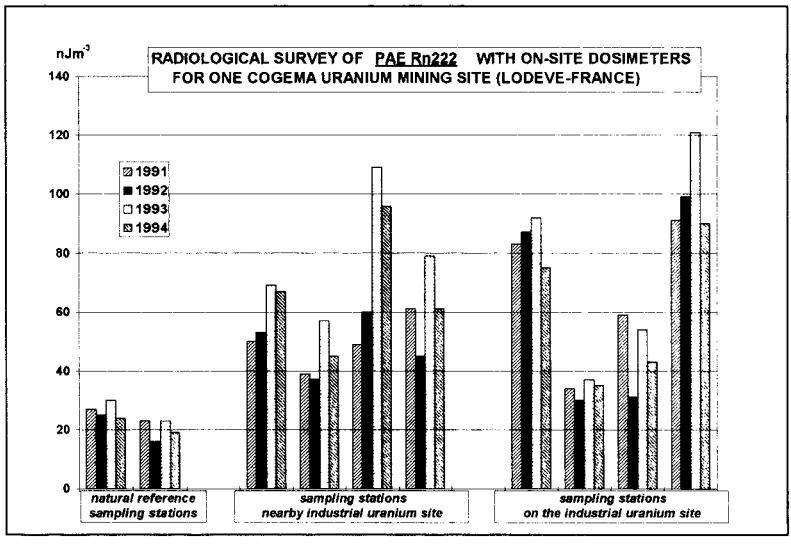
In addition to internal calibration, we try to compair our methods and results with other ones to follow an always possible time-drift . Some results of the last International Intercomparison we have participated concerning PAE measurements with Uranium mines conditions are presented below :



EXEMPLES OF USES

radiological following of uranium mining site

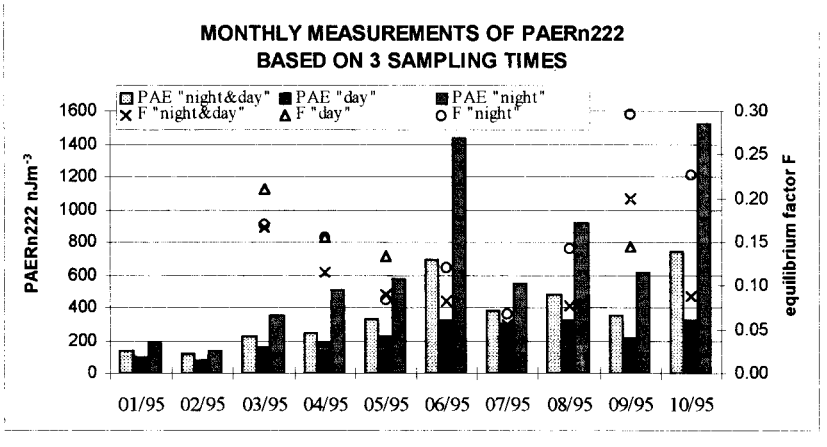
For example in south of France,there is a dosimeter network of about 10 devices measuring PAE around and on one of COGEMA's uranium mining site . Then we can follow since few years the radiological impact on the environment and by the way we can have a very good estimation of PAE exposure of the defined critical group . The diagram below is just a small example of results obtained among all dosimeters installed today.



specific application

In order to have more informations not only about the monthly but also night/day fluctuations of PAE and consequently to better estimate the real impact for the public, we have installed in this way for about 1 year 3 dosimeters sampling at different times during the day . In the same time, one continuous radon monitor (BARASOL) has been set , then permitting us to calculate with PAE and radon concentation the time-integrated equilibrium factor for specific periods .

The diagram below shows seasonaly, monthly, mean night and day fluctuations of PAE in a valley just nearby a radon source, an uranium tailing .



IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE A Method For Assessment Of Environmental Radon Daughter Exposure Using Active Radon Daughter Monitors

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 4.1 or 4.2

ABSTRACT (See instructions overleaf)

WMC (Olympic Dam Corporation) is a copper and uranium producer in the arid region of South Australia. To date, two mathematical models have been used to determine the incremental impact of the operations on the natural environmental radon daughter concentrations. The models are based on measured sources of radon and radon daughters from the process and measured meteorological conditions. Although the modelling approach is scientifically sound, it is limited because it does not take into account field measurements of radon daughter concentrations.

A method has been developed which combines real time environmental radon daughter concentration measurements from a number of sites around the operations and real time meteorological data to assess the impact. The method essentially determines if a particular site is affected by the operations by considering the wind direction. When there is no impact at a monitoring site, as determined from the wind direction, then that site is considered to be measuring a background level. The background levels from the unaffected sites are then averaged and subtracted from affected sites to determine the impact.

This paper will present the new method which confirmed the results of the modelling, thereby giving more confidence to the existing impact assessment process.

SIMULTANEOUS AND AUTOMATIC DETERMINATION OF Rn-222 DAUGHTERS AND Pb-212 CONCENTRATIONS IN THE ATMOSPHERE USING TWO α SCINTILLATION COUNTERS

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ABSTRACT : The method using a monitoring instrument having two α scintillation counters is devised for determining short-lived ^{222}Rn daughters and ^{212}Pb in the atmosphere simultaneously and automatically every 3h. Both levels of these concentrations measured by this method are the same as those observed elsewhere on land.

INTRODUCTION

There are a few reports on simultaneous measurement of ^{222}Rn and/or short-lived ^{222}Rn daughters (the term 'short-lived ^{222}Rn daughters' will hereafter be denoted by '^{222}Rn daughters') and ^{212}Pb in outdoor air using scintillation counters. Druilhet and Fontan [1] measured ^{222}Rn, ^{222}Rn daughters and ^{212}Pb continuously. Nishikawa [2] measured ^{222}Rn daughters and ^{212}Pb continuously. It is very difficult to estimate errors of their concentrations obtained by these method. Fujinami [3] measured ^{222}Rn daughters and ^{212}Pb every 6 h. An interval of 6 h is too long to estimate diurnal variation of their concentrations.

We devise a method using a monitoring instrument having two α scintillation counters for determining ^{222}Rn daughters and ^{212}Pb in the outdoor air every 3 h. This method enables us to measure them automatically for 6 months.

MEASURING INSTRUMENT AND CALCULATION

We remodel a ready-made instrument (Fuji Electricity Co. Ltd., type NAD 23803) for measuring the α-rays of ^{222}Rn daughters and ^{212}Pb daughters. A block diagram of the remodelled instrument is shown in Fig. 1. Air is drawn through a filter of cellulose (Advantec Toyo Co. Ltd., type HE-40T) from an intake at 2 m above the air-ground interface (the first position in Fig. 1). Effective area of the filter for the suction is 0.0023 m^2. After sucking the air for 3 h at a rate of 15 m^3h^{-1}, the part of the filter through which the air is drawn moves from the first position to the second position, the part situated at the second position moves to the third position, and the part situated at the third position moves to the fourth position (see Fig. 1). Then, the α-rays of the radionuclides collected on the filter are measured using two scintillation counters (ZnS and photomultiplier). Since the movement of the filter requiers 2 min, the first measurement starts 2 min from the end of the suction (the first position in Fig 1). This measuring time is 58 min. 7 h from the end of the suction, the second measurement is made for 1 h (the fourth position in Fig. 1).

The distance between the filter and the

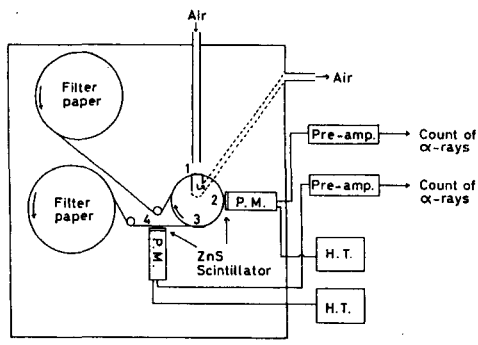

Figure 1. Schematic diagram of measuring instrument.
1 : First position
2 : Second position
3 : Third position
4 : Fourth position

counter is 2 X 10^{-3} m.

In the second measurement, ^{212}Pb is in radioactive equilibrium with its daughters. Therefore, ^{212}Pb concentration A_{Pb} (Bqm^{-3}) is represented by

$$A_{Pb} = \lambda_1\{\varepsilon qG(T_c, T_3, T_4)\}^{-1}C_2 \quad (1)$$

with

$$G(T_c, T_3, T_4) = \frac{\lambda_2}{\lambda_1(\lambda_2-\lambda_1)}(1-e^{-\lambda_1 T_c})(e^{-\lambda_1 T_3}-e^{-\lambda_1 T_4}),$$

where λ_1, λ_2 are the respective decay constants of ^{212}Pb and ^{212}Bi (s^{-1}), ε is the total efficiency for the α-rays, q is the flow rate of air passing through the filter (m^3s^{-1}), T_c is the collecting time (s), T_3 is the time interval from the end of the sampling to the start of the second measurement (s), T_4 is the time interval from the end of the sampling to the end of the second measurement (s), C_2 is the count of the α-rays obtained by the second measurement.

In the first measurement, ^{222}Rn daughters are assumed to be in the radioactive equilibrium in the outdoor air. ^{222}Rn daughters concentration A_{Rnd} (Bqm^{-3}) is represented by

$$A_{Rnd} = \lambda_3[\varepsilon q\{F_1(T_c, T_1, T_2)+F_2(T_c, T_1, T_2)\}]^{-1}(C_1-C_2') \quad (2)$$

with

$$F_1(T_c, T_1, T_2) = \frac{(1-e^{-\lambda_3 T_c})(e^{-\lambda_3 T_1}-e^{-\lambda_3 T_2})}{\lambda_3}$$

and

$$F_2(T_c, T_1, T_2) = \frac{\lambda_4\lambda_5}{\lambda_3(\lambda_4-\lambda_3)(\lambda_5-\lambda_3)}(1-e^{-\lambda_3 T_c})(e^{-\lambda_3 T_1}-e^{-\lambda_3 T_2}) - \frac{\lambda_3^2\lambda_5}{\lambda_4^2(\lambda_4-\lambda_3)(\lambda_5-\lambda_4)}$$
$$X(1-e^{-\lambda_4 T_c})(e^{-\lambda_4 T_1}-e^{-\lambda_4 T_2}) + \frac{\lambda_3^2\lambda_4}{\lambda_5^2(\lambda_5-\lambda_3)(\lambda_5-\lambda_4)}(1-e^{-\lambda_5 T_c})(e^{-\lambda_5 T_1}-e^{-\lambda_5 T_2}),$$

where λ_3, λ_4 and λ_5 are the respective decay constants of ^{218}Po, ^{214}Pb and ^{214}Bi (s^{-1}), T_1 is the time interval from the end of the sampling to the start of the first measurement (s), T_2 is the time interval from the end of the sampling to the end of the first measurement (s), C_1 is the count of the α-rays obtained by the first measurement, C_2' is the count of the α-rays estimated from the second measurement, which is due to ^{212}Pb daughters during the first measurement. C_2' is calculated on the assumption that the ratio of ^{212}Bi concentration A_{Bi} (Bqm^{-3}) to the ^{212}Pb concentration is 0.7 in the outdoor air (A_{Bi}/A_{Pb} ranges from 1.0 to 0.4 [4] in the outdoor air, we adopt the average of these two values) and that ^{212}Bi is in radioactive equilibrium with its daughters.

There are many reports on the deviation from the radioactive equilibrium among ^{222}Rn daughters. Even if the ratios of the ^{214}Pb and ^{214}Bi concentrations to the ^{218}Po concentration are 0.5 and 0.3 respectively, the average concentration of these three radionuclides is only +20 % as compared to the concentration calculated by the assumption of the equilibrium.

EXAMPLE OF RESULTS OBTAINED BY THE MEASUREMENT

The measurement is made at Kamisaibara-mura (35°18'N, 133°35'E) in Japan. The ^{222}Rn daughters concentration ranges from 0.29 to 9.61 Bqm^{-3} and the ^{212}Pb concentration does from 0.006 to 0.433Bqm^{-3}. The average values of the ^{222}Rn daughters and ^{212}Pb concentrations are 3.23 and 0.125 Bqm^{-3} respectively. These values are observed elsewhere on land for the

^{222}Rn daughters and ^{212}Pb concentrations. The ratio A_{Pb}/A_{Rnd} ranges from 0.008 to 0.089, and its average value is 0.032 (the ratio of the average value of A_{Pb} to the average value of A_{Rnd} is 0.039). This average value is a little smaller than the world average of a few available values [5, Annex D, Table 23], which is 0.04. The ^{222}Rn concentration in the Table 23 shows the ^{222}Rn daughters concentration defined in this article.

We show here an example of the results measured for a week, but this instrument permits us to measure the ^{222}Rn daughters and ^{212}Pb concentrations automatically without exchange of the filter for 6 months.

MEASURABLE ZONE

Figure 2 shows the zone where the ^{222}Rn daughters and ^{212}Pb concentrations can be measured simultaneously with this instrument. This zone is enclosed with the strait lines I and II and the curved line III. If the ratio A_{Pb}/A_{Rnd} is larger than the strait line I (2.67×10^{-4}), the contribution of the ^{222}Rn daughters to the second measurement can be neglected. The strait line II indicates merely the lower limit of detection of ^{212}Pb. The curved line III indicates the ratio A_{Pb}/A_{Rnd} estimated from the following equation,

$$C_1-C_2' = 3\sqrt{C_1+C_2'+(C_2'-C_2'')^2}. \quad (3)$$

where C_2'' is the count of the α-rays estimated from the second measurement, which is due to ^{212}Pb daughters during the first measurement. C_2'' is calculated on the assumption that ^{212}Pb is in radioactive equilibrium with its daughters. Considering the variations of the ^{222}Rn daughters and ^{212}Pb concentrations, this zone includes 14 points out of 16 in UNSCEAR Report [5, Annex D, Table 23] where the ^{222}Rn daughters and ^{212}Pb concentrations were measured. Therefore, this method enables the simultaneous measurement of these concentrations over considerably vast extent of land. And, having high sensitivity, this instrument is useful to measure simultaneously the low concentrations of the ^{222}Rn daughters and ^{212}Pb in the open air.

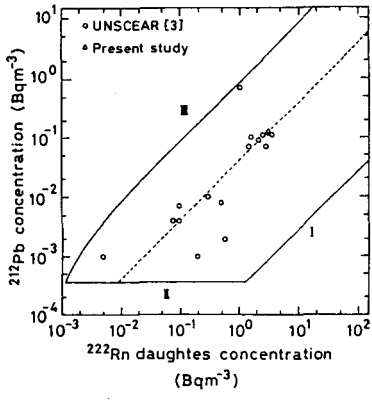

Figure 2. Zone where simultaneous measurement of ^{222}Rn daughters and ^{212}Pb is possible.
----- : A_{Pb}/A_{Rnd} = 0.04

Acknowledgement: The authors thank Prof. M. Shimo of Gifu College of Medical Technology for his technical aid in calibration of this instrument.

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BEHAVIOR OF RADON AND ITS PROGENY AT WORKING PLACE

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INTRODUCTION

Recently the ICRP recommended that steps be taken to reduce radon not only in the home environment but also in the workplace (1). Although nation-wide surveys of radon concentrations have been often carried out in dwellings, very few have been conducted in the workplace. If we consider that workers spend more than 1/3 of their day on the job, workplace surveys are just as significant as those in dwellings. Radon concentrations in several offices were measured with passive devices. Due to the use of air conditioner and the habits of workers, radon concentrations during the actual working period may differ from the averaged radon concentrations throughout a day. Most offices make use of one of two types of air conditioners. With the first, air is re-circulated in the room. Radon concentration remains stable whereas radon progeny concentrations may vary drastically. With the second, outside air is partially injected into indoors and consequently the radon concentration as well as radon progeny may vary due to the effect of fresh air entry. For the dose assessment, determination of parameters such as equilibrium factor(F) and unattached fraction(f_p) is also needed as well as radon concentrations. Therefore, the equilibrium factor and unattached fraction were continuously measured in two places with an active device. The results of these measurements in workplaces were presented in the present paper.

MATERIALS AND METHODS

Radon measurements with passive devices were carried out at five offices which were located near Shinjuku station in Tokyo. Two devices were placed at each site. One provided the average radon concentration over a 24 hour period. The other provided the average radon concentration during working hours. An electrostatic integrating radon monitor developed by T. Iida et al. was used in this survey (2). Although CN film is used in the ordinary monitor, CR-39 was chosen for its stable sensitivity. Its detection limit is below 1 Bq m^{-3} for an exposure time of 2 months. Since most of ^{218}Po atoms are positively charged, they are collected on the negative electrode of the monitor (-300V) (3). In order to control the period of applying a voltage to the monitor, a weekly timer was incorporated into the monitor. The voltage applying period was set from 11:00 to 19:00 taking into consideration the air exchange rate of the monitor (4). The monitor with the timer measure the average radon concentration from 9:00 to 17:00, Monday to Friday. CR-39 were replaced every 2 months for more than 1 year.

So as to measure the equilibrium factor and unattached fraction, an active device developed by S. Tokonami et al. was used (5). This device consists of three parts: an unattached progeny monitor, an attached progeny monitor, and a radon gas monitor, all connected in series. When air is introduced into the device, unattached radon progeny is collected on a metal wire screen (JIS 300 mesh, openings: 118.2 cm^{-1}, wire diameter: 3.75×10^{-3} cm), and its activity is detected simultaneously by a silicon semiconductor detector (6,7). Remaining unattached and whole attached progeny are collected on a fiberglass filter(Whatman GF/F). ^{218}Po concentration and equilibrium equivalent radon concentration(EER) of the unattached and attached fractions were determined by the build-up method (8,9). Radon gas flows into a cylindrical vessel(decay chamber: inner volume of 7 liters) through a drying column filled with $CaCl_2$. Since radon progeny formed in the decay chamber are positively charged, they are collected on the detectable surface of a silicon semiconductor detector with an electric field(-4500V). The radon concentration is estimated from the net count of the collected ^{218}Po. This device can simultaneously provide the equilibrium factor and unattached fraction every 30 minutes. The minimum detectable concentration, derived from counting data in which the relative standard deviation is assumed to be 50%, is 3 Bq m^{-3} for radon concentration and below 1 Bq m^{-3} for both ^{218}Po concentration and EER.

RESULTS AND DISCUSSION

Figure 1 shows the average radon concentrations in the five offices. The radon concentrations varied

nd some relative high radon concentrations were observed. From these limited measurements, it was found that he radon concentrations during working hours were almost the same as those for all day average. Since re-circulating air conditioners were used in each of the offices, it is possible that the radon concentrations were stable whereas radon progeny concentrations varied. Thus the equilibrium factor and unattached fraction during the working period would more than likely be different from the daily average. By using an active device, the authors obtained data for the equilibrium factor and unattached fraction. The measurements were done in the laboratory of Waseda University, Tokyo. The radon concentrations were almost constant all day. However, the equilibrium factor and unattached fraction varied widely when the air conditioner was operated. When the air conditioner was turned off, the arithmetic mean of the equilibrium factor and unattached fraction were 0.6 and 0.05, respectively (5). When it was turned on, the arithmetic mean of the equilibrium factor and unattached fraction were 0.25 and 0.11, respectively (5). As other researchers have pointed out, a negative correlation can be found in the relationship between the equilibrium factor and the unattached fraction (10,11). Since the effects of the equilibrium factor and unattached fraction on dose are counterbalanced, determination of the radon concentration may be more significant than that of these parameters (10,11).

From other measurements carried out in Chiba over a period of 50 days with an active device, it was found that the radon concentrations during the working period were different from the daily average. Table 1 shows the variation of average radon concentrations in a week. The average concentration ratio between the working period and all day was 0.57 on weekdays whereas 0.84 on weekends. Table 2 and 3 show the variation of average equilibrium factors and unattached fractions, respectively. The equilibrium factors during the working period were larger than those during all day. On the other hand, the unattached fractions during the working period were smaller than those during all day. When compared with the radon concentrations, however, the difference was not so significant.

CONCLUSION

Radon measurements in workplaces were carried out. The radon concentrations during the working period and during all day were determined. A difference between the two was not found. The radon concentrations were almost stable whereas the equilibrium factor and unattached fraction varied since air conditioners were used in each office. If the relationship between the equilibrium factor and unattached fraction is taken into consideration, their effect on dose is counterbalanced. When the radon concentrations varied drastically, the equilibrium factor and unattached fraction showed only a slight variation. These findings suggest that for dose assessment, determination of the radon concentration is more significant than that of other parameters.

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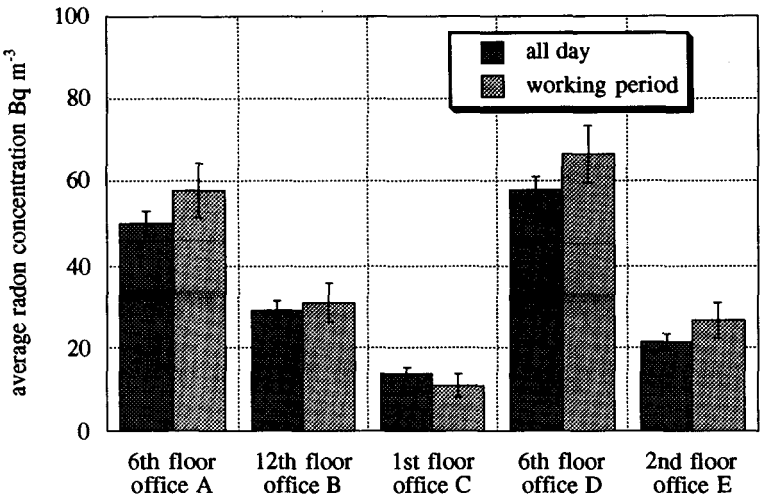

Figure 1. An example of average radon concentrations in five offices.

Table 1 The variation of average radon concentrations in a week

	Mon.	Tues.	Wed.	Thur.	Fri.	Sat.	Sun.
all day	14.2	15.1	14.6	12.8	15.2	20.6	19.7
working period	8.6	8.0	7.9	7.2	9.4	16.9	17.0

Table 2 The variation of average equilibrium factors in a week

	Mon.	Tues.	Wed.	Thur.	Fri.	Sat.	Sun.
all day	0.32	0.37	0.42	0.36	0.37	0.31	0.26
working period	0.41	0.40	0.50	0.42	0.41	0.33	0.24

Table 3 The variation of the average unattached fractions in a week

	Mon.	Tues.	Wed.	Thur.	Fri.	Sat.	Sun.
all day	0.064	0.058	0.060	0.068	0.068	0.078	0.101
working period	0.042	0.055	0.045	0.042	0.054	0.060	0.099

DETECTION EFFICIENCY OF A RADON DETECTOR LR 115 (CELLULOSE NITRATE)

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ABSTRACT

In this paper we propose an evaluation of detection efficiency for the LR 115 alpha particle detector. This evaluation is based on the consideration that the trach etch rate is proportionnal to the ionization rate at a given point of particle's trajectory. A good correlation is obtained when comparing both modelized and experimental detection efficiencies. It is put forward that detection efficiency depends upon the used measurment device (optical or spark counting).For LR 115, spark counting seems to be a much sensitive method when performing radon dosimetry.

INTRODUCTION

To explain the track etching phenomenon in LR 115, the conical model [1] uses the notion of constant etching rate during etching, both for the undamaged zone (V_b) and for the damaged zone(V_t). If this hypotheses seems us reasonable for the V_b it do not seem realistic for the description of V_t. The energy loss of an alpha particle varies when it is slowed down and the resulting damage density is not linear along the travelled distance into the detector material. This consideration suggests us that the track etch rate V_t does vary during the etching process. We will deduce the detection efficiency expressed as a n energetic sensitivity area and the mean critical angle for alpha particles registered in LR 115 and propose some limitations for its use in the field of dosimetry.

HYPOTHESES OF THE MODEL.

-We consider that the dissolution rate V_t (E;x) in cellulose nitrate in the vicinity of a point x of ionizing particle's trajectory with energy E is proportionnal to the ionizing rate J (E;x) at this point:

V_t (E;x) = K J (E;x); K being a proportionality factor for the ratio of these two rates.

-A track only will be visible if the etching process reaches the colourless substrate on which the cellulose nitrate sensitive fim is deposited.

-Ionization rate varies as a function of energy of the incident particle and consequently with its depth or position, x. Nevertheless, as indicated in Figure 1, for energies E greater than 4 MeV, this rate remains quasi-constant. In such a case, we will suppose that V_t (E>4MeV;x) = Cste.

For the experimental determination of K, we first measure the dissolution rate V_b of the non-damaged material using sodium hydroxyde with a molar concentration 2.5 M and an etching bath temperature of 60°C. The obtained value is $V_b = 3.4\ \mu m.h^{-1}$.

To follow the evolution of the specific alpha track etch rate V_t, we irradiated a set of 11 LR 115 films under normal incidence (90°), with alpha énergies of 4 MeV. The etching conditions were the same as indicated above but we used etching steps ranging from 80 to 130 mn.

In these conditions, the etched length along the trajectory L of the particle only exceeds the detector thickness (begins to be visible under an optical microscope) after a time t_d = 100 mn. The mean etching rate then is: V_t (E=4MeV;x) $= h/t_d = 7.2\ \mu m.h^{-1}$. h is the detector thickness (h=12μm).

For cellulose nitrate, the computer code TRIM91 [2] gives J (E=4MeV;x) = 176.74 keV. With these values one obtains $K = 0.0409\ \mu m^2\ h^{-1}\ keV^{-1}$.

RESULTS

1) Calculation of V_t (E;x).

In Figure 2 we present the variation of V_t (E=6MeV;x) in cellulose nitrate along the track axis of length R_6. An energy of 6MeV is choosen in order to cover the energetic detection range of the detector. The best fit for V_t (E=6MeV;x) was obtained using a 9 degree polynomial expression P(x) whose parameters are:

a_0=4.22; a_1=0.49; a_2=0.03; a_3=-0.06; a_4=1.32 10^{-2}; a_5=-1.38 10^{-3}; a_6=7.96 10^{-5}; a_7=-2.61 10^{-6}; a_8=4.55 10^{-8}; a_9=-3.28 10^{-10}.

The V_t (E;x) du to an alpha particle with energy E (range R) lesser than 6 MeV and with an incidence angle θ is obtained from P(x) with the forcoming boundary limits:

$$\Rightarrow L < h/\sin\theta: \quad (R_6 - R) < x < R$$

$$\Rightarrow L = h/\sin\theta: \quad (R_6 - R) < x < (R_6 - R + L)$$

2) Minimum time necessary to etch a track or to open a track.

The model enables the determination of the minimum time needed to to prduce a given track to be performed.

Figure 3. Pénétration of an alpha particle with incidence θ when the substrate is not reached (a) ; when the end point of the alpha particle is out of the detector (the track is open) (b).

Cas(a): The alpha particle is thermalized in the thickness of the detector. The range R is then such that R sinθ < h. The polynomial of V_t (E;x) is used with the boundaries $(R_6-R)<x<R_6$. The time t_p needed to etch a track to its end is:

$$t_p = \int_{\alpha}^{\beta} \frac{1}{V_t(E;x)} dx$$

where $\alpha = (R_6 - R)$ and $\beta = R_6$. Further, the dissolution (enlargement) of the undamaged material will be controled by the constant V_b.The minimum time necessary to obtain an open track is then:

$$(t_p)_{\min} = t_p + \frac{h - R\sin\theta}{V_b}$$

Cas(b): The particle passes through the detector and its range R is such that $R \sin\theta > h$. The condition on x being now $(R_6 - R_i) < x < R_6 - R + h/\sin\theta$, the minimum time needed to create an open track will be:

$$(t_d)_{\min} = t_p - \int_{\alpha}^{\alpha + h/\sin\theta} \frac{1}{V_t(E;x)} dx$$

Table 1 presents calculated values of the times needed to open a track with various energies and incidence angles. These values are in good agreement with those obtained by several experimenter [3,4].

Table 1. Calculated values $(t_d)_{min}$ for various energies E and incidence angles θ.

3) Sensitivity area of the detector

The sensitivity area of the LR 115 detector is determinated by means of the described model (Table 1). All the $(t_d)_{min}$ which are greater than 2h will correspond to non-detectable energies. The so defined sensitivity area is comprised between 2 MeV and 4.5 MeV. To test this result, 11 detectors were used. Irradiations were performed under normal incidence (90°) with an américium (241-Am) source having an initial alpha energy of 5.5 MeV. Energies between 0.5 and 5.5 MeV were obtained by changing the distance from the source to the detector in air. The etching time is equal to 2h. The sensitivity area determinated with an optical microscope is comprised between 1.2 and 4.6 MeV (Figure 3). The poor agreement observed at weak energies can be explained: it is du to the measurement tool. Indeed, for optical reasons, an open track is not necessarely observable under an optical microscope. Spark counting performed on LR 115 detectors irradiated in the same conditions provide a sensitivity area from 1.8 to 4.8 MeV, in better agreement with our predictions.

Figure 3. Experimental sensitivity area of LR 115 (optical microscope)
Figure 4. Critical angles calculated for LR 115

4) Calculation of critical angles.
From the model we deduce the variation of the critical angle θ_c as a function of energy (Figure 4). θ_c varies from a value of 80° to a value of 60° passing by a minimum value of 45° as energy ranges in the calculated sensitivity area. From this result, we deduce a calculated mean critical angle:

MEASUREMENT OF THE POTENTIAL ALPHA ENERGY CONCENTRATION IN AIR BY SEPARATE REGISTRATION OF THE RADON DECAY PRODUCT 214-Po USING NUCLEAR TRACK ETCHED DETECTORS

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INTRODUCTION

Long-term measurements of the activity concentration of ^{222}Rn in air can be easily performed using a solid state nuclear track detector within a diffusion chamber. The radiation exposure of the human respiratory tract is, however, mainly characterised by the potential alpha energy concentration W_p of the radon daughters. The measurement of this quantity over the same period of time is very difficult. Some authors applied the double chamber method, using one detector within a chamber closed by a diffusion filter and another detector within an analogeous chamber, but without the filter (1,2). The results found from the relationship between W_p and the activity concentration of both alpha active decay products (^{218}Po and ^{214}Po) depend strongly on the exposure conditions, e.g. on the equilibrium factor and the fraction of unattached radon daughters. The aim of our studies was to find a new approach to estimating the potential alpha energy concentration on the basis of its relation to the activity concentration of ^{214}Po only.

CONCEPT OF MEASUREMENT

The actual radon daughter concentration in air is affected by the dilution as the result of ventilation as well as the deposition on wall surfaces. The latter effect is different for radon daughters attached on aerosols and unattached daughters, resp. Taking into account realistic values for the attached fraction and the deposition rates the radon daughter concentrations c_1 (^{218}Po), c_2 (^{214}Pb), c_3 (^{214}Bi), c_4 (^{214}Po) can be calculated from the ^{222}Rn concentration c_0 by means of a recursion formula. Inserting the results into the definition equation for the potential alpha energy concentration W_p, the functions

$$\frac{W_p}{c_0} = f\left(\frac{c_1 + c_4}{c_0}\right) \tag{1}$$

and

$$\frac{W_p}{c_0} = f\left(\frac{c_4}{c_0}\right) \tag{2}$$

can be calculated (3). The results are shown in Fig. 1. Determining the activity concentration of radon on the one hand, and that of ^{218}Po and ^{214}Po or of ^{214}Po only on the other hand, the potential alpha energy concentration can be derived from these curves.

Measurements of ^{222}Rn and the total activity concentration of both alpha active decay products are frequently performed by means of the double chamber method. However, because of the marked steepness of the corresponding curve (I) in Fig. 1 a very high uncertainty of W_p results. Therefore, we proposed a new concept detecting ^{214}Po only. Then curve (II) in Fig. 1 is relevant which is much more flat. Therefore, more precise results for W_p can be expected. The superiority of this new method has been theoretically shown in our recent studies, especially in such cases where the unattached fraction of the radon daughters is relatively high (3-5). It needs, however, a method for the separate long-term measurement of the activity concentration of ^{214}Po.

For this purpose, the application of nuclear track detectors with a suitable energy response function is useful. From the alpha energy spectrum in air followed that such a detector should be sensitive only to alpha particles slowed down to energies within a window between 6.1 MeV and 7.4 MeV. This limitation is needed in order to avoid tracks from radon daughters plated out on the detectór surface. The aim of our studies was to optimize the energy response of a Makrofol DE detector for that purpose.

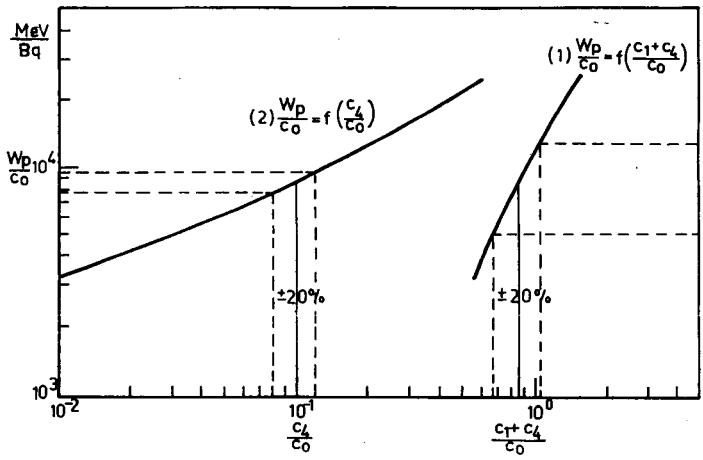

Figure 1. Potential alpha energy concentration as a function of the radon daughter concentrations after eq. (1) and (2)

EXPERIMENTAL AND THEORETICAL BASIS

The detector material used was Makrofol DE with a thickness of 475 µm. During the irradiations the detectors were covered by a protecting Mylar foil with a thickness of about 2 µm.

For irradiations with high energy alpha particles a source has been prepared by electrodeposition of thoron daughters. The alpha particles with an initial energy of 8.8 MeV have been slowed down in the irradiation chamber by variation of the air pressure to fall into the interesting energy interval between 5.8 MeV and 7.7 MeV.

The main task was to investigate whether the energy window characterised by two thresholds can be generated. For that some theoretical considerations were performed. Etchable tracks are formed only when the restricted energy loss REL_{350} of the alpha particles along their paths exceeds the critical energy loss for the detector material. The REL_{350} value depends strongly on the alpha energy and varies, therefore, considerably along the alpha particle trajectories (see Fig. 2).

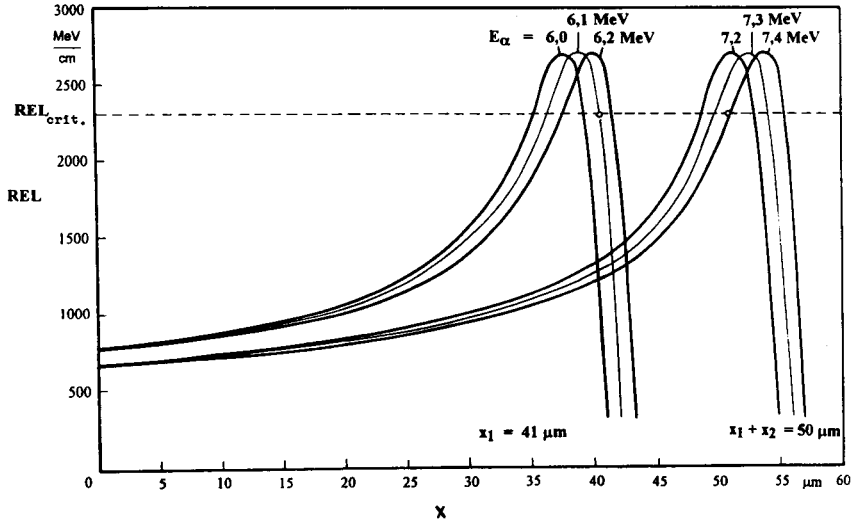

Figure 2. Restricted energy loss REL as a function of the depth x within the detector (including 2µm Mylar) (Parameter: Initial alpha energy E_α; Calculation using the program STOPOW 2000 (6))

The critical REL_{350} value has been found to be about 2300 MeV/cm in separate experiments. To avoid a formation of etchable tracks by alpha particles with energies lower than 6.1 MeV a detector layer (below the Mylar foil) of about 39 µm has to be removed by a chemical etching process. Over-etched tracks caused by lower energy alpha particles may not contribute to the detector reading during a further processing of the detector by electrochemical etching. The treeing process with strong enlargement of the tracks takes place only

if a sharp tip is formed at the end of the chemical etching, i.e. not for over-etched tracks. Figure 2 shows that at a depth x_1 of 41 μm the 6.1 MeV alpha particles are just ready to produce electrochemically etchable tracks. On the other hand, the energy loss of alpha particles with initial energies above 7.4 MeV may not exceed the critical value for track formation at a depth of $x_1 + x_2 = 50$ μm. That means, the detector layer removed during the electrochemical etching must be about 9 μm up to the end of this etching step. Because the formation of electrochemically etched tracks is influenced by the applied field strength it has been expected that the exact thresholds and the shape of the alpha energy window depend on the field strength, too.

RESULTS

The etching was carried out using a mixture of 80% ethanol and 20% KOH at 40°C. During the chemical pre-etching a detector layer of 39 μm was removed. The electrochemical etching was performed at an electric field strength of 51 kV/cm and a frequency of the high voltage of 3 kHz for 1 hour. After that a chemical post-etching followed for 0.5 hours. During this process the tracks are enlarged without changing the energy response of the detector. This last step makes, however, the detector evaluation easier. The total detector layer removed during the electrochemical etching and chemical post-etching was about 14 μm. The maximum track diameter amounted to 200 μm. The resulting energy response of the detector is shown in Fig. 3.

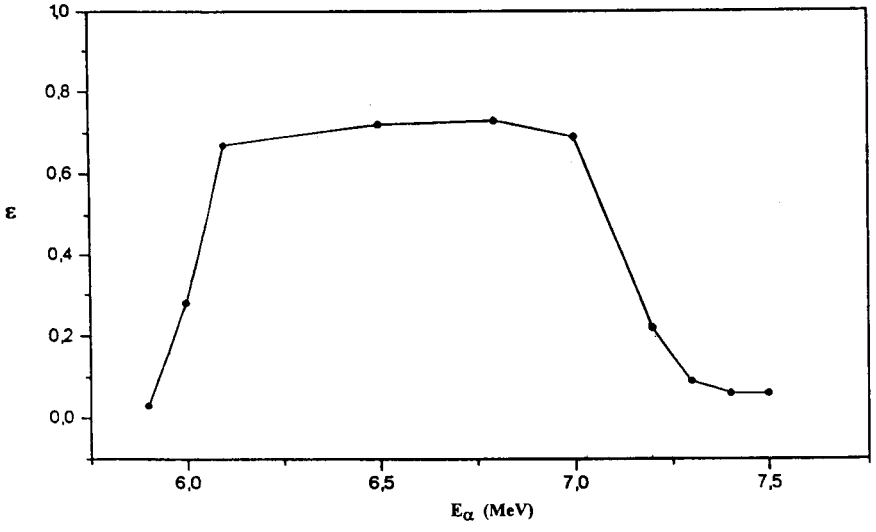

Figure 3. Registration efficiency ε as a function of alpha energy E_α

CONCLUSIONS

The results of our studies confirmed that, using Makrofol-DE detectors treated by a combined chemical and electrochemical etching, the energy window needed for a separate detection of ^{214}Po can be generated. The activity concentration c_4 of ^{214}Po results from the detector reading, i.e. the track density within a defined period of exposure. The calibration factor which links both quantites can be found by an experimental calibration using a reference measuring device for ^{214}Po. On the other hand, the calibration factor can also be calculated. The corresponding relationship was derived in a former paper (3). The most important quantity needed for the calculation is the critical angle of track detection as a function of the alpha energy. The precise determination of this function will be the objectives of our next work. Using the c_4 value and the radon concentration c_0 determined by means of a traditional method, e.g. using the diffusion chamber, the potential alpha energy concentration W_p can be determined..

A test of the method under practical conditions is planned, e.g. in living rooms or at special working places.

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RECENT PROGRESS IN SAMPLING AND MEASUREMENT OF RADON AND THORON DECAY PRODUCTS

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Abstract: **A new approach to sampling and measuring the unattached fraction of radon decay products is presented. Two methods of electrostatic sampling with polystyrene plates charged by friction or with copper plates connected to a high voltage source are compared to conventional sampling on wire mesh screens. A good correlation between Po-218 alpha activities for the three sampling methods for different radon concentrations and aerosol conditions was found. For total beta activities correlation between the methods is less obvious.**

INTRODUCTION

The alpha-decay of Rn-222 and its recoil result in about 90% positively charged Po-218 ions. They have a high mobility, characterized by the diffusion coefficient, which controls the processes of aerosol attachment and deposition on surfaces. The diffusion coefficient of charged and uncharged Po-218 and Pb-212 has been studied with electric fields in diffusion chambers (1, 2). Despite the fast neutralization of the ions, collection is possible with a sufficiently high voltage. The efficient mean life time of free Po-218 ions due to decay, recombination and attachment is typically 12 s (3). Therefore it is possible to determine the Po-218 concentration by collecting the free ions. Likewise Pb-214 may be positively charged, but its amount is unknown, which makes the determination of the Pb-214 concentration by collecting the ions less certain. The recoil energy of the beta decay of Pb-214 to Bi-214 is not sufficient for ionization and thus there are no Bi-214 ions.

All radon decay products quickly attach to aerosol particles or deposit on surfaces. Airborne radionuclides are divided into an "attached" and an "unattached" fraction. Molecular clusters up to 1 nm in size are formed with water molecules and trace gases. The fraction with a thermodynamic diameter below 2 nm is defined as "unattached", the larger one as "attached".

Unattached radionuclides have a higher diffusivity and may be deposited on wire mesh screens. An alternative method is a collection on negatively charged plates, which attract the positively charged radon decay products. The unattached fraction is attracted much more effectively than the attached fraction, due to its much higher mobility. In the following the activity collected on charged plates is compared with the activity on wire mesh screens under different radon concentrations and aerosol conditions.

MATERIALS AND METHODS

The alpha activity was determined by alpha spectroscopy with a 30 cm^2 surface barrier silicon detector in a vacuum chamber and a multichannel analyzer (EG&G, D-81677 München), and by a hand-held alpha spectrometer Alpha Analyzer with a 2" diameter silicon detector (SAIC, San Diego, CA 92121 USA). The beta activity was measured with the Living Level Monitor LLM 500 (Münchener Apparatebau, D-82024 Taufkirchen) with a 11 x 11 cm^2 proportional detector (4).

The attached fraction was sampled with glass fiber filters MN 85/90 (Macherey-Nagel, D-52348 Düren). The unattached fraction was deposited on wire mesh screens (Bucher, D-81669 München) with a mesh width of 0.04 mm and a wire diameter of 0.028 mm (147 mesh/cm = 374 mesh/in). Both have an effective diameter of 10 cm and are mounted in cardboard diskettes, which fit into the LLM. The wire mesh screen was fixed in front of the glass fiber filter on an air sampler (vacuum cleaner). The distance between the wire mesh screen and the glass fiber filter was 4 mm to avoid penetration by alpha-recoil-atoms. After 1 min of sampling at 250 L/min with the air sampler and 10 s transfer time, the filter and the screen were evaluated in the LLM within 1 min for beta activity and equilibrium equivalent radon concentration (EER). Additional screen samples were measured for alpha activity on the front side. 250 L/min through a 10 cm diameter screen yield a subcritical face velocity of 50 cm/s, for which the particle diameter for 50% collection efficiency is about 2 nm.

Collector plates for electrostatic collection are polystyrene plates (PS) of 130 x 200 x 2 mm^3 charged by friction, "Philion Plates" (5), or Cu plates of 100 x 100 x 0.6 mm^3, charged by a

high voltage source. The PS plates were charged by rubbing with wood or wool. With charge saturation on the PS by appropriate rubbing, -20 kV are reproducibly achieved and confirmed by measurement with a "field mill" Static Monitor 140 C (John Chubb Instrumentation, Cheltenham, GL51 8PL, UK). The voltage on the plate decreases faster at higher radon concentrations. The Cu plate was connected to a constant voltage source of -16 kV. No significant differences between PS plates and Cu plates were found, so in the more results from the Cu plate are described, because they are assumed to be more convincing. After sampling for 1 min and transfer time of 10 s Cu plates were measured 1 min for beta and for alpha activity. Using large-area collectors and large-area detectors improves overall efficiency approx 100-1000 fold, relative to wires connected to a voltage source and an electrometer discharge first used by E. Rutherford in 1900 in the laboratory, and by J. Elster and H. Geitel in 1901 in the open air.

The samples were taken in a radon lab (46 m^3), where uranium minerals were stored in air tight containers. The radon concentration was changed by opening and closing containers. The aerosol concentration was changed by burning candles and incense and filtering the air with an air cleaner Breatheasy (12 m^3/min) (Biltwell, Mississauga, Ontario L5C 2Z2). The radon gas concentration was varied between 500 and 12000 Bq/m^3.

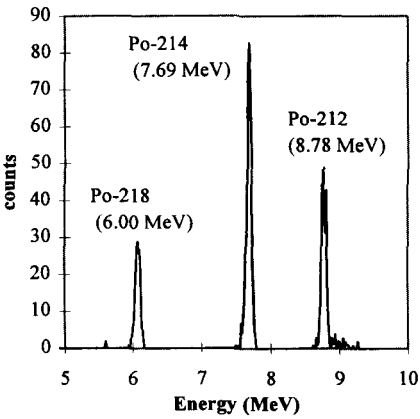

Fig. 1. Alpha spectrum of a Philion Plate, exposed to ambient air for 25 min; 3.7 h count with a 30 cm^2 silicon detector in a vacuum chamber

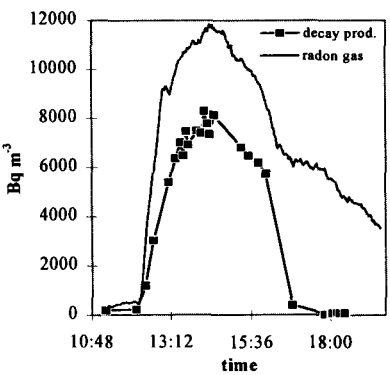

Fig. 2. Concentration of radon gas and EEC of radon decay products in the radon test room

RESULTS AND DISCUSSION

A problem with alpha measurements is the self-absorption in samples like glass fiber filters. Peaks in alpha spectra become very broad with crosstalk between Po-218 and Po-214 peaks. A problem with wire mesh screens is the dependence of the front to back ratio and screen losses on environmental parameters. Fig. 1 shows the alpha spectrum of a Philion Plate, which was exposed to ambient air for 25 min and measured with the 30 cm^2 silicon detector for 3.7 h in the vacuum chamber. The result is a neat spectrum of Po-218 and Po-214 from the radon decay series and of Po-212 from the thoron decay series. All decay products are on the surface of the plate, so there is no self-absorption like in glass fiber filters. Therefore it is possible to count the plates in open air with the SAIC Alpha Analyzer for Po-218 and Po-214 activities without crosstalk. The spectrum on a Cu plate is of similar appearance.

For fast measurements with good counting statistics, high radon levels were used. Fig. 2 shows the variation of radon gas, measured with an instrument Megarad, and of attached radon decay products EER, collected on glass fiber filters and measured with a LLM 500. The radon gas concentration was raised by opening a container with uranium minerals at 12:11. High aerosol particle concentrations of different size distributions were produced by burning candles from 12:45 to 14:40 and incense from 15:15 to 16:10. From 16:10 to 19:00 the air cleaner was put into operation. There is almost no thoron, so measurements were not disturbed by thoron decay products.

After sampling, the Cu plate and the screen were counted for alpha activity with the SAIC Alpha Analyzer and for beta activity with the LLM 500. In fig. 3 the alpha counts in 1 min of the Po-218 peak on the Cu plate and the screen are shown. Pairs of values, measured at nearly the same time are plotted in fig. 4. The numbers indicate corresponding values. There is a good correlation between the activities sampled on the screen and on the Cu plate. This confirms, that there is a fairly constant fraction of Po-218 ions within the unattached fraction for

different aerosol and radon gas conditions. It also shows that the collected activity on the Cu plate or the friction charged polystyrene plate is a fair quantity for the unattached fraction under different aerosol and radon gas conditions.

The Po-214 activity was very low and showed no correlation for both methods of sampling. This is explained by the fact that there is almost no unattached and no charged fraction.

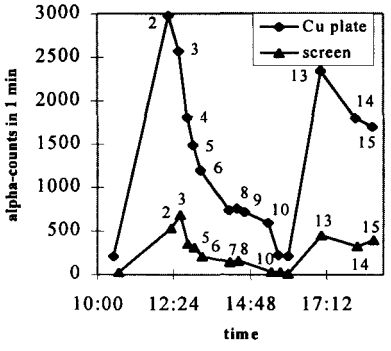

Fig. 3. Alpha counts of Po-218 on the Cu plate (1 min exposition at -16 kV) and on the screen (250 L of air sampled), measured with the SAIC alpha analyzer. Numbers refer to fig. 4

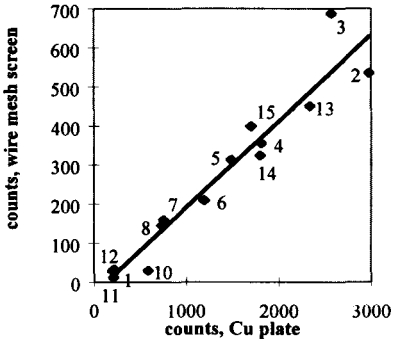

Fig. 4. Alpha counts of fig. 3, plotted as pairs, to show the correlation between the Po-218 activity on the screen and on the Cu-plate. Numbers indicate the points, taken from fig. 3

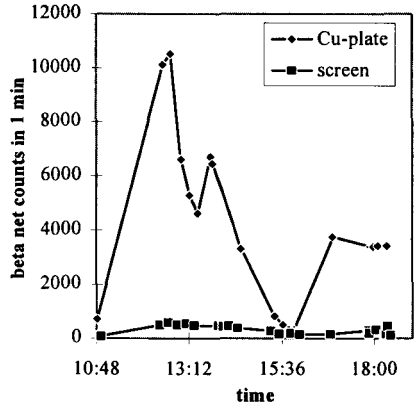

Fig. 5. Beta activity on the Cu-plate and the screen, measured with the LLM 500.

In Fig. 5 total beta counts are shown. There is no obvious correlation between the two series of measurements. The Cu plate shows a wider range of activities for different conditions. This leads to a vague exponential correlation between the two sampling methods. There are several reasons for not having a good correlation. First of all, decaying Po-218 adds to the beta activity. In addition, the correlation between the charged and the unattached fraction of Pb-214 is not known and depends on parameters like humidity and trace gases. Both charged plates sample only the charged unattached fraction, screens sample charged and uncharged, and partly the attached fraction.

CONCLUSIONS

The comparison of the wire mesh screen method with the two types of charged plates - Cu plates and the Philion Plates - gives the following results: A charged plate is a good sampling device for the unattached fraction of Po-218. Despite the very fast neutralization rate of the Po-218 ions, it is possible to collect them by this method, which is simpler and more efficient than wire mesh screens. Because of the sufficiently high voltage the ions are collected before recombination. The determination of the volume, from which Po-218 is sampled, and possible influences of water vapor are still under investigation.

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PAPER TITLE

EQUILIBRIUM FACTOR OF RADON IN A HOUSE AND INFLUENCING FACTORS

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MAJOR SCIENTIFIC TOPIC NUMBER 1.1. (see page 7)

ABSTRACT (See instructions overleaf)

Continous and simultaneous measurements of radon, the attached, unattached radon daughters and aerosol particles were performed in a house under normal living conditions which include ordinary living habits and the normal natural ventilation rate in Japan.

Radon concentration was measured by use of a radon monitor which is composed of an electro-precipitation chamber and a ZnS(Ag) scintillation counter. Concentrations of individual daughter(RaA,B and C) were also measured by a continuous-radon daughter monitor, which has air filtration and gross alpha three counting systems. The unattached atoms of radon daughters were measure with another radon daughter monitor with wire screens substituting for the filter.

The experimental data through one annual cycle were focused on the temporal variation and the influencing factors of equilibrium factor. The equilibrium factor of the house was with the range from 0.35 to 0.45 in monthly mean. The present results showed that there are a high correlation between the equilibrium factor and the unattached fraction of radon daughters and the relation is in agreement with the theoretical prediction, which describes a mass balance of indoor radon daughters.

APPLICATION OF PASSIVE TYPE RADON DETECTORS TO FIND FISSURES IN BANKS CAUSED BY THE SOUTHERN HYOGO PREFECTUAL EARTHQUAKE IN JAPAN

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INTRODUCTION

Innumerable fissures were formed widely in Hanshin area in Japan by the former southern Hyogo prefectural earthquake occurred on Jan.17, 1995. It is an urgent necessity for the science of disaster prevention to find these fissures simply and non-destructively. There are thousands of holding reservoirs in the Hanshin area and it is expected that some bank of reservoirs might have fissures inside the bank. In this study, as a preliminary investigation, we applied passive type radon detector Pico-Rad (U.S. Packard Instrument Co. Ltd.) (1, 2) with hemispheric plastic cover over them around the fissure along on the bank of two reservoirs to examine whether there is possibility to find fissures using the characteristics of ^{222}Rn migration. At the same time, we measured the total exposure rate on the ground with NaI(Tl) scintillation surveymeter (Hamamatsu Photonics, SS-γ), radioactivity of soil by sampling method (and low back ground germanium detector) and water contents of soil as supplementary factors of the investigation.

One of the reservoir, Taniyamakami-ike, is located at the north of the Awaji-shima island at a distance of about 4 km from the seismic center of this earthquake. On the whole, ^{222}Rn exhalation rates were high on a line of 2 m below the edge of the bank and got lower in proportion to the distance from this line. Those top of the bank had middle values among the lines. The mean ^{222}Rn exhalation rate was 4.7 mBq m^{-2} s^{-1}, and in these data we found some singularly high ^{222}Rn points.

The other reservoir named Hosho-ike is located at northwestern Nagaokakyo city of Kyoto prefecture and the Komyoji active faulting runs from north to south parallel with the bank about 120 m to the west. In this bank, a fissure about 50 m in length and 0.1 m in width was taken shape. ^{222}Rn exhalation rates were relatively high on the top of the bank compared with on the slope of the bank. The mean ^{222}Rn exhalation rate of 5 points which were measured on the fissure was 16 mBq m^{-2} s^{-1}, and that which were measured on about 1 m to the east from the fissure was 4.9 mBq m^{-2} s^{-1} and in case about 1 m to the west was 4.2 mBq m^{-2} s^{-1}.

From these results we concluded that there is a possibility to find fissures inside the bank using the characteristics of ^{222}Rn migration. Our future objective is to find fissures inside banks non-destructively.

PRINCIPLE AND METHODS

^{222}Rn ($T_{1/2}$=3.8 d) originates from the decay of ^{226}Ra ($T_{1/2}$=1600 y) that distributes widely in the soil. Typical processes of the ^{222}Rn transport below the ground surface are molecular diffusion and/or convective flow, and also the macroscopic flow in channels or fissures as this case. So if there is any fissure in the bank, it is expected that high ^{222}Rn concentration will be observed above the fissure.

In this survey we used Pico-rad detectors for the measurement of ^{222}Rn concentration. It is a compact ^{222}Rn detector using the adsorption of ^{222}Rn to the activated charcoal and ^{222}Rn concentrations are evaluated by LSC counting. The detectors are passive collection devices requiring no power. It is come onto market by U.S. Packard Instrument. Some of the detectors were covered with hemispheric plastic cover to get higher ^{222}Rn

activity and to evaluate area exhalation rates.

Besides there are a lot of papers related to the radiological survey using the terrestrial gamma ray (3), we had not been certain about applying these method to the investigation because our object is the fissures which were made recently not the fault. But for the present we measured the total exposure rate on the surface of the earth with NaI(Tl) scintillation detector (Hamamatsu Photonics, SS-γ), radioactivity of soil by sampling method (and low background germanium detector) and water contents of soil as supplementary factors of the investigation. Measuring methods and conditions of the survey of the Taniyamakami-ike and Hosho-ike are shown in Table 1.

Table 1 Measuring methods and conditions of the survey on the Taniyamakami-ike and Hosho-ike.

	Taniyamakami-ike reservoir	Hosho-ike reservoir
Measuring date (weather)	15-17 Oct., 1995 (clear and cloudy)	17,18 Apr., 1995 (clear)
Pico-rad detectors used	**No. 1-95** set on the ground with the hemispheric plastic cover **No. 96-100** set at 5 cm high above the ground without cover	**No. 1-21** set on the ground with the hemispheric plastic cover **No. 22-24** set in the fissure without cover **No. 25-26** set at 5 cm high above the ground without cover
Arrangement	lattice (interval 2 m)	across (5 lines) and along the fissure
Total sampling time	23~25 hr	about 19 hr
Supplementary measurements	**Total exposure rate** (NaI(Tl) scintillation detector, 56 points) **Radioactivity of soil** (sampling, Ge detector, 9 points) **Water content of soil** (sampling, 28 points)	**Total exposure rate** (NaI(Tl) scintillation detector, 8 points) **Radioactivity of soil** (sampling, Ge detector, 1 point) **Water content of soil** (sampling, 21 points)

RESULTS AND DISCUSSION

1) Taniyamakami-ike reservoir.

The 9 points (on the same line across the bank) mean radioactivity and s. d. of ^{214}Bi, ^{208}Tl and ^{40}K about 10 cm below the ground surface were 23 ± 2 Bq kg^{-1}, 33 ± 3 Bq kg^{-1}, 1140 ± 40Bq kg^{-1}, respectively. We did not find any characteristic feature in this line on the three radioactivity of soil and also on the ^{214}Bi/^{208}Tl ratio. The atmospheric ^{222}Rn concentration measured at 5 cm high above the ground (No. 96~100) was 9.2 ± 3.2 Bq m^{-3}. Figure 1 shows a horizontal distribution of ^{222}Rn concentration inside the hemisphere (No. 1~96). The mean total exposure rate on the ground surface measured by SS-γ was 100 ± 10 nSvh^{-1} (58 points, max. 137, min. 65 nSvh^{-1}). On average, ^{222}Rn exhalation rates were high on a line of 2 m below the edge of the bank and low nearby the bed and at the back slope. Those top of the bank had middle values among the lines. The mean ^{222}Rn concentration was

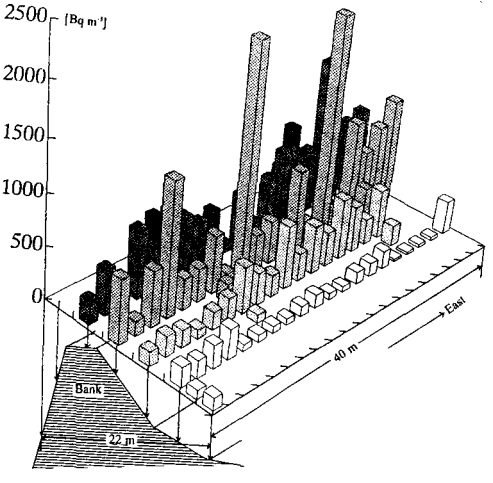

Figure 1. Horizontal distribution of ^{222}Rn concentration around the fissure in the bank of Taniyamakami-ike.

290 Bq m^{-3} (corresponding to 4.7 mBq m^{-2} s^{-1}), and in these data we found some high ^{222}Rn points that concentration was over one thousand Bq m^{-3}. We expect that there are some buried fissures around below this line. The mean water content of soil and its s. d. at about 10 cm deep of 28 points was 14±12% (max. 58, min. 1.1 %) and we did not find the correlation of ^{222}Rn concentration with water content of soil.

2) Hosho-ike reservoir.

The radioactivity of ^{214}Bi, ^{208}Tl and ^{40}K 10 cm below the ground surface were 7.9, 14, 560 Bq kg^{-1}, respectively. The mean total exposure rate on the ground surface measured by SS- γ was 70 $nSvh^{-1}$ (6 points, max. 100, min. 57 $nSvh^{-1}$). These values are within the range of that of the normal natural environment. The atmospheric ^{222}Rn concentration measured at 5 cm high above the ground (No. 25, 26) was 11± 10 Bq m^{-3}, and ^{222}Rn concentration measured in the fissure at 0.3～0.5 m deep (No. 22～24) varied widely from 40 to 4200 Bq m^{-3}. Figure 2 shows a horizontal distribution of ^{222}Rn concentration inside the hemisphere. ^{222}Rn concentration were relatively high on the top of the bank compared with the slope of the bank.

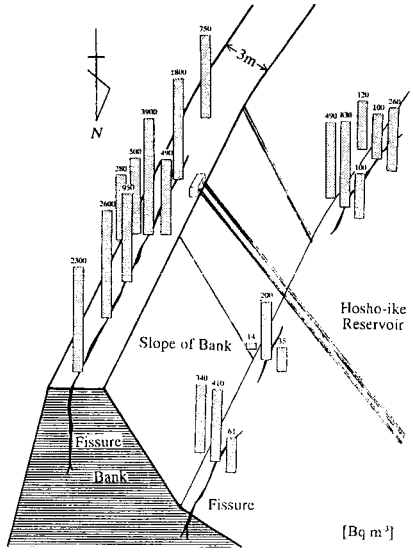

Figure 2. Horizontal distribution of ^{222}Rn concentration around the fissure in the bank of Hosho-ike.

The mean ^{222}Rn concentration of 5 points which were measured on the fissure was 1000 Bq m^{-3} (16 mBq m^{-2} s^{-1}), and that which were measured on about 1 m to the east from the fissure was 300 Bq m^{-3} (4.9 mBq m^{-2} s^{-1}) and in case about 1 m to the west was 260 Bq m^{-3} (4.2 mBq m^{-2} s^{-1}). The mean water content of soil at about 10 cm deep of 21 points was 11.4 % (max. 18.5, min. 5.6 %) and there was no correlation of ^{222}Rn exhalation rates with water content of soil. (After the survey, repair work is now in progress on the Hosho-ike reservoir. About 7 m fissure was found in the cross section of the bank (Photo 1)).

Above mentioned two results showed that it is effective for the survey of new fissures to apply ^{222}Rn gas method rather than the radiological survey using gamma-ray.

Photo 1. The fissure in the bank of Hosho-ike.

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DIFFUSION OF RADON DAUGHTERS IN LAMINAR TUBE FLOW

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KEYWORDS

diffusion, deposition, laminar flow, radon daughters, unattached fraction

INTRODUCTION

To study the plate-out of the first three unattached radon daughters (^{218}Po, ^{214}Pb, ^{214}Bi), we have to know the diffusivities of these particles. One way to determine these diffusivities is to use the so-called two-filter method (4, 17). By theoritical calculation and experimental measurement of the penetration fraction of a radon daughter through a circular cross-section tube, with filters at each end, we can determine its diffusivity. There are, however, uncertainties in this method and the aim of this work was to re-analyse the theories used to calculate the penetration fraction and to employ experimental conditions which avoid errors in determining these diffusivities.

TWO-FILTER METHOD APPLIED TO RADON DAUGHTER DIFFUSIVITIES MEASUREMENTS

A known concentration of radon (^{222}Rn) is introduced into the tube. Due to the upstream filter, the initial radon daughter concentrations are zero. The radon daughters are then formed by radioactive decay, transported by convection and deposited on the tube walls by brownian diffusion. Those which are not deposited on the tube walls and have not decayed are collected on the outlet filter and form the penetration concentration.

THEORETICAL STUDY

Published work (2, 5, 9, 12) on radon daughter penetration through a tube is limited to studies of the first radon daughter and a number of assumptions were made to allow an analytical determination of the penetration fraction. By using a computer program, we can study the penetration fractions of the first three radon daughters and take into account phenomena neglected in the analytical theories.

We used the TRIO EF program (10), a fluid-mechanics code, based on the finite-element method, developed by the CEA Nuclear Reactors Direction.

VALIDATIONS

Our program was validated by :

- diffusive deposition of non-radioactive aerosols in laminar tube flow (3, 6, 7, 16)
- diffusive deposition of aerosols with in-flight formation (1, 8, 14, 15)
- study of diffusive deposition of aerosols with in-flight formation and radioactive decay: comparison with results of a finite-difference method (13).

There is a good correlation between our results and those in the literature ; TRIO EF can thus be considered to provide good modelling of these diffusion and decay phenomena.

^{218}Po DIFFUSIVITY CORRECTION

The analytical theories (1, 3, 6, 7, 8, 14, 15, 16) used for the two-filter method do not make allowances for phenomena that can be taken into account with TRIO EF, i.e. :

- loss of radon daughters by radioactive decay - leading to a penetration fraction decrease ;
- Poiseuille flow development : this effect creates a radial velocity field component directed to the tube centre and also enhances the penetration fraction
- downstream filter effect on flow : this effect creates a radial velocity field component directed to the tube wall, and also reduces the penetration fraction

These effects result in correcting of the penetration fraction by around 5-20% depending on tube length, tube radius and mean axial velocity. This leads to correction of the ^{218}Po diffusion coefficient determined by the two-filter method by around 5-25%. Sasse (13) found a correction of around 10-20%.

^{214}Pb AND ^{214}Bi DIFFUSIVITIES

We computed the second and third radon daughter penetration fractions using this program. These penetration fractions are different and this will be explained in our poster by using concentration profiles of the three species.

This program can also help us to determine ^{214}Pb and ^{214}Bi diffusion coefficients.

EXPERIMENTAL STUDY

The experimental measurements can give rise to errors in determining radon daughter diffusivities. We tried in our work to reduce the number of factors that could result in such errors.

RADON CONCENTRATION ERRORS

The ^{226}Ra source used to produce radon should have a constant emanation factor, which we verified.

We also ensure that we do not have any ^{220}Rn (from impurities in the ^{226}Ra) by allowing it to decay in an upstream volume and verify that there are no ^{220}Rn daughters.

The radon concentration measurements must be very precise. We use a gross alpha counting method calibrated with a reference radon monitor. We also correct the radon concentration for temperature and pressure variations.

DEPOSITION ERRORS

Deposition must be due only to brownian diffusion and we take care to avoid thermophoresis and electrostatic deposition.

RADON DAUGHTERS DIFFUSIVITIES ERRORS

Trace gas can affect radon daughter diffusivities and thus we measure both trace gas concentration and relative humidity and keep these constant as far as possible.

The radon concentration affects the air ionisation rate and thus the radon daughters neutralisation rates and their diffusivities. We thus divid our measurements into those with low and high radon concentrations.

RADON DAUGHTERS MEASUREMENTS ERRORS

Radon daughter concentrations are usually obtained with a total alpha three-count method developed by Thomas (18) which results in uncertainties at low radon daughter concentrations. We use a more precise method optimised by Nazaroff (11).

SAMPLING TIME ERRORS

In the theoretical study, the radon concentration is assumed to be constant and uniform throughout the tube. This is not true in the experiments and we have to use a sampling time which is much greater than the time needed to have this condition. This time can be computed with our program.

Experimental results are now being obtained for comparison with the computer program and to have more precise diffusivity values for the three radon daughters.

CONCLUSION

In this work, we have re-analysed the two-filter method for determining radon daughter diffusivities.

We used a computer program which takes into account phenomena neglected in analytical determinations. Making allowances for these phenomena leads to corrections of around 5-25% for ^{218}Po diffusivity. This program can also help us to determine the other radon daughter diffusivities.

In our experiments, we considered various parameters that can affect diffusivity determination.

Corrections from using theoretical and experimental approaches should give us much more accurate values for the first three radon daughter diffusivities.

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PAPER TITLE PRODUCTION AND PROPERTIES OF CARNAUBAWAX AEROSOLS FOR RADON REFERENCE CHAMBERS

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The measurement of radon and radon progenies as well and the calibration of their detection systems require the production and measurement of well-defined aerosols in size and concentration. In the German radon reference chamber carnaubawax is used for the production of standard aerosols due to their unique chemical and physical properties. The aerosol size spectra are measured on-line by an aerosol measurement system in the range of 10 nm to 1 μm aerodynamic diameter and compared with theoretical calculations from molecular dynamics for the first time.

The experimental set-ups will be described for the study of adsorption of radioactive ions on aerosols as function of their size and concentration, the results presented and further adaptions for an aerosoljet (for example measurement of short-lived neutron-rich isotopes) introduced.

A LABORATORY FOR STUDYING RADON MITIGATION METHODS IN HIGH-RISE OFFICE BUILDINGS IN HONG KONG

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ABSTRACT

A territory-wide survey of indoor radon level in 1993 showed that 17% of offices in Hong Kong have radon concentrations above 200 Bq m^{-3} compared with 4% for dwellings. Consequently, the Radioisotope Unit Radon Analysis Laboratory (RURAL) is being built for studying radon mitigation methods applicable to high-rise office buildings. The laboratory consists of three rooms; the main exposure room is built of concrete and is surrounded by the buffer room; and all controls and operations are done inside the control room.

The exposure room can, with the aid of the buffer room, simulate any environmental conditions that can be faced by a real building. The pressure, temperature and humidity can be adjusted to any meteorological conditions that can be found in Hong Kong. Pressure differential and temperature differential can be adjusted to simulate the arrival of fronts, troughs or typhoons. Aerosol concentration and distribution inside the exposure room are controllable as well as the ventilation conditions.

Various mitigation methods will be tested under different conditions. Passive methods include application of radon barriers to building structures and active methods include the use of air cleaners; techniques to increase radon daughters plateout or reduce their attachment to aerosols; and various modifications to the ventilation systems. Mitigation techniques involving modifications to the building strictures and building services will also be developed with the help of the RURAL.

INTRODUCTION

Building materials containing naturally occurring ^{232}Th and ^{238}U, which decay to ^{220}Rn and ^{222}Rn respectively, are the major sources of indoor radon, especially in poorly ventilated environment (1). Some European and U.S. countries have proposed regulations to control the use of building materials in order to minimize their hazards to the people. Hong Kong is a densely populated city, most people spend a lot of time, both living and working, in high-rise concrete buildings; and owing to the hot and humid weather, the use of air-conditioning systems with limited air change rate is very common. According to the territory-wide indoor radon survey conducted in 1993, 17% of offices in Hong Kong have radon concentrations above 200 Bq m^{-3} compared with 4% for dwellings. Moreover, the contribution of ^{220}Rn in Hong Kong is significant, since the predominant geological material is decomposed granite that has a higher content of ^{232}Th. According to another previous survey conducted in 1986 (2), the hazard of ^{220}Rn is almost half of that of ^{222}Rn. Because of the potential high radon dose, there is a necessity to build the RURAL to investigate radon mitigation methods under situations in Hong Kong.

STRUCTURE OF RURAL

The RURAL consists of three rooms, the Exposure Room, Buffer Room, and Control Room. The total area is 8 m x 5 m, and the overall height is 3 m. A schematic diagram of the RURAL is given in Fig. 1.

The Exposure Room is the room where indoor environment of high-rise building is simulated, and where most of the measurements will be done. Its is 3.5 m x 2.4 m x 2 m (L x W x H) and is built of 10 cm thick concrete. The Exposure Room is completely isolated from outdoor environment by the Buffer Room which completely encloses it. The floor of the exposure Room is also isolated from the floor of the Buffer Room by raising the Exposure Room with supports underneath it. Entrance to the Exposure Room is through double air-tight doors which minimize disturbance to the exposure condition.

The Buffer Room is 5 m x 5 m x 3 m (L x W x H) and is built of bricks. Its main function is to simulate all sorts of meteorological conditions external to the Exposure Room. Hence by changing the pressure and temperature inside both the Exposure Room and Buffer Room, the arrival of fronts, troughs or typhoons can be simulated and their effects to indoor radon concentration can be studied. Entrance to the Buffer Room is again guarded by double air-tight doors. All the walls, ceiling and floor of the Buffer Room are coated with polyurethane paint which reduces the generation of radon from the building material of the Buffer Room thus keeping the radon concentration at the low outdoor level.

The Control Room is where all controls and operations take place. Sample of air in the Exposure Room can be taken here through ducts connected to the Exposure Room. The room also contains a stainless steel radon exposure chamber (3) where radon related experiments are done.

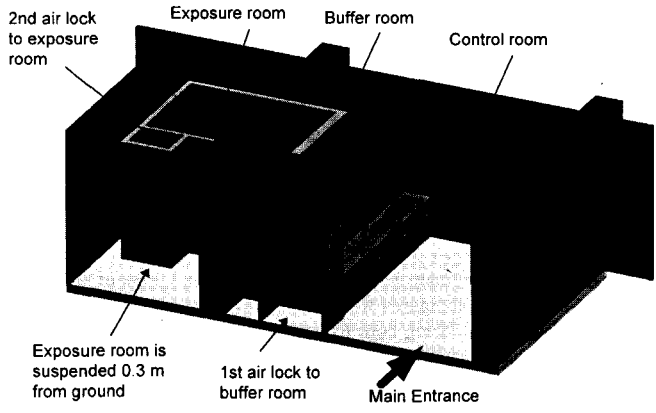

Figure 1. The RURAL (Radioisotope Unit Radon Analysis Laboratory)

FEATURES OF RURAL

The followings are brief descriptions of major equipment installed in the RURAL.

1. Aerosol generator

To create a steady concentration of monodispersed aerosol, a high output atomizer (model 3076, TSI Inc., USA) and a diffusion dryer (model 3062, TSI) are installed. The aerosol particle size can be altered by changing the concentration of solute in the solution of the atomizer. Typical particle size ranges from 0.02 m to 0.3 m. By adjusting the injection duration and interval, a wide range of aerosol concentration can be achieved.

2. Aerosol measurement

The size distribution of the aerosols are measured by two equipment. The Scanning Mobility Particle Sizer (SMPS) from TSI selects particle size with an electrostatic classifier (model 3071A, TSI) and then measure their concentrations with a condensation nuclei counter (model 3052A, TSI). The SMPS can measure the size distribution of aerosols from 5 nm to 1 μm. Above 1 μm, the aerosols are measured by a laser optical particle counter (model LASAIR 510, PMS) which has eight particle size channels from 0.5 μm to 10 μm.

Activity size distribution of ^{222}Rn daughters is determined by six wire screens-filter-detector assemblies (3). By varying the number of the 400 mesh stainless steel wire screen on each assembly, different sizes of aerosol particles will be collected on the six filter papers. By stripping the recorded activities on the filter papers by an iterative nonlinear algorithm (4), the activity size distribution in the range 0.002 μm - 0.5 μm can be obtained.

3. Ventilation measurement

Ventilation and air flow in the Exposure Room is monitored by a SF_6 tracer gas monitor (model 5302, B&K). The monitor detects SF_6 based on the photoacoustic infra-red absorption method.

4. ^{220}Rn & ^{222}Rn gas measurement

The gas inside the exposure room is continuously pumped through two Radiation Systems (model AB-5, Pylon Enc.) with scintillation cells, connected by a long delay tube. The first AB-5 measures the concentration of ^{220}Rn and ^{222}Rn, and the second measures that of ^{222}Rn alone after most ^{220}Rn decays inside the delay tube. In both AB-5 systems, the ^{220}Rn and ^{222}Rn daughters are first removed by a filter paper and only gas that decays inside the scintillation cells are recorded. The continuous radon concentration in the second AB-5 system is calculated by an algorithm (5) which takes into account of activities of previously deposited ^{222}Rn daughters and which also corrects for different humidity and flow rates. Concentration of ^{220}Rn is obtained by noting the difference in activities of the two scintillation cells.

5. Radon daughters measurement

The ^{220}Rn and ^{222}Rn daughters are sampled intermittently by a filter and Si detector assembly. The energy spectra of the radon daughters collected in the filter paper are recorded in a multichannel analyzer. Alternatively, three-count method (6) and five-count method (7) are used to determine the daughter activities and working levels. Unattached fraction of ^{222}Rn daughters are measured by computing the collected activities

on 2 filter papers, one of them is preceded by a single 400 mesh wire screen that collects the unattached daughters.

6. Radon Chamber (3)

The chamber is made of stainless steel and has a volume of 1.46 m^3. It is connected to the Exposure Room where samples of air can be taken for conducting experiments. Similar to the Exposure Room, radon gas and radon daughter concentrations, aerosol size distribution, activity size distribution, etc. in the chamber can be controlled and monitored.

METEOROLOGICAL CONDITION

According to information provided by the Hong Kong Royal Observatory, the mean atmospheric pressure is 101.3 kPa, ranging from 95.3 kPa to 105.3 kPa, being maximum in winter during the cold fronts, and minimum in middle of the year during troughs and typhoons. The mean temperature ranges from 8 °C to 32 °C, being hottest in July and August, and coldest in January.

The exposure room can, with the aid of the buffer room, simulate any environmental conditions that can be found in Hong Kong. The designed pressure can range from 96 kPa to 103 kPa and temperature from 10 °C to 35 °C. The ventilation systems in both the Exposure Room and Buffer Room contain pressure sensors which automatically maintain the pressure by varying the input and output air flows. Similarly, the air change rate can be controlled and maintained automatically.

MITIGATION METHODS

It is well known that some paints or wall papers are effective radon barriers and provide an economic passive method to reduce indoor radon level. In addition to that, the RURAL is designed to study active mitigation methods such as controlling the ventilation system. Since change in pressure and temperature will result in change of radon exhalation rate, so by quantifying this change with respect to temperature and pressure differential between the two sides of the walls or between interior and exterior of the walls, the reduction of indoor radon can be accessed. Plate out of radon daughters can be increased by use of air cleaners or ionisers (8). Devices using similar principle will be installed and tested in the ventilation system. The air flow pattern in wall surface is also known to affect deposition rates of radon daughters, hence air flow system will be studied and adjusted to provide maximum reduction of suspended radon daughter concentration.

Radon-220 daughters attract less attention in other countries but they are certainly important contributors to the total radon dose in Hong Kong. Mitigation methods aiming to reduce ^{220}Rn exhalation and working level will be studied.

CONCLUSION

Rather standard mitigation methods have been developed in western countries for the detached or semi-detached houses but little was done for high-rise buildings. In places such as Hong Kong where the population density is high and ^{220}Rn and ^{222}Rn exhalation from local building materials are also high (9), the population exposure to radon is significant. Though indoor radon can be removed efficiently by increasing ventilation, there are many situations where this is not economically feasible, such as inside commercial buildings, so other methods have to be used and these may involve modifications to the building structures and building services.

With the help of the RURAL, reduction of indoor ^{220}Rn and ^{222}Rn hazards in high-rise buildings under situations in Hong Kong can be investigated. It is hoped that appropriate mitigation methods can eventually be developed.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
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PAPER TITLE THE APPARATUS FOR CONTINUOUSLY MEASURING SIZE DISTRIBUTION OF RADON PROGENY

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MAJOR SCIENTIFIC TOPIC NUMBER 1.1 (see page 7)

ABSTRACT (See instructions overleaf)

The deposition of radon progeny in respiratory organs depends on the size of the radon progeny. In this study, the apparatus for continuously measuring size distribution of radon progeny has been developed. This apparatus consists of 10 ZnS(Ag) scintillation counters and 10 screen type diffusion batteries. In the diffusion batteries, various numbers (from 0 to 100) of 500-mesh wire screens are installed in series. The range of measurable particle diameter is from 0.016 to 1.6μm. Measurement can continue automatically for about 10 days, if the data are required every two hours.

Size distributions of radon progeny were measured in a tunnel where radon concentration was high. Simultaneously size distributions of aerosol particles were also measured using condensation particle counter and electrostatic classifier. The size distributions of radon progeny obtained by this apparatus agreed well with the calculated ones from aerosol data.

In outdoor air in Tokyo, the similar measurements were carried out for about two months. Activity median diameter of radon progeny were in the range of 0.095 to 0.34μm. The representative value of activity median diameter was 0.17μm. This was almost the same as 0.2 μm, which was reported in UNSCEAR 1988 REPORT.

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PAPER TITLE CHARACTERISTICS OF THE RADON DECAY PRODUCTS FORMED IN THE ATMOSPHERE ABOVE A RADIUM-BEARING WASTE PILE

AUTHOR(S) NAME(S) RAVIART S., RICHON P., HARISTOY D., HOBE M.C., BELOT Y.

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MAJOR SCIENTIFIC TOPIC NUMBER 5.1. (see page 7)

Short-term measurements (5 min. sampling) of radon-222 and its decay products concentration, and also of the concomitant meteorological parameters have been performed on a waste-rock pile (Crosses, Saxony) during the week 17-21 August 1994. As could be expected from measurements made directly above the source, the equilibrium factor is low (mean value at 0.5m-1.5m,0.13-0.16), while the daily radon concentration (152-136 $Bq.m^{-3}$) and the daily potential alpha energy concentration (0.122-0.136 $\mu J.m^{-3}$) are relatively high. The relationships binding these radiological parameters to the state of atmospheric stability has been investigated. The concentrations of radon and its daughters increase with the stability of the atmosphere. On the contrary, the equilibrium factor variations are not completely explained by the state of atmospheric stability. Other parameters such as wind velocity and dry deposition of radon decay products might be taken into account. The complexity of studying the outdoor equilibrium factor resides in the fact that, besides the necessity to integrate numerous explanatory phenomena, the equilibrium factor value above a finite area source is also influenced by the presence of radon from the regional background.

VARIATIONS OF Rn-222 CONCENTRATION IN THE BRATISLAVA AIR

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INTRODUCTION

^{222}Rn is produced by alpha decay of ^{226}Ra in soil. A small fraction of totally produced ^{222}Rn escapes from soil particles into soil air. Then ^{222}Rn is transported predominantly by molecular diffusion into outdoor atmosphere. The radon concentration in the outdoor atmosphere is not stable. It varies irregularly depending on meteorological conditions. However, there were found out regular daily and seasonal variations of ^{222}Rn concentration in outdoor atmosphere. These variations were measured in numerous works and results are summarized f.e. in work of Gesell (1). A simple model described the annual variations of ^{222}Rn concentration was published by Minato (2). A mathematical analysis of daily course of ^{222}Rn concentration in outdoor atmosphere was realized by Garzon et al. (3).

Some results of our study of ^{222}Rn variations in outdoor atmosphere of Bratislava are shown in this report.

METHODS

The ^{222}Rn concentration in the outdoor atmosphere has been studied at our department since 1987. Up until the end of 1991 the method of measurement of radon concentration in the outdoor atmosphere was as follows. Radon was concentrated from the air volume of 10 l on an active carbon and after its transfer radon was measured by means of the scintillation chamber of Lucas-type (4). A sampling was realized every morning at 9 o'clock in the hight of 1,5 m above the ground surface. Since Februar of 1991, radon in the outdoor atmosphere has been monitored continueously by a large volume scintillation chamber which volume is 4,5 l (4).

RESULTS

We obtained about 80 000 data of ^{222}Rn concentration (February 1991 - August 1995) by continual monitoring of radon in the outdoor atmosphere of Bratislava. This large number of data with great variability have to be processed statistically to reveal some regularities. Table 1. gives the arithmetic means of ^{222}Rn concentrations averaged throughout the years 1991-1995.

Table 1. ^{222}Rn concentrations [Bq.m^{-3}] - years 1991 - 1995.

Hour	Jan.	Feb.	Mar.	Apr.	May	June	July	Aug.	Sep.	Oct.	Nov.	Dec.	Mean
00.00 - 02.00	7,22	6,01	4,99	4,57	5,97	6,36	7,85	9,01	8,01	8,6	6,62	6,77	6,83
02.00 - 04.00	7,68	6,23	5,61	5,15	6,66	6,54	8,89	10,29	8,35	8,39	6,71	6,93	7,28
04.00 - 06.00	7,91	6,51	5,54	5,27	6,58	7,18	9,13	9,89	9,22	8,87	6,87	6,96	7,49
06.00 - 08.00	7,88	6,41	5,41	5,57	5,88	6,39	8,74	9,91	8,43	8,63	7,55	7,87	7,39
08.00 - 10.00	7,39	6,27	5,25	4,02	4,73	4,85	5,87	7,09	7,04	7,99	6,14	7,13	6,15
10.00 - 12.00	6,82	5,81	3,74	3,32	3,51	3,65	4,65	5,67	5,41	6,42	6,72	6,94	5,22
12.00 - 14.00	6,31	5,12	3,75	2,98	3,25	2,97	3,77	3,76	4,26	5,87	6,24	6,29	4,55
14.00 - 16.00	6,85	4,99	3,51	2,62	3,11	3,01	3,71	4,05	4,13	5,26	6,01	6,57	4,48
16.00 - 18.00	6,67	5,27	3,89	2,89	3,42	2,84	3,52	3,88	4,55	6,06	6,57	6,41	4,67
18.00 - 20.00	7,17	5,54	4,07	3,35	4,27	3,72	4,96	5,54	5,59	6,83	6,52	7,03	5,38
20.00 - 22.00	6,85	5,74	4,21	4,05	4,95	4,99	6,51	6,77	6,41	7,65	6,69	6,51	5,94
22.00 - 24.00	7,21	6,05	5,03	4,22	5,28	5,31	6,84	7,62	7,01	8,18	6,79	6,44	6,33
Mean	7,16	5,83	4,58	4,01	4,79	4,82	6,19	6,96	6,53	7,41	6,62	6,82	5,98

The results averaged in this way enable us to demonstrate the average daily courses of ^{222}Rn concentration for individual months and average annual courses for various time intervals.

The average daily course of ^{222}Rn concentration calculated on the basis of all measurements reaches a maximum between 4 and 6 a.m. and a minimum between 2 and 4 p.m. The ^{222}Rn concentration reaches its average daily value at 9: 30 a.m. and at 9 o'clock in the evening.The ratio of the maximum and minimum values in the average daily course of ^{222}Rn concentration amounts to 1.67. The daily variations of the ^{222}Rn cocentration are ascribed to variations of atmospheric stability and vertical mixing (5).

The average annual course of ^{222}Rn concentration calculated on the basis of continual measurements during the years 1991 - 95 reaches the maximum value in October and the minimum value in April. Further we can see in Table 1, that the average annual course of ^{222}Rn concentration calculated from all the data is in good agreement with the average annual course of ^{222}Rn concentration calculated on the basis of measurements made between 8 and 10 p.m.

Nearly five years lasting continual monitoring of the ^{222}Rn concentration in the outdoor atmosphere enables us to make an analysis of mean daily waves for different months of the year. According to Garzon et al. (3), the average daily course of the ^{222}Rn concentration in the outdoor atmosphere can be expressed by the following equation :

$$\frac{C(t)}{\bar{C}} = 1 + A_1 \cos\left(\frac{2\pi}{24}t + \Phi_1\right) + A_2 \cos\left(\frac{2\pi}{12}t + \Phi_2\right) \qquad (1)$$

where C(t) is the average ^{222}Rn concentration at the time t, $\bar{C}$ is the monthly average of the ^{222}Rn concentration, A_1 is the first harmonic amplitude, A_2 is the second harmonic amplitude, Φ_1 is the first harmonic phase, and Φ_2 is the second harmonic phase.

In this equation, the first harmonic term describes the turbulent dispersion process originated by solar heating. The second harmonic term describes the influence of the diurnal instability interval varying with the season. The values of the parameters in Eq.(1) are summarized in Table 2. They were obtained by the Fourier's analysis of the average daily courses of the ^{222}Rn concentrations.

Table 2. Results of Fourier analysis of the mean daily waves.

Month	A_1	A_2	A_1/A_2	Φ_1	Φ_2
1	0,08	0,04	1,87	298,2	173,3
2	0,11	0,04	2,78	295,9	117,5
3	0,22	0,05	4,75	302,8	162,4
4	0,32	0,08	4,16	303,6	158,5
5	0,36	0,06	5,58	315,7	186,6
6	0,43	0,06	7,69	309,4	174,2
7	0,45	0,08	5,91	309,9	175,2
8	0,46	0,07	6,93	306,7	167,4
9	0,35	0,08	4,74	305,9	156,7
10	0,22	0,06	3,75	312,5	116,3
11	0,05	0,03	1,49	307,3	142,4
12	0,06	0,05	1,23	263,1	145,7
Mean	0,26	0,06	4,24	302,6	156,4

Further we were looking for correlations between Φ_2 and the hourly difference between the sunrise and the sunset ΔH, and for correlations between A_1 and the intensity of solar radiation, so as it was done in Ref.(3). We determined the followed expression for Φ_2 and A_1 :

$$\Phi_2 = (34,8 \pm 23,3) + (9,2 \pm 1,7)\Delta H \qquad (2)$$

where ΔH was calculated for the 15 th day of each month and taken as the monthly average value,

$$A_1 = (-0,053 \pm 0,024) + (0,00069 \pm 0,00005)Q \qquad (3)$$

where the global solar radiation Q was calculated according to an equation published in (6), for the 15 th day in a month at 12 a.m. of the local time.

The phase Φ_1 of the first harmonic term is practically constant for all the months and its average value is equal to (302,6°±13,6°).

The amplitude A_2 of the second harmonic term shows only small variations during the year. Therefore we did not search for any correlation for A_2 and we calculated an average value :

$$A_2 = 0,056 \pm 0,015$$

CONCLUSION

In Table 3., there are compared the results of analysis of our measurements with results of analysis done by Garzon et al. (3) for data from various places in the world .

Table 3. The values of Φ_1 , Φ_2 and A_2 for the monthly mean waves at various places in the world.

	Φ_1	Φ_2		A_2
		$\Phi_2 = a_2 + b_2 \Delta H$	r^2	
Oviedo	276,8±3,7	-116,3+19,5ΔH	0,94	from 0,05 to 0,1
Socorro	278,2±9,7	-155,7+23,1ΔH	0,75	
Toulouse	276,4±8,4	12,1+10,6ΔH	0,98	
Brazaville	271,4±10,7	-1494+133,7ΔH	0,41	
Bratislava	302,6±13,6	34,8+9,2ΔH	0,90	0,056±0,015

The amplitudes and phases of the harmonic terms determined by Fourier's analysis of mean daily courses for different months have the same behaviour as those from Oviedo (3). The amplitude A_1 of the first harmonic term is well correlated with the global solar radiation fallen down on the Earth's surface.

The phase Φ_1 is constant and its value found out for Bratislava is about 10 % higher than its average value given in Ref.(3).

The expression of Φ_2 determined by us approaches mostly to the relation that is valid for data from Toulouse. The average value of the amplitude A_2 is between 0,05 and 0,1. This is in the range of the values found out also for the other places.

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POPULATIONAL EXPOSURE TO ^{210}Pb AND ^{210}Po IN TWO CITIES WITH VARIOUS INDUSTRIAL ACTIVITIES IN THE SOUTH - WEST OF ROMANIA

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INTRODUCTION

In the present work, we proposed to estimate the contribution of the thermal power plants (TPP) relying in fossil fuel to the modification of the natural radioactive background and implicitly to the population's exposure in two urban centers - district residences - Timisoara and Deva.

In the range of natural radioelements released in the atmosphere we considered as critical, ^{210}Pb and ^{210}Po for the following reasons:

- radiolead is one of the long lived daughters of ^{222}Rn in secular equilibrium with its daughter ^{210}Bi. It is a beta active hard element, generator of ^{210}Po, an alpha active element with high energy. ^{210}Pb, and ^{210}Po are in the group of very high radiotoxicity; ^{210} Bi high radiotoxicity;
- during the technological process of coal combustion of thermal power plants, there take place an accumulation of radiolead in the fly ash by a concentration rate of about 3 to 50.

MATERIALS AND METHODS

Two thermal power plants were taken into account: Timisoara Sud TPP, a power industrial of the city of Timisoara being partially set working; Mintia TPP, placed at a distance of 11 km from Deva, one of the greatest industrial units in Romania.

Within the framework of the study were performed radiochemical determinations of ^{210}Pb and ^{210}Po concentrations in atmospheric depositions (wet and dry cumulated), aerosols, food chain (drinking water, bread and milk) and processed food (daily menu).

Cumulated wet and dry atmospheric depositions were sampled monthly and analyzed quarterly. Aerosols were collected twice in a year by aspiration on membrane filter, with 40 to 60 m^3 aspired air volume.

The food chain were investigated twice in a year, by samples taken from centralized distribution sources. Processed food were investigated once in a year, by daily menus taken from representative sources.

All samples were mineralized with nitric and fluorohydric acid. After the concentration through coprecipitation of the microcompounds the critical radionuclides were wholly separated by spontaneous electrochemical deposition on pure nickel disks.

The radiochemical recovery output was 65% for ^{210}Pb and 80% for ^{210}Po.

Beta activity of the disks were measured after 30 days, using a NK-350 counting equipment with anticoincidence GM counter (background 4 cpm, counting efficiency 19.4%). Alpha activity of ^{210}Po were measured using a RFT 20026 counting equipment with ZnS scintillation detector (background 0.72 cpm, counting efficiency 18%).

RESULTS AND DISCUSSION

The data of the analyses are registered in tables 1 to 4

Table 1 indicates the total quantity of powders falling out on the soil surface. There are presented the quarterly and yearly values expressed in gm^{-2} . It shows a relatively moderated level of powdering in Timisoara and a level of about two times increased in Deva. In both cases there is a minimum value in the third quarter, in rest the powdering level is near constant.

Table 1 Powdering degree (total and quarterly values) in Timisoara and Deva

Investigated locality	Powdering degree (gm^{-2})				
	I^{th} quarter	II^{nd} quarter	III^{th} quarter	IV^{th} quarter	Total
Timisoara	20	20	14	18	72
Deva	50	47	22	40	159

Table 2 refers to the concentration of ^{210}Pb and ^{210}Po in atmospheric depositions. There are presented quarterly values in Bqm^{-2}. Both of the nuclides present higher levels in Deva, with increasing factors depending on the time of deposition .

Table 2. ^{210}Pb and ^{210}Po content in atmospheric deposition

Investigated locality	QUARTERLY CONTENT (Bqm^{-2})							
	^{210}Pb				^{210}Po			
	I^{th} q.	II^{nd} q.	III^{th} q	IV^{th} q.	I^{th} q.	II^{nd} q.	III^{th} q	IV^{th} q.
Timisoara	2.96	1.18	2.35	2.49	3.14	0.65	2.04	2.14
Deva	10.49	4.52	3.55	4.39	7.35	3.8	2.78	3.28

Table 3 presents the estimated total exposure levels through atmospheric deposition, with the necessary informative elements. It shows a near five times higher global (^{210}Pb and ^{210}Po) exposure level in Deva in comparison to Timisoara.

Table 3. Estimated global exposure levels

Informative elements	Investigated localities	
	Timisoara	Deva
Total deposition of ^{210}Pb (Bqm^{-2})	8.98	22.95
Total deposition of ^{210}Po (Bqm^{-2})	7.97	17.21
Total surface of deposition (ha)	$1.3\ 10^4$	$5.99\ 10^3$
Number of inhabitants	$3.34\ 10^5$	$8.05\ 10^4$
Global exposure level (kBq man year)	6.59	29.87

Table 4 presents the average value of internal irradiation levels for the population of the considered localities, by ingestion of processed food due to natural uranium, ^{210}Pb, ^{210}Po, ^{226}Ra.

Table 4. Internal irradiation of population by ingestion of processed food

Investigated locality	Effective dose ($\mu Sv\ y^{-1}$)				
	U_{nat}	^{210}Pb	^{210}Po	^{226}Ra	Total
Timisoara	2.9	59	4.14	2.75	68.79
Deva	1.8	155	3.15	3.05	163

CONCLUSIONS

1. ^{210}Pb and ^{210}Po from the fly ash of the TPP spread in the atmosphere under the form of sedimentable powder and fall out on the soil surface either through free falling, or in association with impurities or rain. Their concentrations in the aerosols was significant less in the atmosphere.
2. The population's exposure levels to ^{210}Pb and ^{210}Po was estimated to 6.59 KBq $man^{-1}year^{-1}$ in Timisoara and to 29.87 $man^{-1}year^{-1}$ in Deva; the levels of internal irradiation due to ingestion of processed food to 63 $\mu Sv\ y^{-1}$ in Timisoara, 158 $\mu Sv\ y^{-1}$ in Deva. The estimated data suggests the opportunity of some epidemiological studies upon the population living in the area of influence of these thermal power plants.
3. The monitoring of the ^{210}Pb and ^{210}Po content in atmospheric deposition constitutes an available indicator in finding out the modifications of the natural background induced by thermal power plants.

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EFFECTS OF WEATHER ON THE AMBIENT GAMMA RADIATION LEVELS IN HONG KONG

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ABSTRACT

A nuclear power station was recently constructed in the People's Republic of China some 50 km to the northeast of Hong Kong. A radiation monitoring network equipped with high pressure ionisation chambers was set up in Hong Kong to continuously monitor the ambient gamma radiation level over the territory. The network consists of ten field stations and one central station. Ambient gamma radiation levels at each station are transmitted to the central station via radio link and dedicated telephone lines once every minute. An alarm level can be set to each high pressure ionisation chamber to generate an alarm at the central station when the ambient gamma radiation level exceeds a preset threshold.

In the past few years, abnormal increase in ambient gamma radiation level was observed in some rainy days and onset of the winter monsoon. It was found that the ambient gamma radiation levels might increase by more than 100% of the background level on some of these occasions. These increases were primarily due to increases in the amount of natural airborne radionuclides, such as radon daughters. In rainy conditions, the downpour brought natural airborne radionuclides higher up in the atmosphere down to the ground while during the onset of the winter monsoon, the cool stable layer of air in the lower atmosphere tends to limit the dispersion of natural airborne radionuclides, resulting in increased ambient gamma radiation level.

INTRODUCTION

A nuclear power station was recently constructed in the People's Republic of China some 50 km to the northeast of Hong Kong. A radiation monitoring network (RMN) was set up in Hong Kong to continuously monitor the ambient gamma radiation levels over the territory and to provide an alarm should an abnormal increase in the radiation level be detected. The network consists of ten field stations (Fig. 1) and a central station. Each field station consists of a High Pressure Ionisation Chamber (HPIC), one high volume air sampler, one radioiodine sampler and one set of total deposition collector. The HPIC continuously monitors the ambient gamma dose-rates at the field station. Data in units of microgray per hour (μGy/h) are transmitted to the central station by both radio and dedicated phone lines at one minute intervals. An alarm threshold can be set for each field station. When the ambient gamma radiation level exceeds the threshold, an audio and visual alarm will be generated at the central station. The central station will then automatically send a command to that field station to turn on the high volume air sampler and the radioiodine sampler to collect air samples. The air samples and the total deposition samples will then be collected by staff on emergency duties for detailed isotopic analysis in a laboratory.

Studies of the RMN data showed that the average ambient radiation levels in Hong Kong varied from place to place from 0.09 μGy/h at Ping Chau to 0.14 μGy/h at Tsim Bei Tsui, amongst which about 0.03μGy/h originated from cosmic radiation. Shown in Fig. 1 is the average ambient dose-rates at the ten stations in 1994. The spatial variation of the ambient radiation levels is mainly due to the differences in the geology of the stations where the HPICs are located.

Since the operation of the nuclear power station in 1992, there was no incident at the nuclear power station which would result in accidental release of radionuclides. However, abnormal increases in the ambient radiation levels were detected by the RMN in rainy days and during onset of the winter monsoon.

SEASONAL VARIATIONS IN THE AMBIENT RADIATION LEVEL IN HONG KONG

Since the operation of the RMN in 1987, it has been observed that there is a marked seasonal variation in the ambient gamma radiation levels at each of the field stations. For all ten stations, the ambient gamma dose-rates in winter were higher than in summer. The differences varied from 3.6% at Tsim Bei Tsui

to 13 % at King's Park. Figure 2 shows the monthly average ambient gamma dose-rates of King's Park in 1994. Isotopic analyses of the air particulate samples collected at the station showed a similar seasonal variation in the activity concentrations of radon daughters e.g. Pb-212 and Pb-214 (also shown in Fig. 2). The main reason for the higher activity concentrations of radon daughters and subsequently higher ambient gamma dose-rates in winter was believed to be the prevalence of the northeasterly monsoon. The winter monsoon affected the ambient radiation level in two ways. Firstly, the airstream which originated from mainland China carried along with it an abundance of natural terrestrial radionuclides. Secondly, the stable layer of air in the lower atmosphere that appeared after cool northeasterly winds became established locally tended to limit the dispersion of natural radionuclides. Both effects would result in an increase in the ambient radiation levels. This is exemplified by the gamma dose-rate readings recorded at Yuen Ng Fan (YNF) on a cloudy but rainless day on 7 November 1995. Fig. 3 shows a 24-hour trend plot of the one-minute ambient gamma dose rate at YNF together with the wind field and surface temperature recorded by the nearest automatic weather station at Sai Kung, about 7 km west of YNF, over the same period. The figure shows that winds changed from moderate easterly to light northeasterly at about 4 a.m. as a result of the arrival of a surge of the winter monsoon. At the same time, the ambient gamma dose-rate started to rise slightly from 0.12 μGy/h to 0.13 μGy/h and remained at that level until early afternoon. The change in wind direction could have brought more natural radionuclides from mainland China to the territory resulting in a rise in the ambient gamma dose-rates. Just after midday, winds picked up significantly from the north and temperature started to fall from about 25 to 20 degrees Celsius at around 4 p.m. The fall in surface temperature created a cool layer of air at the lower atmosphere thus trapping the radionuclides. This was evident from the T-Φ diagram (insert in Fig. 3) which showed an inversion at the surface level up to about 500 meters after the arrival of the monsoon. This resulted in a further rise in the ambient gamma dose-rates in the evening of that day.

In contrast, during the summer months the airstreams are mostly of maritime origin which contain less terrestrial radioactivity. Thus a lower ambient radiation level would be expected.

EFFECTS OF PRECIPITATION ON THE AMBIENT RADIATION LEVEL IN HONG KONG

Short-term abnormal increases in ambient gamma radiation levels have also been observed in rainy days. The ambient gamma radiation levels typically increase by 30 to 50% of the background but have been observed to increase by up to 100% of the background level in occasions of heavy rain. Figure 4 shows a trend plot of the one-minute average ambient gamma dose-rates recorded by the field station at King's Park on 23 September 1994. It also shows the 24-hour trend plot of the hourly rainfall recorded at the Royal Observatory Headquarters, about 1.5 km from King's Park on the same day. The abnormal rise in the dose-rates which happened twice on that day was in phase with the two episodes of heavy rain. The downpour brought natural airborne radionuclides higher in the atmosphere down to the ground resulting in increased ambient radiation level.

CONCLUSION

Studies of RMN data showed that the ambient gamma radiation levels in Hong Kong are significantly affected by weather conditions. Precipitation has the largest effect which in some cases may result in abnormal rises in the ambient radiation level by up to 100%. The primary reason is that the heavy precipitation would result in a higher activity concentrations of radon daughters at the ground level and hence higher ambient gamma dose-rates. During the onset of winter monsoons, the northerly winds originating from China carry with them an abundance of natural terrestrial radionuclides. Furthermore, the cool stable layer of air in the lower atmosphere in winter tends to limit the dispersion of natural airborne radionuclides. These two factors would result in increased ambient gamma radiation levels in the winter months.

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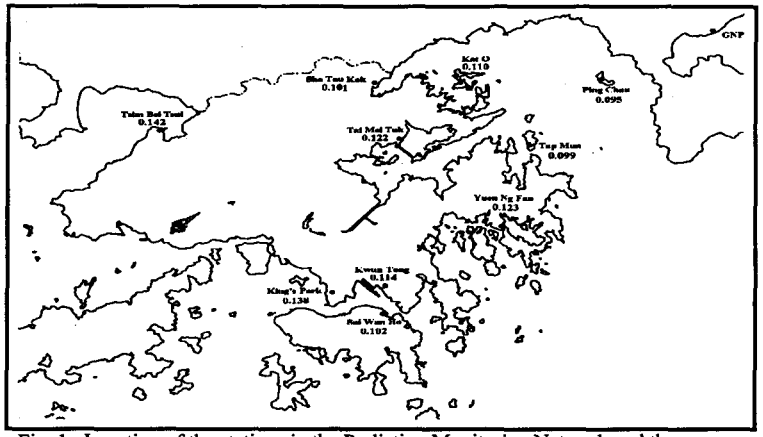

Fig. 1 Location of the stations in the Radiation Monitoring Network and the average ambient gamma dose-rates at the ten stations in μGy/h in 1994

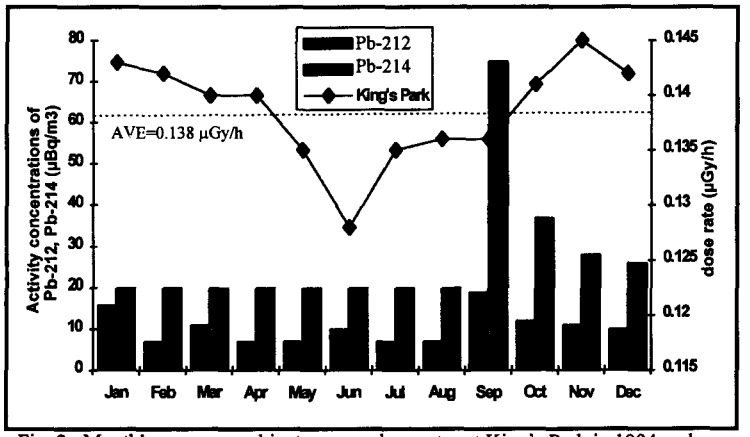

Fig. 2. Monthly average ambient gamma dose-rates at King's Park in 1994 and the activity concentrations of Pb-212 and Pb-214 in air particulate samples

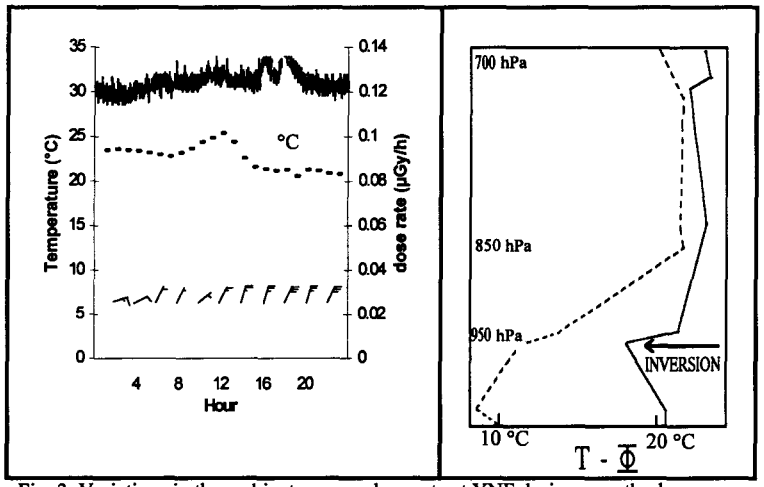

Fig. 3 Variations in the ambient gamma dose-rate at YNF during a northerly surge on 7/11/95. Insert shows T-Φ at 1200 UTC on same day at the King's Park Upper Air Station

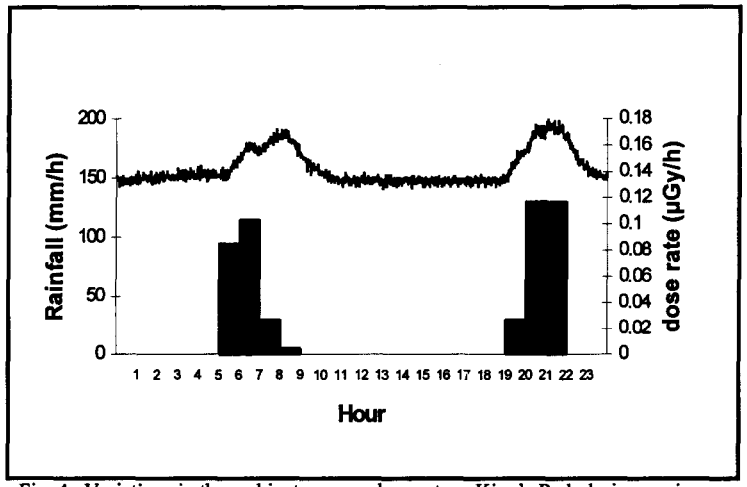

Fig. 4 Variations in the ambient gamma dose-rate at King's Park during a rainy day (23 September 1994)

CHARACTERISTICS OF RADIATION FIELD IN LIVING SURROUNDINGS AND IN NATURAL ENVIRONMENTS

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INTRODUCTION

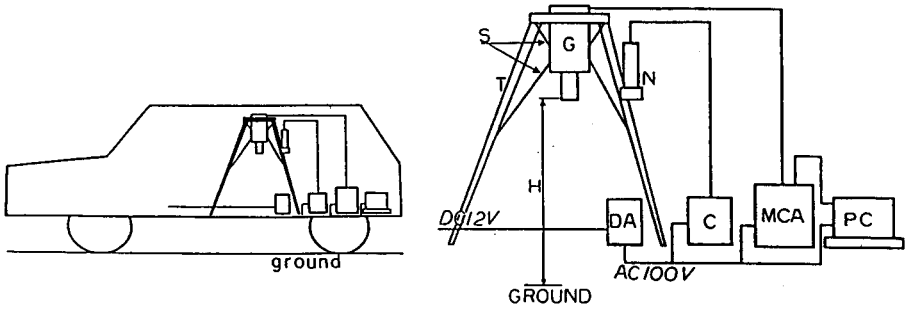
Levels of environmental radiation vary upon the environmental conditions and have own characteristics locally. Therefore it is very important to survey the distribution of environmental radiation levels anywhere for assessing its variation thereafter[1)2)].

For the porpose of grasping of environmental radiation fields in Hokkaido, a survey on the distribution of environmental gamma rays was made, using a portable spectrometer fitted with a high purity Ge semiconductor detector installed in a car - a car-borne survey -, during the period from Apr. 1994 to Aug. 1995. In addition to car-borne survey, in-situ monitorings were performed at spots sporadically .

In this paper, we will present the results obtained from the survey in Hokkaido Island, Japan.

METHOD OF MEASUREMENT AND INSTRUMENTS USED

Measuring method adopted for this survey was a car-borne survey in which measurements were made with the measuring systems installed on a car and made on cruising. Detectors used for gamma ray detection were a portable high purity Ge semiconductor detector and an NaI(Tl) scintillator detector. Figure 1 shows a schematic diagram of the measuring system installed on a car. During this survey we used the same car to be available the



T:Tripod G:HPGe detector N:NaI detector DA:DC/AC inverter C:Controller MCA:Multi-channel analyzer PC:Personal computer S:Hanging spring H:Height from the ground surface(=80 cm)

Fig.1. Schematic diagram of the measuring systems.

same conditioning not to vary the shield effect by car. Two detectors were set on a tripod to be fixed in the same position during the survey. Height of the detectors were 80 cm above the ground. Power from the cigar lighter of the car was supplied to the measuring systems through a DC/AC inverter.

Cruising speed of the car was kept 40 through 50 km/h. Every cruising of 40 or 50 km, spot monitoring was made. Measured intervals and spots were plotted on a map manually. The output from the high purity Ge semiconductor detector was stored on floppy disks by a personal computer. The output from the NaI(Tl) scintillator detector was printed out by its own printer.

TERM AND AREA FOR SURVEY

The survey over Hokkaido Island was taken place five times of cruising during the term from early spring in 1994 to summer in 1995. The areas covered were the proximity of Muroran City(i.e. proximity of Institute), northern part of Hokkaido, eastern part, south-western part, central part, and partially duplicated. Figure 2 shows the survey routes over Hokkaido.

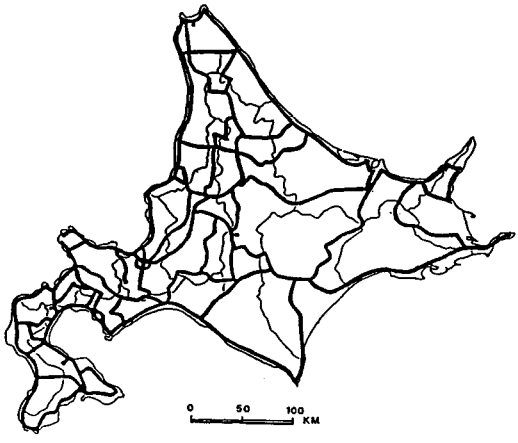


Fig. 2. Map of the survey routes over Hokkaido.

ANALYSIS OF DATA

Data obtained directly from the detectors are energy spectra of counts on gamma events. We used the method of the HASL-258 by Beck et al.[3] to discriminate nuclides of the gamma sources from the obtained spectra and summed up classifying into uranium-series, thorium-series, potassium-40, and fallout nuclides. The shield effect by car was 26 %.

RESULTS AND DISCUSSIONS

Figure 3 shows an example of the analyzed result obtained from the survey on eastern part of Hokkaido mainly. The lower part of the figure, lines denote the values of exposure rate in air in μR/h due to uranium-series, thorium-series, potassium-40, respectively, and dots denote the total values of dose rate in nGy/h. The upper part of the figure shows ratio among three groupes of nuclides mentioned above.

The figure shows not only remarkable variation on total absorbed rate against difference of area but also remarkable variation on constitutive ratio among uranium-series, thorium-series and potassium-40. In the figure, exposure rate of thorium-series varys remarkably as variation of location where measurement was made, but those of uranium-series and potassium-40 vary a little. Therefore amount of thorium-series seems to contribute mainly to the variation of the total absorbed dose rate.

Figure 4 shows a summary of the analyzed results on total absorbed dose rate in nGy/h over Hokkaido Island obtained from

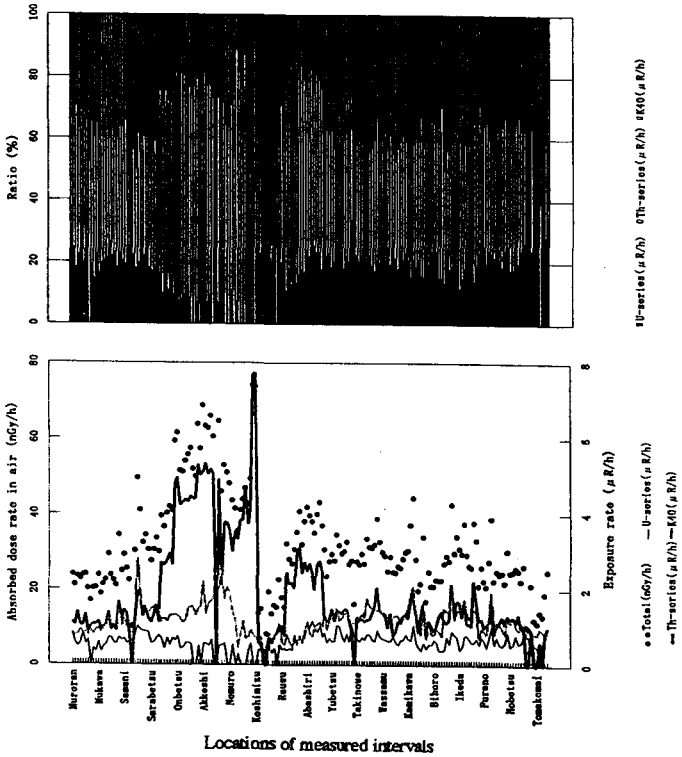


Fig.3. An example of the analyzed result as the analyzed results, obtained mainly from the survey on eastern part of Hokkaido.

the survey carried out during the period from Apr. 1994 to Aug. 1995.

Higher total absorbed dose rates were measured sporadically in some areas over Hokkaido Island, i.e. in northern part, in eastern part, and in south-western part of it. In these areas radiation due to thorium-series were higher than other districts. And also in some areas along survey routes, remarkable jump-up or jump-down of total absorbed dose rate was sporadically found.

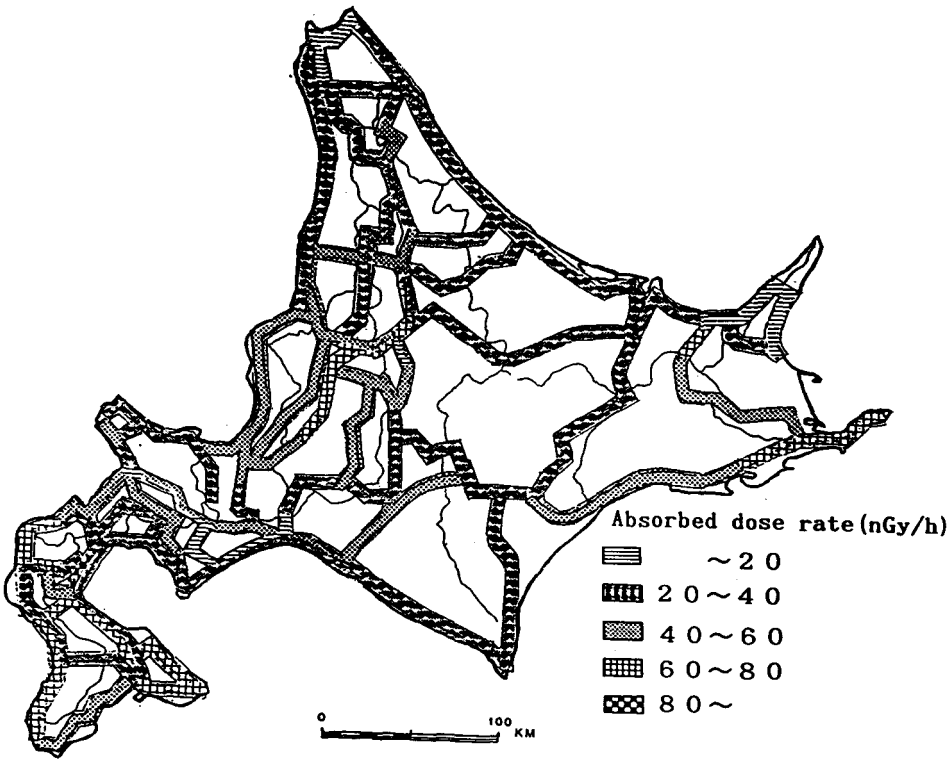


Fig.4. Summary of the analyzed results on total absorbed dose rate in nGy/h over Hokkaido Island.

CONCLUDING REMARKS

A survey on distribution of environmental gamma-rays was made along the almost all main roads in Hokkaido Island, using a portable spectrometer fitted with a high purity Ge semiconductor gamma ray detector installed in a car - a car borne survey -, during the period from Apr. 1994 to Aug. 1995. The values of absorbed dose rates in air due to gamma rays were found to distribute from 20 to 106 nGy/h, and mean dose rate was calculated to be 38.2 nGy/h.

The variation of total absorbed dose rate was seemed to be originated mainly in the variation of dose rate due to gamma rays radiated from the nuclides of thorium-series. Higher total absorbed dose rates were measured in some districts over Hokkaido Island, i.e. in northern part, in eastern part, and in south-western part of it. In these districts radiation due to thorium-series were higher than other districts.

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^{238}U SERIES ISOTOPES AT DIFFERENT SOIL DEPTHS AND DISEQUILIBRIUM OVER VARIOUS GEOLOGY AND SOIL CLASSIFICATIONS ALONG TRANSECTS IN SELECTED PARTS OF IRELAND.

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Abstract - Sampling of soils was carried out along linear transects in selected regions of the country, a technique known as Transect Sampling. This was a controlled rather than a random sampling technique. The transects were located in regions which were previously known to contain high levels of the ^{226}Ra isotope, from the ^{238}U series. The soil sampling was carried out at selected sites along these transects. At each transect site, two different soil depths were examined and the soil samples collected were identified as the top and bottom soil samples. This transect data set, consisting of the isotope activity levels and the influencing variables transect geology and soil types, provided a data base for investigation.
Comparisons were made between the soil isotope activity levels measured at different soil depths. An examination of the ^{238}U decay series showed the existence of disequilibrium. Relationships between the disequilibrium data and the associated geology and soil types were investigated.

INTRODUCTION

Previous surveys of radioactivity in soil in Ireland had shown that radium concentrations were considerably higher than average in a small number of regions (1 - 2). Comparison of these regions with a geological map indicates that some of the high values for radium in soil activities might be associated with certain categories of underlying geological strata.

To investigate the existence of such an association, it was decided to take a series of soil samples at selected sites along a number of linear transects. These transects were laid down with the assistance of geological maps in such a way that they crossed a number of geological strata within regions where previous surveys had shown greater than average concentrations of radium in soil samples. The sites were selected so that each lay within the boundaries of a geological stratum as indicated on the map and the number chosen gave a good coverage of the different strata lying along the transect. Because of the uneven distribution of underlying geological strata, the sites were not evenly spaced. The length of the transects was between 30 and 50 km and was chosen to cover several different geological strata in each case.

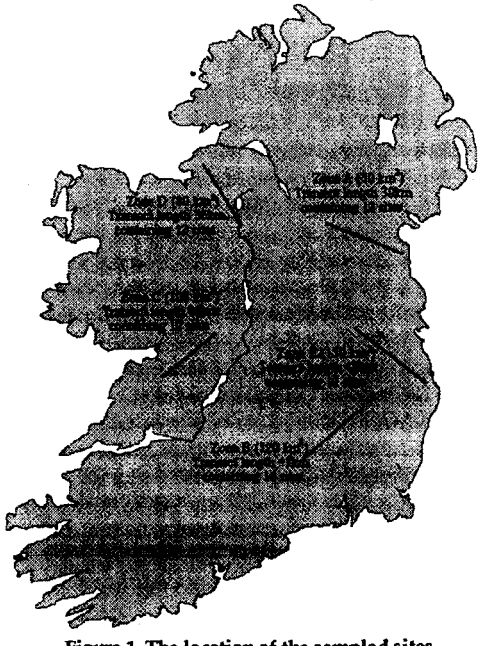

Figure 1. The location of the sampled sites

Figure 2. A section of the Geological map of Ireland showing the location of the first transect in the north-east in county Louth

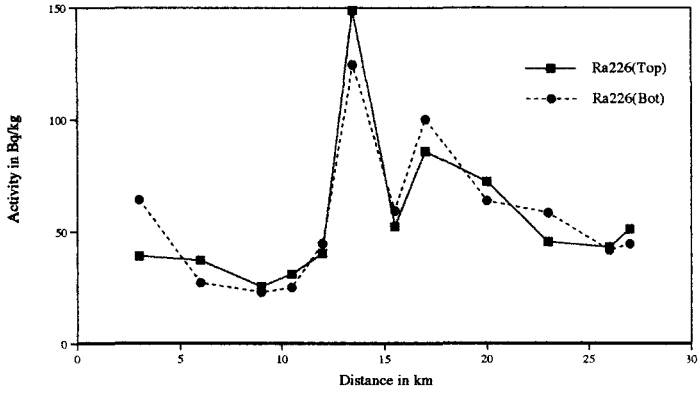

Figure 3. ^{226}Ra top and bottom activity levels at the Louth transect

SOIL SAMPLING

The selected transects are shown in figure 1 and details of the first transect are shown superimposed on the geological map in figure 2. Two soil samples were taken at each site, one of surface soil and one of soil from below the agricultural horizon which was assumed to be at a depth of 25 cm beneath the surface. The surface soil sample was made up by taking equal quantities of soil from 12 randomly located points within an area of approximately 1000 m^2. These sub-samples were mixed, dried and ground to a fine consistency (particles > 1 mm diameter) after removing stones and pebbles of greater than about 1 cm diameter. The deep soil sample was taken at a single location at each site and was obtained by removing soil from the bottom of a hole dug at the location. The uranium content was determined by measuring the ^{234}Th activity from the principle gamma ray in the prepared sample and the radium concentration was found using the 186 keV gamma ray emitted in the decay of ^{226}Ra.

The results from the first transect are shown in figure 3. It is clear that there is no significant difference between the samples obtained from surface soil and those taken below the agricultural horizon. The figure also indicates that the ^{226}Ra activity levels show variations related to the nature of the underlying strata.

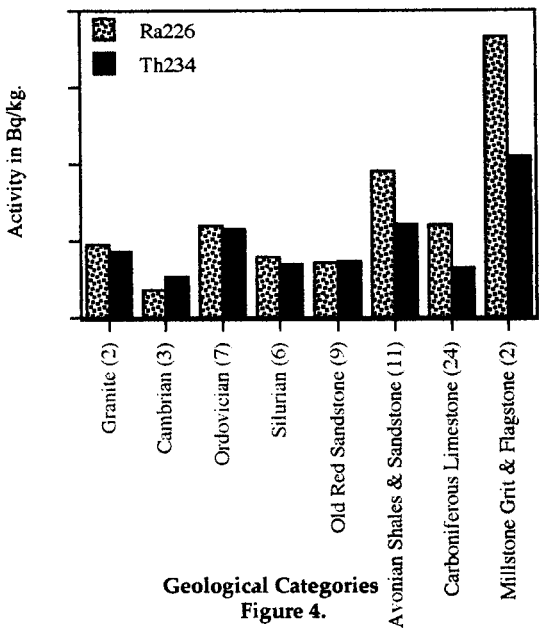

Figure 4.

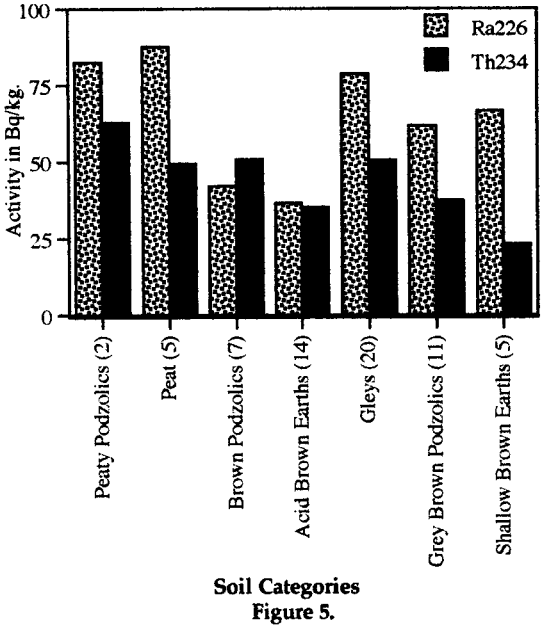

Figure 5.

Ra/U Disequilibrium over the Geological and Soil Categories of all transects

DISCUSSION OF RESULTS

The mean activities for ^{226}Ra and ^{234}Th for all transects are shown on figures 4 and 5, as a function of geological category and soil category respectively. The numbers shown in brackets for each category represent the number of sites within that category for which measurements were made. Although there are some indications that radioactive disequilibrium exists for a number of the different categories shown, the number of sites is rather small for several of the classifications used. However, for those categories within which there are more than about six samples and a difference in activities of greater than about 15 Bq/kg between ^{234}Th and ^{226}Ra, the measured activities support the hypothesis that a radioactive disequilibrium exists. The results indicate that the disequilibrium is particularly well confirmed for soils overlaying carboniferous limestones and avonian shales and sandstone. The disequilibrium is in the direction of an excess of ^{226}Ra and, when considered as a function of soil category, is most marked for gleys and grey brown podzolics.

It is concluded that geological and soil maps can, at least in the conditions that pertain in Ireland, be a useful indicator of areas in which radioactive disequilibrium may occur in soil. Measurements of soil radioactivity may also be used as indicators of changes in underlying geological strata on a relatively small scale. Further work will be carried out to investigate possible associations of these results with high values of radon in the domestic environment for the areas surveyed.

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INDOOR EXPOSURE TO GAMMA RADIATION IN ITALY

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ABSTRACT

A representative survey to evaluate the exposure to natural radiation in dwellings was conducted in Italy from 1989 to 1994. The survey was coordinated by ANPA (former ENEA/DISP) and Istituto Superiore di Sanità, and it was carried out in collaboration with 19 Regional Laboratories. Beside radon concentration, absorbed dose rate in air was also measured. For gamma measurements, whose results are reported in the present paper, ENEA collaborated, too. Thermoluminescent dosemeters (TLDs) were used as detectors. Four TLDs were exposed for six months in a bedroom of each surveyed dwelling. Particular attention was devoted to calibration and treatment procedures of the dosemeters (annealing, transport dose etc.), according to detailed protocols. Intercomparison exercises were conducted for all the Laboratories involved. The national and regional averages were calculated from the results in the actual sample (4234 dwellings) taking into account the sampling scheme. The national average absorbed dose rate in air was around 105 nGy/h, after the subtraction of the cosmic ray contribution. This value is located in the high range of the national average values of other Countries given in the 1993 UNSCEAR Report. Using the dose factors assumed in the same report, the estimated effective dose due to indoor gamma exposure at home results about 0.4 mSv/y. Appreciable differences exist among the average dose rates measured in various Italian Regions. The highest values were found in Central Italy where natural building materials with high uranium and thorium content are widely used.

INTRODUCTION

In the period 1989-1994, a representative national survey was conducted in Italy, to evaluate the indoor exposure to natural radiation, both radon and gamma radiation. The characteristics of the survey, its organisation, the sampling criteria and the general policy adopted have been described elsewhere (1,2,3). Although it is well known that radon is the main source of indoor exposure (4), measurements of outdoor gammas carried out during the seventies and indoor gamma measurements conducted in the eighties, pointed out the existence of areas where the exposure rate was particularly high (5,6). These areas are characterised from the geological point of view by the existence of effusive and explosive materials (recent volcanitis); moreover, building industry generally makes use of local materials which significantly contribute to the indoor exposure (6).

MATERIALS AND METHODS

As described in ref. 3, 19 Regional Laboratories participated in the survey. Standardised measurement protocols were used in order to assure comparability and to minimise uncertainties. Detailed information on the procedures for gamma measurements are reported elsewhere (7) and are summarised here. Thermoluminescent dosemeters were used, in particular LiF sintered chips from the same batch doped with Mg and Ti or Mg, Cu and P. Two dosemeters

were positioned in one room of each dwelling, usually the main bedroom, and exposed aside the radon detectors for six months. Each dosemeter was made of two chips enclosed in a light-tight plastic sachet of 100 mg cm^{-2} thickness. As far as quality assurance procedures are concerned, intercalibration exercises were carried out, exposing dosemeters both to a reference Cs–137 gamma beam (see Fig. 1), with a dose of 500–1500 μGy, and to environmental radiation in high background room, where the dose rate (~320 nGy/h) was measured by means of a ionisation chamber.

RESULTS AND DISCUSSION

Gamma measurements were carried out in 18 Regions. The national distribution of absorbed dose rate in air detected indoors is shown in Fig. 2. A cosmic ray contribution of 32 nGy/h has been subtracted from TLD results. This contribution has been estimated applying a building shield factor of 0.8 (4) to the measured value of 40±9 nGy/h (6), which is compatible with the value calculated taking into account latitude and altitude. The national average value is 105 nGy/h (3), with an estimated overall uncertainty of about 10%, which takes into account both statistical and systematic errors. The world population weighted average assessed by UNSCEAR is 83 nGy/h (4). This value was obtained from surveys of different statistical significance, whose averages are generally lower than the Italian one, e.g. 71 nGy/h in Austria, 60 nGy/h in UK and 37 nGy/h in USA, while only few values are equal or higher, such as 103 nGy/h in Australia, 105 nGy/h in Portugal and 110 nGy/h in Sweden. Regional averages in Italy are quite different, with the lowest figures in Emilia–Romagna, Toscana, Veneto and Puglia (40–50 nGy/h), and the highest ones in Lazio and Campania (>210 nGy/h). For the dose assessment, an average occupancy factor at home equal to 0.6 – obtained from the survey questionnaires (3) – was applied to the average dose rate in air. The estimated gamma radiation contribution to the effective dose at home resulted around 0.4 mSv/y.

CONCLUSIONS

The average absorbed dose rate in indoor air, obtained through the Italian national survey, is about 105 nGy/h, which is rather high when compared with the values obtained in other national surveys. The regional averages show a large variability over the Italian country: the highest values have been detected in those Regions, such as Lazio and Campania, where local building materials with high uranium and thorium content are widely used. These data stress the importance of studying natural building materials, as both gamma radiation and radon sources, in order to reduce the presence of radioactivity indoors. In authors' opinion, a regulation to limit the radioactivity content and the radon exhalation rate for building materials of future dwellings should be promulgated.

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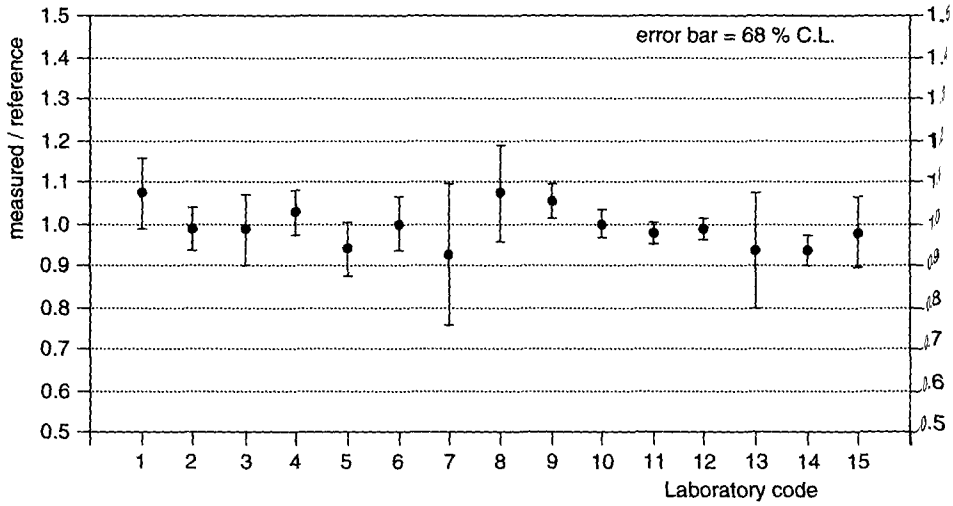

Figure 1. Results of the intercomparison exercises with a reference Cs–137 gamma beam.

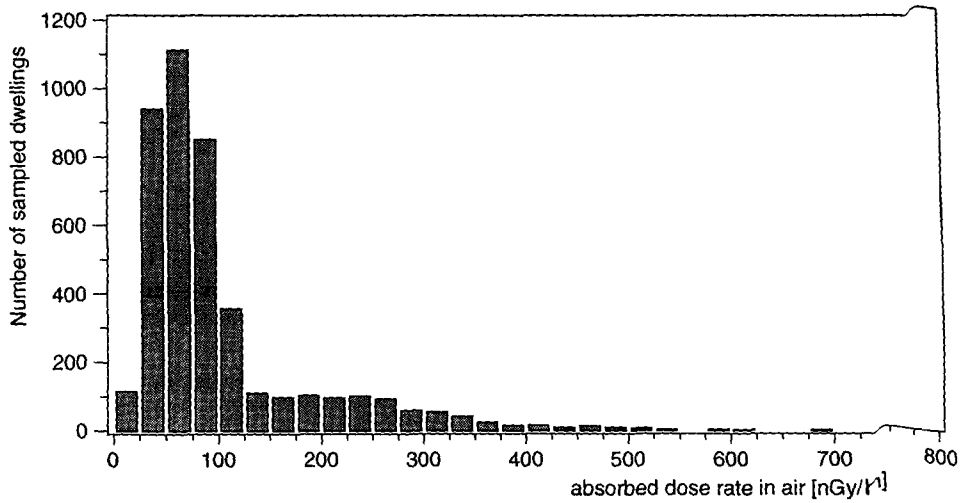

Figure 2. Distribution of the absorbed dose rate in air due to gamma radiation in the sampled dwellings. The cosmic ray contribution, estimated as 32 nGy/h, has been subtracted from measured values.

THE ADDITIONAL EXPOSURE DUE TO THE USE OF UNCOMMON BUILDING MATERIALS IN ROMANIA

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INTRODUCTION

Beginning with 1981, the district ionizing radiation hygiene laboratories, charged by the Ministry of Health with the surveillance of the work places and the environment, have signaled the attempt of the industry to introduce new highly radioactive materials in the building products. Phosphogypsum, by products of the phosphate fertilizer industry, and slag and ash, wastes of the coal-fired power plants, are the most important of them.

In order to establish radioactivity levels and limits for the building materials and the exposure due to their use, some studies have been elaborated by the specialized laboratories of the Ministry of Health. As a result, admissible limits for the building material radioactivity have been chosen the maximum values obtained for the building materials commonly used, are presented in Table 1.

The additional exposure and fly ashes have been assessed by two ways:

1) The measurements of the potential alpha energy concentrations, PAEC (mWL) and equilibrium equivalent concentrations (Bq/m^3) for radon level were carried out in a flat with internal walls of phosphogypsum. The measurement results were compared both with those obtained for the flat without internal walls and those obtained for the houses from Romania (50% of the country).

2) For the same flat of the experimental house, annual effective dose were calculated.

MATERIALS, METHODS AND RESULTS

Samples of phosphogypsum obtained from the main producers of phosphate fertilizers and samples of the fly ashes collected on filters came from 18 coal-fired power plants were analysed by the high resolution gammaspectrometry method using a 4096 MCA Canberra systems equipped with Ge(li) detectors. Results are presented in Table 2.

An experimental detached house (basement, ground floor, garret) was built with usual building materials (concrete 15 cm thick and aerated concrete bricks 30-35 cm thick for walls, concrete 20 cm thick for floor and ceiling, the flat of ground floor having the internal walls of phosphogypsum (7 cm thick).

In order to estimate the contribution of the Rn level in the flat, PAEC or EEC levels were determined by Nazaroff's method [1] occasionnally doubled by Sima's method [2]. The annual effective dose given by the building materials have been estimated on the basis of the data delivered by UNSCEAR Reports 1982 and 1993 [3,4] and W van Dijk [5].

The results are presented in Table 3. The exposure generated by the phosphogypsum only partly used in a flat is very high compared to both ground floor values (4-5 times higher) and the mean value obtained for the houses in Romania (2-3 times higher).

CONCLUSIONS

As a result of these studies, the Ministry of Health included two articles regarding the use of the building materials in the "Regulations concerning the radioprotection of the population and environment"

a) "The natural radioactive elements in building materials should correspond to the raw materials from which are derived, the intentional addition of the natural and artificial radioactive elements being prohibited"

b) "The use as building materials of the sterile, slag and by-products of the fertilizer industry, with a natural or artificial content higher than the value of the products commonly used in the building industry is forbidden"

The use of phosphogypsum as building material for social dwellings was banned in any way (no mass proportion was addmitted). In the case of the potential building materials containing fly ash, each solution should be analysed and licensed by the Ministry of Health on the basis of the Ra-226, Th-232 and K-40 radionuclide concentrations of the products and dose assessements.

Table 1. Concenrations of the ^{226}Ra, ^{232}Th, ^{40}K radionuclides in the building materials commonly used in Romania (Bq/kg)

Type of material	No. of samples	^{226}Ra		^{232}Th		^{40}K	
		Arithmetic mean	Max.	Arithmetic mean	Max.	Arithmetic mean	Max.
Red brick	32	35.9	**100.0**	32.2	**53.3**	493	**833**
Cement	25	33.9	**66.3**	17.8	**97.0**	152	**504**
Natural gypsum	14	17.8	**43.0**	9.6	**27.0**	103	**277**
Lime	8	13.3	**41.0**	8.2	**18.5**	68	**167**
Concrete	16	27.8	**78.5**	20.0	**38.5**	201	**452**
Mortar	4	5.9	**7.8**	5.9	**12.2**	426	**611**
Aerated concrete brick	7	16.7	**32.2**	15.6	**36.7**	163	**277**
Sand, gravel	14	7.8	**30.0**	27.4	**91.7**	557	**870**
Various rocks	11	25.2	**63.0**	21.5	**75.9**	434	**1370**
Clay	3	24.8	**30.4**	49.3	**66.7**	861	**1140**

Table 2. Concentrations of the ^{226}Ra, ^{232}Th, ^{40}K radionuclides in the phosphogypsum and fly ash (Bq/kg)

Type of material	No. of samples	^{226}Ra		^{232}Th		^{40}K	
		Range	Arithmetic mean	Range	Arithmetic mean	Range	Arithmetic mean
Phospho-gypsum	54	237-970	702	11.1-42.2	22.6	60-167	113
Internal walls of the experimental house (phospho-gypsum)	10	554-809	659	6.7-32.2	13.1	15-43	18
Fly ash	18	64.4-218.5	154.6	73.6-125.9	88.8	277-945	569

Table 3. Additional exposure for the flat with internal walls of phosphogypsum.

	Houses in Romania	Flat without internal walls from phospho-gypsum	Flatwith internal walls from phospho-gypsum	Ratio		Additional exposure (mSv/year)	
	1	2	3	3/1	3/2	3-1	3-2
EEC(radon) Bq/m^3 measured	a=21.6 g=17.2	a=11.7 g=10.1	a=54.0 g=48.5	4.6 4.8	2.6 2,8	-	-
Total effective dose (mSv/y)	1.4	0.9	4.2	3	2.1	2.8	3.3

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RADIATION EXPOSURE POTENTIAL FROM COAL-FIRED POWER PLANTS IN ROMANIA

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INTRODUCTION

Thermoelectric power generated by coal represents about 30% of the electric power production in Romania (1). The combustion of coal results in release to environment of naturally occuring radionuclides. These are reasons for the research about the radiological impact of coal-fired power plants on the surrounding zones and population living there.

With that end in view we selected ten representative modern and older coal-fired power plants in Romania with a total installed capacity of 2.5 Gw.

MATERIALS AND METHODS

This work performed during 1984-1994, involved the measurement of activity concentrations of natural radionuclides in coal (235 samples) bottom and fly-ash (820 samples) originating from the power plants. Soil, vegetation and snow samples (910) have also been collected at 20 locations from the surroundings of each thermal power plant within a 5 Km radius from the stack. All the samples were counted for their natural radioactivity content using gamma ray spectrometry techniques and radiochemical methods (2,3).

RESULTS AND DISCUSSION

In the investigated power plants they burn brown coal, lignite and/or mixture of different kinds of coal : brown coal, lignite, pit coal, pitch coal, bituminous coal. The activity concentrations measured in the coal samples varied over two orders of magnitude (Table 1). The natural radionuclide concentrations in fly ash are significantly higher than the corresponding concentrations in the coal (Table 1).

The normalized discharged activities for the investigated power plants are much higher than those estimated in the UNSCEAR 1988 Report for typical old and modern plants (Table 2). Firstly, accounting for this is the low ash retention efficiency of the particulate control devices of power stations, especially for the older ones,and secondly,the high ash content of the coal: 26-60%. The low quality of coal leads to the higher coal consumption; thus the combustion of up to 20.10^9 Kg of coal is required to produce 1 Gwa of electrical energy.As a result,the activities of radon-222 and of radon-220 released per Gwa have been assessed at 25 to 770 GBq (2,3).

Table 1. Activity mass concentrations of natural radionuclides in coal and ash samples ($Bq.Kg^{-1}$)

Radio-nuclide	Brown coal	Lignite	Mixture of coal	Bottom ash	Collec-ted fly - ash	Escaping fly - ash
^{238}U	39[x]	41	30.6	38	71	80
	7-101[xx]	1.5-96	1.1-112	2 -160	3 -312	4 - 420
^{226}Ra	38	74	53.5	69	113	126
	1 -91	8 -152	4 -120	5 -237	4 -528	6 - 558
^{210}Pb	-	-	-	-	206	210
					10 -500	11-510
^{210}Po	-	-	-	-	240	260
					10 - 540	11-580
^{232}Th	30	25	16.5	49.5	59	62
	5 -45	10 - 35	1 - 43	1.5-147	2 - 170	2.2-170
^{228}Th	-	-	-	-	70	79
					1.2- 175	1.0-175
^{40}K	310	274	305	526	500	547
	230-590	182-490	30-615	158-1000	160-1200	160-1300

x - Arithmetic average ; xx - Range

Table 2. Estimates of annual atmospheric discharges per unit energy generated (GBq per Gwa)

Radionuclide	Coal - fired power plants (10)	
	Modern (7) [x]	Old (3) [xx]
^{238}U	0.19 - 50	28 - 281
^{226}Ra	0.07 - 47	28 - 201
^{210}Pb	0.17 - 86	52 - 300
^{210}Po	0.50 - 85	60 - 330
^{232}Th	0.04 - 17	10 - 65
^{228}Th	0.06 - 22	13 - 80
^{40}K	10.50 - 236	200 - 915

x - Fly-ash removal efficiency 96-99.5%
xx - Fly-ash removal efficiency 60-80 % ; in 1993,the worst of these plants was modernized.

Mesurements in soil, snow and vegetation had clearly shown the presence of increased concentrations of the natural radionuclides in the surroundings of the coal - fired thermal power plants,particularly for the oldest and most poorly controlled ones (Table 3).

Table 3. The average activity concentrations of natural radionuclides in soil,vegetation and snow (Bq kg^{-1})

Natural radionuclide	Soil (560 samples)		Vegetation	Snow
	the upper 5cm layer	the 5-15 cm layer	175 samples	175 samples
^{238}U	33 ± 30	26 ± 25	51 ± 46	$(4.2\pm3.1).10^{-3}$
^{226}Ra	38 ± 30	30 ± 27	30.8 ± 7.5	$(2.6\pm2.0).10^{-2}$
^{232}Th	36 ± 17	29 ± 16	83 ± 61	$(2.5\pm1.5).10^{-3}$
^{40}K	535 ±180	420 ±170	451 ±100	$(4.5\pm3.8).10^{-1}$

The resulting normalized collective effective doses (Table 4) have been estimated taking into account: the activity released per electrical energy unit; the population density; the wind rose and the deposition (dry and wet) velocity; the effective stack height (ranging from 55 m to 220m) (4,5).

Table 4. Estimates of normalized collective effective doses (manSv per Gwa)

Coal-fired power plants	Inhalation during the cloud passage	Irradiation due to deposited activity		Total
		Internal	External	
Modern	0.06-2.39	0.17-2.31	0.009-0.101	0.24 - 4.8
Old	10.4-34.5	7.2 -31.0	2.4 - 10.5	20 - 76

CONCLUSIONS

1. The population living in the neighbourhood of a inadequately controlled coal fired power plant is exposed to enhanced levels of natural irradiation.

2. The mass of $384x10^6$ Kg fly ash annually released to the atmosphere from ten coal - fired thermal power plants was considered to be the main cause of disturbance of natural radiation background and of enhancement of population doses.

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THE PRESENCE OF ^{226}Ra IN THERMAL SPRINGS OF CROATIA AND HEALTH RISK FROM MINERAL WATER CONSUMPTION

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INTRODUCTION

The presence of ^{226}Ra in water and possible health effects require particular attention considering the great radiotoxicity of ^{226}Ra, its long half-life ($T_{1/2}$= 1.622 yrs) and its high potential for causing biological change (1). Numerous studies have shown that many thermal water springs around the world contain relatively high levels of natural radionuclides, in particular ^{226}Ra which may present a possible risk of increased exposure to ^{226}Ra in the general population (2).

The purpose of this study was to determine ^{226}Ra concentrations in thermal water springs in the Republic of Croatia at six sampling sites, mostly spas and health resorts. These thermal waters have been used for medical and bathing purposes, for therapy, rehabilitation and recreation, and also for drinking for its good mineral composition.

Radium contamination in the samples of thermal waters was measured in order to estimate the radiation doses received by the population - patients or tourists during a stay in the spa, and to calculate the radiation dose originating from drinking spring waters. Equivalent dose received from a medical drinking cure over a four week period was estimated and compared to the dose received from ingestion of public system tap water over a year.

The Republic of Croatia is one of the richest countries in hot springs in the world and many health resorts are based on thermal and mineral water springs. Considering popularity of the spas and habit of the Croatian population to use beneficial effects of these springs it is of interest to estimate to what extent is radioactivity of thermal and mineral waters absorbed by bathing and drinking practices.

MATERIAL AND METHODS

The samples were collected several times over the period of last four years at six locations of selected thermal water springs, most popular tourist spas and health resorts, majority of which are clustered in North-western Croatia.

In all samples ^{226}Ra was determined by alpha spectrometric measurements after radiochemical separation. The counting time for each measurement was 60000 s or longer.

RESULTS AND DISCUSSION

Values of the average ^{226}Ra concentrations measured at six locations of selected thermal water springs are given in Figure 1 showing deviations from the average values in the observed period.

In the spas in which waters were used for bathing and drinking, ^{226}Ra activities measured in the samples of thermal waters did not exceed maximal permissible level of 1000 Bqm^{-3} for drinking water (3). In Croatia, health-based acceptable level of exposure from drinking water is based on the latest IAEA recommendation taking 1000 Bqm^{-3} as the maximum permissible level for ^{226}Ra in drinking water which is a derived concentration (DC) for a group of individuals (3). Guideline activity refers exclusively to ^{226}Ra concentration in municipal drinking water supplies and there is no regulation that would refer to the occurrence of ^{226}Ra in other water categories, such as thermal and mineral waters, and to possible contribution by routes of exposure other than ingestion. In other words, natural mineral water has not been included in the existing Croatian legislation on drinking water and has not been taken into consideration as a distinct category of water although the analyses show that ^{226}Ra concentrations contained in thermal and mineral waters are higher than in other categories of water in Croatia.

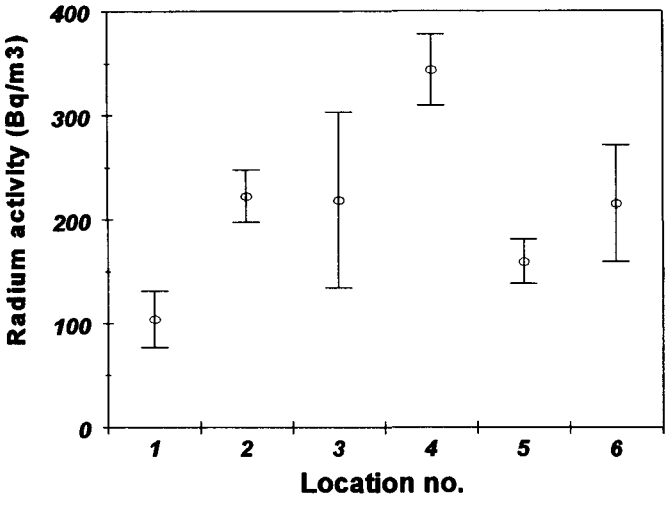

Figure 1. Average ^{226}Ra concentrations measured at six thermal springs

According to the study by Marović and co-workers ^{226}Ra concentration measured in the tap water in several major urban centres of Croatia was 2.1 ± 1.4 Bqm^{-3} (4).

In selected spas all waters have been used for medical and bathing purposes, and also recommended for drinking cure. The water from location no. 1 (Figure 1.) known for its good mineral content, is bottled and commercially sold in the country and abroad.

Considering relatively high ^{226}Ra content in thermal waters of the spas and health resorts of Croatia and on the assumption that a drinking cure contains 1 L of mineral water daily for one week, a possible influence on body burden from drinking this water would be higher than from consumption of the tap water over a year.

Estimating the dose due to consumption of spring waters with increased radium content, it has to be emphasized that these waters are applied as medicinal waters for definite periods and in given amounts. Therapeutical application usually takes daily doses of 0.2-1 litres for several weeks.

Table 1. The effective dose received from a medical drinking cure

Effective dose over four weeks (mSv)		
Sampling location no.	**0.2 L/day**	**1 L/day**
1	$1.28 \cdot 10^{-4}$	$6.41 \cdot 10^{-4}$
2	$2.74 \cdot 10^{-4}$	$1.37 \cdot 10^{-3}$
3	$2.70 \cdot 10^{-4}$	$1.35 \cdot 10^{-3}$
4	$4.24 \cdot 10^{-4}$	$2.12 \cdot 10^{-3}$
5	$1.97 \cdot 10^{-4}$	$9.83 \cdot 10^{-4}$
6	$2.65 \cdot 10^{-4}$	$1.33 \cdot 10^{-3}$

The assumed received dose is compared to that received by consuming only tap water over a year. The effective doses received from a medical drinking cure, that is 0.2 L/day of thermal water over the period of four weeks (Table 1) were estimated to be 1.3 times higher than the dose received from drinking 2 L/day tap water over a year ($3.37 \cdot 10^{-4}$ mSv). The doses received from 1 L/day of thermal water over a four-week period estimated to be as were much as 2 to 6 times higher, than the dose received from drinking tap water over a year.

According to ICRP recommendations (5) the limit for public exposure should be expressed as an effective dose of 1 mSv in a year. Since the doses obtained in our study are well below the recommended dose of 1 mSv the practice of using thermal waters (bathing and drinking for medical therapy, recreation and rehabilitation) should not be restricted provided that other sources of exposure are taken into account and controlled.

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THE NATURAL RADIOACTIVITY OF SOME FOODSTUFF FROM MOLDOVA

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INTRODUCTION

This work is a contribution to the national programme for the supervision of the food radioactivity and provides data on natural radionuclides in wheat, maize and potatoes.

These foodstuffs are major part of Romanian population diet (32%) therefore they are being cultivated on over half an area of main crops in Romania.

MATERIALS AND METHODS

During 1984-1994 there were taken from the territory of districts Suceava,Botosani,Neamt,Iasi,Bacau,Vaslui,Vrancea and Galati, samples of wheat grains, corn and potatoes. We made up 900 average samples which represented about 30% from all the area of these crops in Moldova.

The taking of samples (in the harvest time) and their analysis were carried out by standardized methods in Romania. The samples of 1 Kg of grain respectively peeled potatoes were dried, calcinated and wet ashed. Arising acid extracts were utilized for the assay of natural radioelements: uranium-238, thorium-232, radium-226, polonium-210 and potassium-40. Uranium-238 and thorium-232 were calculated after determining the content of natural uranium and thorium by the method based on their separation and purification on a strong basic anion exchange resin and spectrophotometric measurement in the form of their Arsenazo III complex. Radium-226 was determined through its decay descendent radon-222, and by alpha rays measurement in a scintillating chamber. The assay of polonium-210 was done by electrochemical deposition and alpha counting of polonium-210 deposited on the nickel disc in a low background ZnS (Ag) scintillation counter. Potassium-40 was found by calculation after the photometric dosing in flame emission mixture of potassium natural isotopes.

RESULTS AND DISCUSSION

Natural radioactivity levels due to uranium-238,radium-226 polonium-210,thorium-232 and potassium-40 which were determined in wheat grains, maize and potatoes are presented in Table 1.

There is no significant difference between the yearly determinations values in the investigating period of eleven years

with but few exceptions. The samples prelevated in 1984-1987 have had increased values for uranium-238 and radium-226. We suppose that the intensive use of chemical fertilizers in that period is responsible for this. During the eighties, a mean quantity of 40 Kg $P_2 O_5$ per hectare was used in Romania for wheat, and about 150 Kg $P_2 O_5$ per hectare for potatoes (1,2). The measurements of the uranium-238 and radium-226 content performed in our country, showed values of up to 580 Bq/Kg fertilizer respectively 351 Bq/Kg fertilizer for the superphosphate (1,3).

Table 1. The activity concentrations (mean and limits) of the natural radionuclides in some foodstuffs

Food product	^{238}U mBq/Kg	^{226}Ra mBq/Kg	^{210}Po mBq/Kg	^{232}Th mBq/Kg	^{40}K Bq/Kg
Wheat (grains)	39 6.1-84.5	155 10-455	135 11-370	12.3 1.6-33.2	79 35-152
Maize	32 2.4-69.8	72 10-268	89 10-372	8.2 0.8-18.9	68 30-115
Potatoes (peeled)	24 6.1-115	30 9-120	51 15-138	1.7 0.4-2.05	121 35-285

One can see from our data the high variability of natural radioelements values in the samples of food products, very likely due to the various activities of the soil in the sampling zones (4).

Potatoes from Suceava, Botosani and Neamt districts, showed somewhat higher local activities for all the five nuclides with the highest uranium-238 content.

The values are generally comparable to those found in the other zones of Romania and are situated between the variation limits of concentration values determined and reported for the other countries in areas of normal natural radioactivity (1,5,6, 7,8).

The annual ingestion of natural radioelements per capita was assessed for each of the three food-products, taking into account the yearly average consumption. Yearly average consumption of food products per capita in natural units represents the quantities of food products effectived for human consumption irrespectively of the consumption type (gross or processed), of supply source (food stores, free market, consumption from own resources of agricultural producers, etc) as well as the consumption place (households, restaurants, canteens, etc) (2,9).

The relative contributions of three major dietary categories to the total diet intake are listed in Table 2.

In the previous studies (8,9) there were determined natural radioelements in samples of different food products (representing components of human diet), menus and drinking water. The annual ingestions of natural radioelements was calculated and were used in Table 2 for the data comparing. One observes that the cereals products and the potatoes have the greatest contribution to ingestion of radium-226 and polonium-210.

Table 2. The annual ingestion of natural radionuclides (Bq y^{-1}) (region Moldova - Romania)

Food product	^{238}U	^{226}Ra	^{210}Po	^{232}Th
Wheat *	1.69	7.21	9.65	0.62
Maize *	1.42	3.02	6.80	0.43
Potatoes	1.31	1.30	2.64	0.09
Total dietary ingestion in Moldova (8,9)	6.2-11.2	12.4-20.7	24.8-40.1	1.6-4.4

* - Cereals products in equivalent flour

CONCLUSIONS

1. The natural radioactivity of the analysed foodproducts generally varied in the order: wheat grain maize potatoes excluding potassium-40, which was higher in potatoes.

2. The contribution of wheat and maize(equivalent flour) and potatoes, which are the main components of Moldavian diet, to the total dietary intake of natural radionuclides is over 60 %.

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NATURAL RADIONUCLIDE CONCENTRATIONS OF CEMENTS IN IZMIR AND DOSE ASSESSMENTS

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ABSTRACT

The growing demand of electric power and the large domestic deposites of lignite coal have made coal-fired plants grow in number in Aegean Region of Turkey. Some of this coals like Yatağan lignites are known to have high uranium concentration (315-405Bq/kg for coal, 746-1076Bq/kg for collected fly ash). The stockpiles of fly ash of these power plants are readily available for industrial uses as in the case of cement production. Therefore to assess the doses arising from building materials especially from cement in the cities of Aegean Region a project has been started beginning from Izmir.

The traces of radium, potassium and thorium in 45 cement samples which are collected from building constructions in Izmir have been analysed by gamma spectrometry. The mean concentrations of Ra-226, Th-232 and K-40 were determined. The indoor radionuclide doses were calculated using the mean concentrations found in the measurements of this work. The results were compared with those given for other countries.

INTRODUCTION

In Turkey as well as many other countries one can observe increasing interest in industrial and extractive wastes as substitutes for natural products in building industry. They usually have higher content of naturally occuring radioactivity than traditionally applied raws.

Knowledge of the radioactivity levels of materials used in the building construction is useful in the assessment of indoor radiation exposure of the population. This is also useful for the development of standarts and guidelines for the use and management of these materials for building low background laboratories.

EXPERIMENTAL

The materials were obtained randomly from local building material suppliers and manufacturers. Different brand names were sought. All samples were oven-dried at 105° C for 16h. and then sealed in 100g. portions in cylindrical polyethylene boxes. They were kept for 1 month to attain radioactive equilibrium between radium and radon for high energy γ-spectrometric analysis. The natural radioactivity of the building materials was determined from their Ra-226, Th-232 and K-40 contents. The eU, eTh and % K contents were determined from 1.76MeV Bi-214, 2.62MeV Tl-208 and 1.46MeV K-40 lines, respectively, with a method given elsewhere (Killeen, 1979). The sample and the standard γ-spectra were taken for 4000s. with the spectrometer consisting of NaI(Tl) crystal, 4096 channel Ortec .7010 analyzer and related electronic components.

The activity concentrations of the radionuclide in the samples were given by (Chung-Keung Man et al. ,1989)

$$A_i = [(C(E_i) - B(E_i))/(m\,f\,t\,P(E_i))] \qquad (1)$$

A_i : is the activity concentration of radionuclide i.
$C(E_i)$: is the net γ-counts above continum at the characteristic energy E_i
$B(E_i)$: is the background counts at E_i
m : is the mass of sample in kg.
f : is the branching ratio of the γ-emission at the energy considered.
$P(E_i)$: is the absolute efficiency at energy E_i

To compare specific activities of these materials a common index is required to obtain the total activity. This index called radium equivalent activity (Ra_{eq}) is based on the fact that 0.37Bq/g of Ra-226, 0.26Bq/g of Th-232 or 4.8Bq/g of K-40 produce the gamma dose rate (M.R. Menon et al. 1987).

$$Ra_{eq} = A_{Ra} + (A_{Th} \times 1.43) + (A_K \times 0.077) \quad (2)$$

where A_{Ra}, A_{Th}, A_K are activities of Ra-226, Th-232, K-40 respectively in Bq/g. A criterion formula corresponding to a maximum of 0.37 Bq/g is

$$H = A_K / 4810 + A_{Ra} / 370 + A_{Th} / 259 < 1 \ (A \text{ in Bq/kg}) \quad (3)$$

If the value of H is less than 1, the result signifies that the radiation hazard is about the same as that existing in a cave in the ground. The dose rate D, within a building is estimated from the emperical relation

$$D(\text{mrem/year}) = 1.16C_K + 11.45C_{Ra} + 17.8\,C_{Th} \quad (4)$$

where C is concentration in pci/g (K. Mamont-Ciesla et al. ,1983)

RESULTS AND DISCUSSION

The natural radioactivity in a variety of cement samples has been measured by gamma ray spectrometry . The concentrations of Th-232, Ra-226 and K-40 measured in a selection of cement samples available in İzmir are listed in table-1. The natural radioactivity of cement samples varies considerably depending on their origion. In the last columns radium equivalent concentration and dose rate are given. It can be seen that the Ra-equivalent and H values of samples measured in cement samples are lower than the limits given by Eq.2-3. The gonods, the active bone marrow, and the bone surface cells are considered by the UNSCEAR as the organs of interest. For this reason, many dosimetry models can be found in the literature for these organs. The maximum permissible gonadal dose from gama radiation due to the radioactivity in building materials amounts to 112 mrem/year. For comporison, the dose from the terrestrial radiation in İzmir (Turkey) varies from 34.41 to 89.73 mrem/year. These values can be compared with similar measurements performed by researchers other countries, they are also listed in Table 2.

Table 1. Natural Activity Concentration in Bq/kg of Cement Samples in İzmir

Sample	*eU (Ra-226)*	*eTh (Th-232)*	*%K*	Ra_{eq}	$C_K/4810 + C_{Ra}/370 + C_{Th}/259$	*Estimated gonadal dose mrem/year*
tc*	108.0	39.7	363.5	192	0.51	61.9
pc*	83.7	23.8	270.1	138	0.37	45.7
ppc*	128.2	16.9	198.9	175	0.51	56.3
wpc*	97.7	21.3	130.3	138	0.37	44.5

*tc: tras cement, pc: portland cement, ppc: portland cement with fly ash, wpc: white portland cement

Table 2. Activity Concentration in Bq/kg of Cement Samples from İzmir and Other Countries

Countries	*K-40 (Bq/kg)*	*Ra-226 (Bq/kg)*	*Th-232 (Bq/kg)*
Norway	241.7	30.0	18.5
U.K.	155.0	22.0	18.0
USSR	148.1	25.9	14.8
Germany	222.0	25.9	22.2
USA	126.0	----	14.8
Finland	241.0	44.4	26.0
India	114.2	86.0	25.6
İzmir (Turkey)	241.0	104.4	25.4

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DETERMINATION OF CONCENTRATIONS OF SELECTED RADIONUCLIDES IN SURFACE SOIL IN THE STATE OF KUWAIT

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INTRODUCTION

Exposure from radioactive nuclides of terrestrial origin has remarkable input to the total effective dose equivalent for human population. The substantial part of this exposure is external irradiation, which depends on activity mass concentration of primordial radionuclides in surface soil, mainly ^{238}U, ^{232}Th series and long-lived ^{40}K. From geological point of view the surface soil of Kuwait is relatively uniform: desert type with prevailing smooth sand sheets and one cannot expect the big differences in surface concentration of natural radionuclides, except of coastal deposits. However, Kuwait territory was heavily contaminated during the Gulf War due to large quantities of released crude oil and the atmospheric discharges from incomplete oil combustion from the burning of oil wells. Although oil and natural gas contain only trace quantities of the naturally occurring long-lived radionuclides, because of the large scale of atmospheric discharge the emitted activities were apprised as : 6.7 G Bq- ^{238}U, 10 GBq - ^{226}Ra and 5 GBq- ^{232}Th (1). In addition, the use of uranium tipped antitank shells during the Gulf war was another source of possible uranium contamination of the surface soil of Kuwait. Therefore continuous background radionuclide concentration measurements were initiated as a part of a program for assessment of radiation doses from natural sources for Kuwait population.

METHODS

The location of the 69 sites where soil samples were taken are shown in Fig. 1. These locations are non random and are positioned in the Kuwait City, its vicinity, along major highways and in the special geological formations. The several samples were also collected from the Gulf War battle-field (Mutla area). Three samples from the uninhabited islands: Guruh, Maradin and Kubar were also analyzed. The soil samples (top 5 cm) were collected from 100 sq. cm surface- three samples from each site. All samples were dried at 110° C and 1.5 kg fraction which passed through 12 mesh sieve was placed in the plastic Marinelli beaker and stored for three weeks to allow build up of radon and its daughters. The samples were counted using high purity Ge detector and the spectra were analyzed using Sampo-90 program. The detector was calibrated by adding the mixed radionuclide standard - QCD 1 (Amersham) to soil sample followed by homogenization of sample.

Concentrations of ^{238}U were determined directly from 92.6 keV and ^{40}K from 1461 keV peaks. In identifying ^{226}Ra the five principal γ-rays were analyzed: 295, 325, 609, 1120 and 1765 keV from ^{214}Pb and ^{214}Bi radionuclides. Also for analyzing of ^{232}Th five γ-lines: 239, 583, 727, 911 and 969 keV from its daughters. ^{208}Tl, ^{212}Bi and ^{228}Ac were used. Additionally the concentration of ^{137}Cs deposited after nuclear bomb tests were also determined.

RESULTS AND DISCUSSION

The calculated average concentrations of measured radionuclides are as follows:

^{232}Th ---	10.0 ± 3.5 Bq kg^{-1}	ranged from 0.57 to 16.3 Bq kg^{-1}
^{238}U ---	13.4 ± 5.2 Bq kg^{-1}	ranged from 7.0 to 30.7 Bq kg^{-1}
^{226}Ra ---	11.8 ± 4.4 Bq kg^{-1}	ranged from 2.25 to 22.8 Bq kg^{-1}
^{40}K ---	329.3 ± 111 Bq kg^{-1}	ranged from 4.4 to 496.5 Bq kg^{-1}
^{137}Cs ---	2.8 ± 2.6 Bq kg^{-1}	ranged from 0.1 to 9.9 Bq kg^{-1}

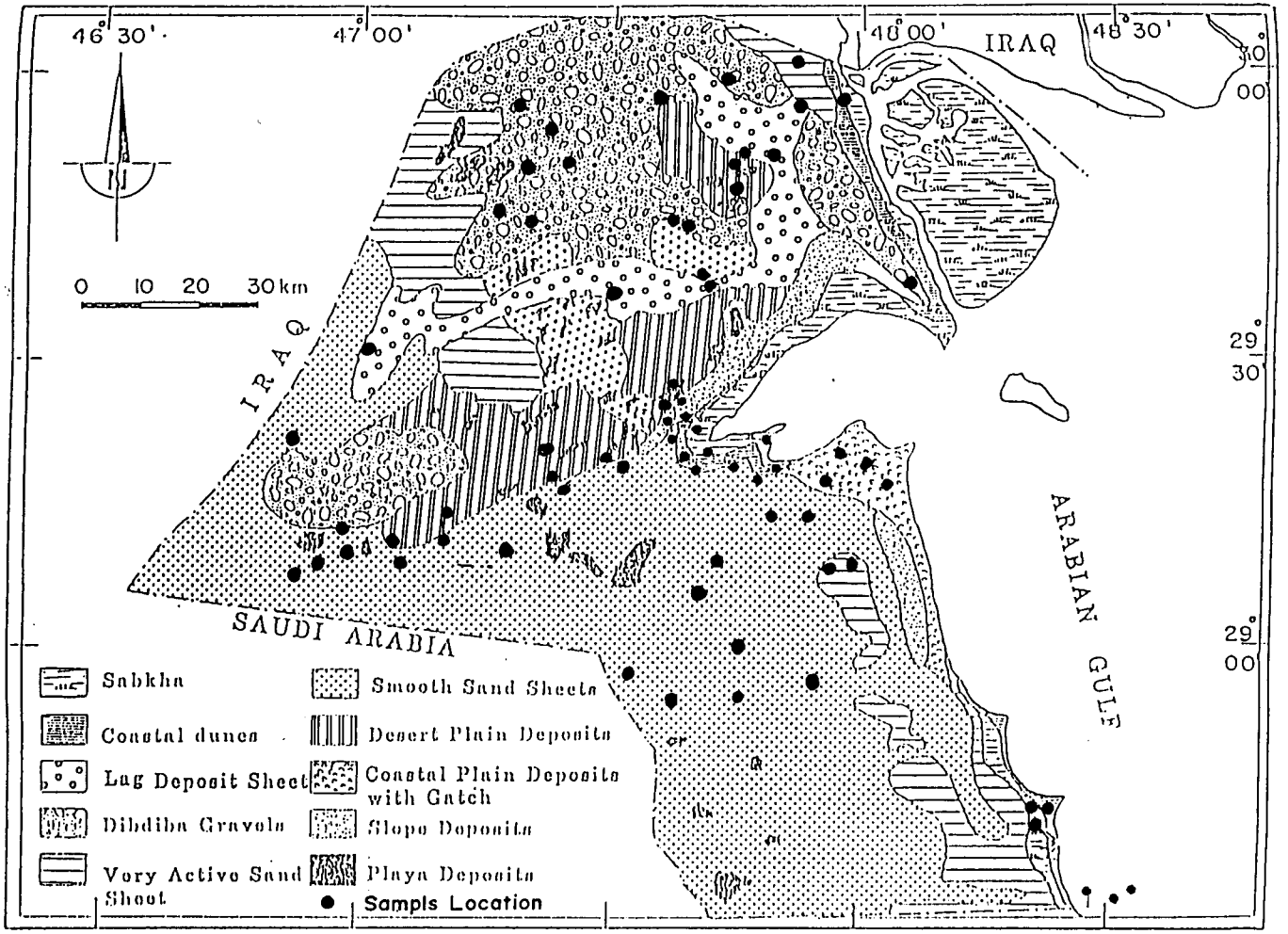

Fig.1. Surface deposit distribution in Kuwait and location of background soil samples

The highest concentrations of ^{238}U were observed in the samples from coastal area (Ras Al-Zoor) and from the island soils. The same samples showed very low ^{232}Th, ^{226}Ra, and ^{40}K values. No significant differences in uranium concentrations for Mutla area and other places have been found. The average concentration of ^{232}Th and ^{238}U in Kuwait surface soil were lower than the world average value of 40 $Bq\text{-}kg^{-1}$ (2) since the large area of Kuwait is covered with desert sandy soil. In the most cases daughter radionuclides of uranium and thorium series were found in a radioactive equilibrium except of those from Ras Al-Zoor and islands-samples.

On the basis of these measurements and dose rates per unit concentration in soil, adopted from UNSCEAR (3,4) the average absorbed dose rate in air (at 1m height) from gamma field of ^{232}Th, ^{238}U series and ^{40}K radionuclide was estimated on 0.234 mGy per year for Kuwait territory.

ACKNOWLEDGMENT

This study was supported by a grant SG042 from Kuwait University.

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RADIATION EXPOSURE OF POPULATION DUE TO CARELESS USE OF RADIOACTIVE RAW MATERIALS IN THE URALS

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HISTORY and BACKGROUND

In the middle of the forties a deposit of thorium monacyte sand was found in the Urals. Thorium monacyte is a physically and chemically resistant mineral, practically insoluble in water. The formula for monacyte is $(Ce,La,Y,Th)P_2O_5$. The monacyte sand contains high concentrations of long-live natural nuclide ^{232}Th. Characteristic ThO_2 concentration in sand is about 3-9%.

During 1949-64 in village Ozerny located in 70 km to the north-east from Ekaterinburg a concentrating mill processing thorium containing monacyte sand for the purposes of nuclear and military industry was functioning. In summer the mill processed local staff and in winter the staff brought from another places. The territory of railway station Kostousovo was used for loading and storage of staff and as a result of this the territory of temporary monacyte sand storage was subjected to technogenic radioactive contamination.

After the concentration mill has ceased its operation the careless use of thorium monacyte sand for private and public construction, for paving the roads along with natural dispersion of sand took place. Thus in the village and at the railway station some centres of technogenic radioactive contamination by natural radionuclides has been formed.

One of the reasons of careless use of radioactive raw materials was the absence of objective information on the character of the staff.

OBJECTIVES

Our main objective was to single out all cases of radioactive contamination for planning the rehabilitation and deactivation measures and for calculating the accumulated exposure dose for population of village Kostousovo. It was assumed besides, that the main ways of population exposure beyond the natural radioactive background was the external gamma-irradiation and inhalation receipt of ^{220}Rn (thoron) along with its daughter products in houses built with the use of thorium-containing materials. It was also necessary to ascertain the level of natural radioactive background for comparing it with the additional irradiation.

METHODOLOGY

For revealing the centres of radioactive contamination the method of continuous pedestrian sampling point by point with distance 5 m (scale 1:5000) at the territories of common use (streets, vacant lots, railway station), in private gardens, along the external fronts of buildings and indoors - ceilings, walls, stoves was conducted. Scintillation radiometers were used.

The measuring of equivalent equilibrium concentration (EEC) of thoron was conducted by aspiration method implying aerosol adsorbtion on filter with further measuring of alpha-activity on the filter. Besides the equivalent volume activity of ^{222}Rn (radon) is revealed, being an important share in population exposure. The obtained results of seasonal samplings were converted to mean annual activity by the methodology tested and applied in the Urals. This methodology accounts for typical value of equilibrium factor and relation between seasonal and annual concentrations. For the purposes of evaluating the ^{232}Th supplies the samples of soil were taken in the areas with elevated gamma-irradiation. To estimate the specific activity of long-live radionuclides in air and water we conducted air- and water- sampling.

RESULTS

The whole territory of village and 235 private and public buildings were tested for radiation exposure dose rate. 16 places of concentration of radiologically contaminated staff were found along with 25 cases of using thorium-containing materials for building purposes.

Total contaminated territory is about 10200 m^2 The most extensively contaminated spot is at the former loading site of the concentration mill (8000 m^2), now it is a vacant lot. Other contaminated centres are located in the vicinity of, or directly in residence zone; their square is from 1 up to 1.300 m^2. Total Th concentration at the contaminated territory is about 1.6 Ci. Such Th concentration was formed as a result of spreading of about 300 tons monacyte sand (5% concentration) by a thin layer of 1 sm.

The range of radiation exposure dose rate in the middle of the most contaminated site was from 0.8 up to 1.0 mR/h, and Th specific activity in ground samples reaches 600 kBq/kg.

In all 25 tested houses the elevated rate of exposure dose (up to 150 μR/h) is caused by non-controlled use of monacyte sand as building material. As a result such parts of buildings as the fundamentations, wall plasterer, stove grout and garret fillings were radioactively contaminated. The total Th content in building materials do not exceed 0.15 Ci.

These materials also served the source of thoron emanation . The volume activity of thoron in contaminated buildings was 2.5-15 Bq/m^3, whereas the mean value for the village in non-contaminated buildings equals to 1.7 Bq/m^3 and doesn't exceed 2.5 Bq/m^3. The mean value of indoor volume radon activity in the village is 77 Bq/m^3. There was no difference in radon concentrations in "clean" and contaminated houses in the village.

The air and ground-water samplings don't show the elevated level of contamination (exceeding the background).

DISCUSSION, DOSES

It was estimated that the natural background radiation is rather high at this territory. Mean outdoor exposure dose at the territory of the village is equal to 20 μR/h, and indoors - 23 μR/h. Indoor radon concentrations are also rather high.

Basing on the results of estimating the external dose of gamma-irradiation in private houses in village Kostousovo the collective exposure dose was calculated. Population of the village is 430 people. Taking into consideration that population spends indoors about 7000 hours per year this dose equals to 56 cSv/year. Besides, 77% (43 cSv) of this dose is formed by the sources of natural gamma-irradiation, whereas 23% (13 cSv) - by technogenic concentrations of natural radionuclides (addition). Maximum individual effective dose by gamma-irradiation is a private house is 16.2 mSv/year.

The collective exposure dose formed as the result of inhalation penetration of thoron and daughter products is equal to 32 cSv/year. Besides, 81% of this dose (26 cSv) is caused by exhalation of thoron from the building materials and soil with natural concentration of ^{232}Th, and 19% (6.0 cSv) - by Th exhalation from building materials with elevated concentration of natural radionuclides (addition). Maximum individual effective dose in a private house constitutes 3.8 mSv/year. In the result of penetration of radon and daughter products the annual collective dose (59 cSv) is formed.

The structure of collective exposure dose H for population of village Kostousovo is shown in table 1.

Table 1.

The structure of collective exposure dose H for population of village Kostousovo

	H_{Tn}	H_{Rn}	H_{γ}	$H_{Tn}+H_{Rn}+H_{\gamma}$
Total, cSv/year	32	59	56	147
Excess, cSv/year	6,0	-	13	18
Excess, %	19	-	23	12

In the most contaminated private house there are living 7 persons. The structure of annual individual effective dose E is shown in table 2. This house was built in 1970. The accumulated for 25 years over-background individual exposure dose constituted almost 0.5 Sv.

Table 2.

The structure of annual individual exposure dose E for inhabitants of most contaminated house in village Kostousovo

	E_{Tn}	E_{Rn}	E_{γ}	$E_{Tn}+E_{Rn}+E_{\gamma}$
Total , cSv/year	5.3	1.4	16.2	22.9
Excess, cSv/year	4.7	-	15.2	19.9
Excess,%	89	-	94	87

CONCLUSION

Careless storage of radioactive staff at the railway station Kostousovo has led to its non-controlled use in building construction and thus to contamination of the territory by long-live natural radionuclides and daughter products. About 10200 m^2 of the territory and more than 10% of houses were subjected to contamination. As a result the external gamma-irradiation and indoor ^{222}Rn concentrations have increased.

Annual collective dose has increased by 19% through inhalation of thoron and daughter products, by 23% through external gamma-irradiation , an increase of 12% is registered in cumulative exposure dose. In case of extreme radioactive contamination of private house there was registered an 11-fold increase of the individual effective exposure dose.

Basing on the results of investigations some rehabilitation measures were proposed.

DOSE MEASUREMENT AND ITS PROBLEMS ON THE HIGH BACKGROUND RADIATION AREA(HBRA) IN CHINA

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INTRODUCTION

As a part of China-Japan cooperative research on radiation epidemiology, we carried out dose-assessment study in the HBRA since 1991 to 1994. Particularly, we have examined quantitatively on the characteristics of distribution of the numerical measurements and analyzed the variables concerned, using different kinds of dosemeters in order to obtain the information with regard to accuracy of the data for dose estimation. We selected for the dose measurement, 陽東県 and 陽西県 as the HBRA and 恩平県 as the CA.

INHOMOGENEOUS SPATIAL DISTRIBUTION OF RADIATION EXPOSURE IN THE HBRA

Average dose rates for indoor and outdoor in different hamlets of the HBRA measured by NaI (Tl) scintillation survey meter (TCS-166) are shown in **Figure 1.** It was remarkable that indoor dose rates were about 2 times higher than those of outdoor, although the ratios of the dose rates between indoor and outdoor in the CA were not so remarkable, rather almost similar. Further study showed that the dose rates of indoor had a good correlation with the dose rates of the house wall, and decreased with increase of the distance of the measured site from the house wall. These results strongly suggest that levels of radionuclides containing house wall will be deeply concerned with the radiation dose exposed. Thus, we have performed radionuclide analysis of the building materials such as bricks, tiles and soil samples using Ge-semiconductor diode detector. Nuclides of Th-232 decay products in the HBRA were 3-4 times higher than those of the CA. Similarly the decay products of Th-232 in the HBRA were about 2 times higher than those of U-238. These results were verified the data obtained by Chinese investigators. We have further measured personal doses with PDMs and TLDs for 24 hours and 2 months respectively. It was obvious that there were good correlation between personal radiation dose rates and indoor dose rates. Personal dose rates largely depends upon exposure dose rates originated from radionuclides in building materials of house.

VARIATION OF THE EXPOSED PERSONAL RADIATION DOSE RATES WITHIN A HAMLET

We have examined variation of the indoor dose rates of 32 different houses within a hamlet of 牛根 . We observed there were fairly large variation of the indoor dose rates with different houses in a same hamlet, presumably due to the different housing conditions such as levels of

radionuclides of the building materials and spatial location in a hamlet. It should be noted that this range of variation on the indoor dose rates was almost comparable with those of distribution of different hamlets in the HBRA as shown in **Figure 2.** Further we have examined variation of personal dose rates measured by 96 TLDs as well as PDMs in a hamlet of 麻地. Fairly wide range of variation for personal dose rates within a hamlet demonstrated in **Figure 3.** This wide range of distribution for personal dose rates is considered to be due to, at least, two kinds of independent factors;
1) different levels of indoor dose rates with houses and 2) different personal behavior which is reflected as an occupancy factor.

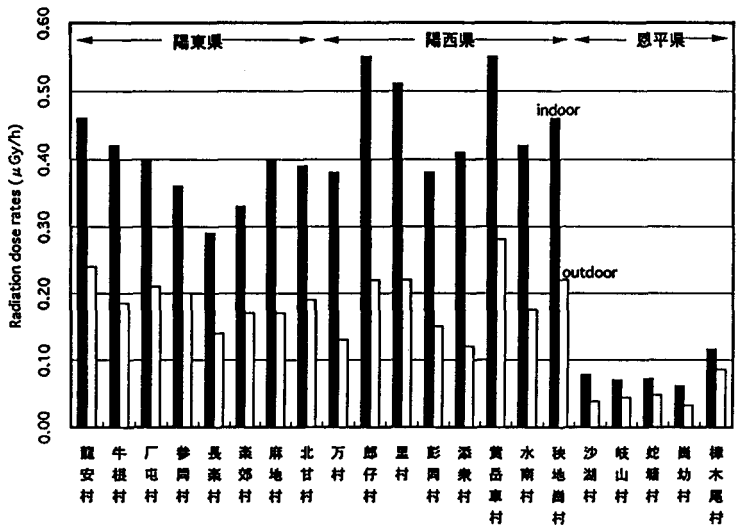

Figure 1. Variation of the mean environmental radiation dose rates on the HBRA (measured by TCS-166).

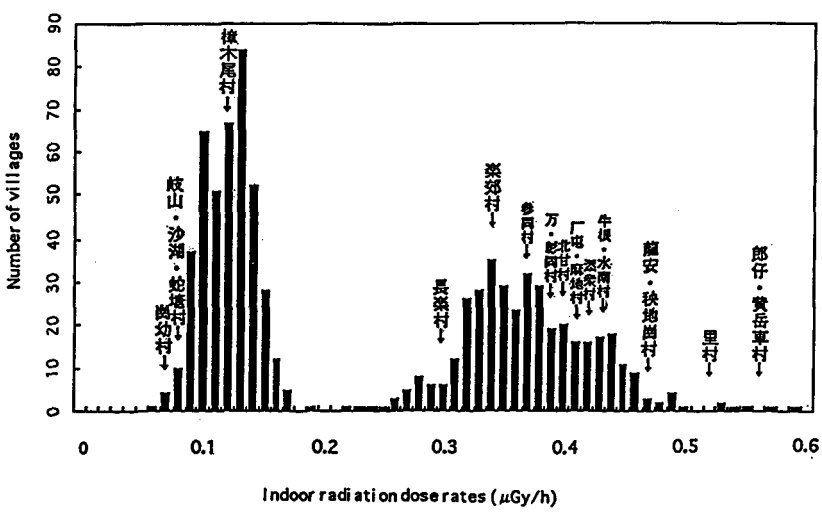

Figure 2. Frequency distribution of indoor radiation dose rate on HBRA (measured by survey meter).

VARIATION OF OCCUPANCY FACTORS CONCERNED AND ESTIMATION OF PERSONAL DOSE RATES

We have three models for personal dose measurement. They are two direct methods using electronic pocket dosimeter(PDM-101) and thermoluminescence dosimeter(UD-200S) and indirect method. The different approaches will be undertaken; direct methods and indirect method by using occupancy factor. In direct approach to estimate personal dose rates with hamlet population, since the difference between indoor dose rate and outdoor dose rate is so large, personal exposed dose rate depends upon their living life style which is related to occupancy factor i. e., ratio of the resident time on indoor per one day. This relation will be shown as following equation;

$$Dp=Di(q) + Do(1-q), \quad q=(Dp-Do)/(Di-Do)$$

where Dp, Di, Do and q are personal dose rate, indoor dose rate, outdoor dose rate and occupancy factor, respectively. Thus we can estimate personal dose rate if the values of Di, Do and q are given.

Coefficient q value of occupancy factor could be estimated by different two ways;
1) Indoor occupant time is available by direct measuring using the method of questionnaires and 2) Effective occupancy factor which is estimated from equation mentioned above and known values of Dp, Di and Do.

As it is impossible to measure the personal dose rate directly for everyone and them we will estimate indirectly using the measured environmental radiation dose rates. Further study to elucidate a nature of occupancy factor in various conditions is needed. And we should be carefully applied the occupancy factor for estimation of personal dose rates in hamlet population.

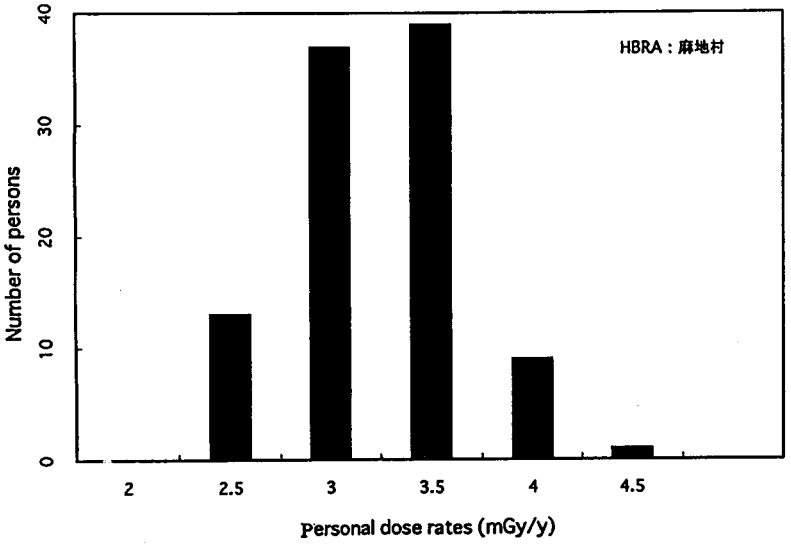

Figure 3. Frequency distribution of personal dose rates in a hamlet of the HBRA (measured by TLD).

A TERRITORY-WIDE SURVEY OF INDOOR GAMMA DOSE RATE AND ITS RELATION TO INDOOR RADON LEVEL IN HONG KONG

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ABSTRACT

A territory-wide survey of indoor radon level and indoor gamma dose rate was conducted in 1993. This paper reports the result of the indoor gamma dose rates and assesses the relationship between gamma dose rate and radon concentration in various types of buildings in Hong Kong. The TLDs and passive radon detectors were exposed simultaneously for about six months.

The overall arithmetic mean gamma dose rate is 0.22 $\mu Gy\ hr^{-1}$ with a standard deviation of 0.04 $\mu Gy\ hr^{-1}$. The modal gamma dose rate in high radon level rooms (>100 $Bq\ m^{-3}$) is 0.28 $\mu Gy\ hr^{-1}$ and in low radon level rooms (<100 $Bq\ m^{-3}$) is 0.22 $\mu Gy\ hr^{-1}$. The correlation between gamma dose rate and radon level is ventilation dependent and the regression coefficient ranges from 1.5×10^{-4} to 1.8×10^{-4} $\mu Gy\ hr^{-1}$ per $Bq\ m^{-3}$. Other factors that affect the indoor gamma dose rate and its relation to radon level are also discussed.

INTRODUCTION

Measurement of building materials and subsequent use of mathematical models to estimate the exhaled radon have shown that building materials, in particular the concrete, are the main contributors of indoor radon in high rise buildings (1). In 1993, a territory-wide survey of indoor radon level and gamma dose rate was jointly conducted with the Environmental Protection Department of the Hong Kong Government. Apart from knowing the population exposure due to radon and gamma irradiation, the survey also provides information on correlation between them.

METHODS

The whole territory of Hong Kong was divided into 19 districts and according to the land use and population in each district, a total of 1774 sites were measured. The sites were classified into residential and non-residential. Solid state track detectors consisting of CR39 chips were used to measure radon. Design, fabrication, calibration and readout of the radon detectors were all done in the University. On the other hand, the TLD badges for measuring gamma dose were borrowed from the China Institute of Radiation Protection in Taiyuan, China and they have to be returned to Taiyuan for reading out the TL signals. Each TLD badge contained five LiF chips and the badges were sent to Hong Kong by air and returned to Taiyuan by air again after the exposure. In order to determine the exposure of the badges during the freights, four background TLD badges were shipped with the batch. Immediately prior to deployment of the TLDs, the first background TLD was annealed at 240 °C and returned immediately to Taiyuan by air with the second un-annealed background TLD. The exposure of the whole batch of TLDs during the freight to Hong Kong and during the period before deployment was then given by the difference of the two background TL values. Similarly the exposure of the batch during the return freight was determined by annealing the third background TLD prior to be returned to Taiyuan. The fourth background TLD was kept inside a lead shield on arrival to Hong Kong. It was returned with the batch and its TL reading was used to compare with the known gamma dose inside the lead shield determined by a pressurized ionization chamber.

Each TLD badge was fixed to the bottom of a radon detector so that both could be hung from the ceiling of the sampling site. Because only 1000 TLDs were available, some radon detectors did not have associated TLD badges. Despite of this, the distribution of TLDs among the different types of buildings was similar to that of radon detectors. As a precaution and as a measure of the reliability of the detectors, some sites were given 2 detectors so that variation in sensitivity of the detectors could be determined. As a result, in addition to losses, the final number of sites that have simultaneous measurement results of radon level and gamma dose rate is only 664.

The cosmic ray contribution at each measurement site was not determined individually. But from a previous gamma dose rate measurement conducted in the sea, the cosmic ray contribution in Hong Kong was found to be 0.035 μGy hr^{-1} (2). The cosmic ray transmission factor in high-rise buildings similar to those in Hong Kong was found by Cui et al (3) to be 0.66 and 0.72 for floor 1 to 3 and floor 4 to 13 respectively. Using an average transmission factor of 0.69, the indoor gamma dose rate due to cosmic ray is about 0.024 μGy hr^{-1}. This value was considered as the background and was subtracted from all the gamma dose rates.

Over 80 science students from The University of Hong Kong and a few other part-time workers participated in the survey. They visited occupants of the buildings during the months January to February 1993, installed the radon detectors and TLDs and recorded particulars about the exposure environment. After about half a year, self addressed envelops were sent to the occupants asking them to return the detectors. The CR39 chips were etched in 6.25 N NaOH at 70 °C for 16 hours. Track density was counted manually using an optical microscope. The TLD badges were sent back to Taiyuan for readout of the gamma dose.

RESULTS

664 sites have measurement data for both gamma dose rate and radon concentration and they are plotted in Fig. 1. It is evident that the radon level increases with gamma dose rate. But because of the small fraction of radon generated inside the building material that is exhaled into the room and the uncertain gamma dose contribution from the cosmic ray, the relation can only be scarcely described. Similar analysis were made for sites having different ventilation conditions. It is found that the scattering of data points in the plots are similar and the corresponding slopes of the radon concentration weighted regression lines are 1.5 x 10^{-4}, 1.7 x 10^{-4} and 1.8 x 10^{-4} μGy hr^{-1} per Bq m^{-3} respectively for poor, good and intermediate ventilation. Poorly ventilated homes have higher radon concentrations for the same gamma dose rate and therefore they have a smaller value of slope.

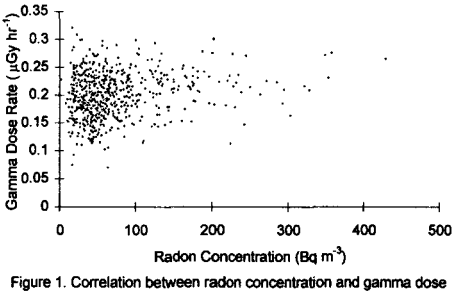

Figure 1. Correlation between radon concentration and gamma dose

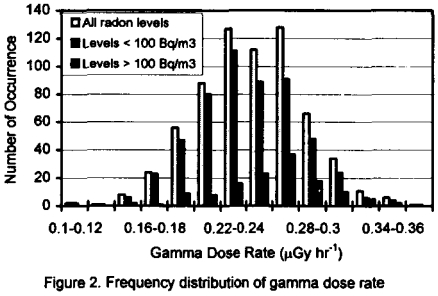

Figure 2. Frequency distribution of gamma dose rate

The frequency distribution of gamma dose rate is plotted in Fig. 2. Distributions of gamma dose rate for low radon concentration (<100 Bq m^{-3}) and high radon concentration (>100 Bq m^{-3}) were also plotted to illustrate the relation between radon concentrations and gamma dose rates. The arithmetic mean gamma dose rate for all measurement sites is 0.22 μGy hr^{-1} with a standard deviation of 0.04 μGy hr^{-1}. For low radon and high radon sites, the means and standard deviations of gamma dose rates are 0.22; 0.04 μGy hr^{-1} and 0.24; 0.04 μGy hr^{-1} respectively.

Attempts were made to find out the effects of other factors (apart from ventilation) that affect radon level but not the gamma dose rate. Wall covering materials and floor covering materials are the two most significant ones. But the floor only occupies about 1/6th of the total exhaling surface, therefore floor covering materials will not be considered. The two wall covering materials most commonly used in Hong Kong are emulsify paint and wall paper. A plot of gamma dose rates against radon levels for dwellings with low ventilation and painted walls is shown in Fig. 3. In the analysis that follows, only sites that have low ventilation were selected. Other ventilation were not considered because the ventilation effect will overshadow all the effects of the other factors. A similar plot for wall paper showed that the dependence of radon level on gamma dose rate is more

pronounced for painted walls than or wall paper covered walls. The greater deviation exhibited for wall paper is probably due to the fact that diffusion coefficient of wall papers for radon are more diverse than of emulsify paints.

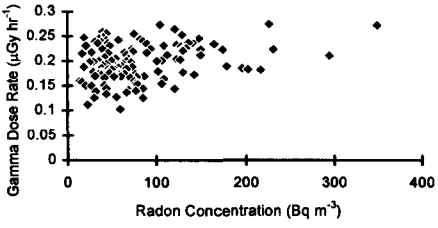

Figure 3. Correlation between gamma dose rate and radon concentration for rooms painted with emulsify paint and under closed window conditions.

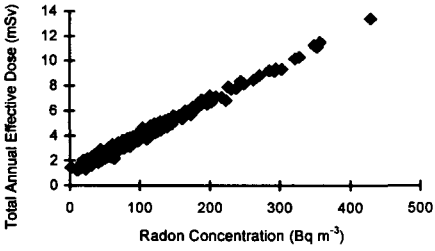

Figure 4. Total effective dose (radon + gamma) in houses of different radon concentrations.

For the case of emulsify paint, it is noted that the mean radon concentration in these sites increases quite rapidly with gamma dose rate. The arithmetic mean radon concentrations are 59, 67, 89 and 136 Bq m^{-3} respectively for gamma dose rate ranges in 0.1-0.15, 0.15-0.2, 0.2-0.25 and 0.25-0.3 μGy hr^{-1}. For wall paper covered walls, they are 84, 78, 131 and 236 Bq m^{-3} respectively for the same gamma dose rate ranges. The higher mean radon levels for wall paper covered walls are due to the fact that they are mainly found in offices and the majority of offices have central ventilation resulting in higher indoor radon levels. Of the buildings that have painted walls and low ventilation, only 15 % are offices. But of buildings that used wall paper, 56 % are offices.

The type of building material will certainly have effects on the radon-gamma relationship. But in Hong Kong, the majority of buildings are built of concrete. In our sampling sites, there are only about 3 % brick houses, 2.5 % fly ash concrete buildings and 0.2 % stone houses. Hence, their numbers are too small to allow any meaningful analysis of the radon-gamma relationship.

Variation of radon-gamma relationship at different floor levels was investigated. But it was found that there was neither a trend for indoor radon level nor gamma dose rate for different heights of a building even though walls in lower floors are usually much thicker than in upper floors. Hence plots of the radon-gamma data points at different floor levels look similar to Fig. 1.

The total effective dose at each site was computed by adding the radon dose and gamma dose at that site. Radon dose was obtained by using a dose conversion factor of 0.061 mSv/(Bq m^{-3}) given in ICRP Publication 50 (4). The equilibrium factor was taken as 0.43, 0.30 and 0.29 for poor, intermediate and good ventilation sites respectively. These values were obtained by a subsequent survey of equilibrium factors in 1994. The total effective dose against radon concentration is plotted in Fig. 4. Clearly the radon dose dominates over the gamma dose.

CONCLUSION

There are indications that indoor radon level increases with gamma dose rate which is directly related to the radionuclide contents in the building materials. But indoor radon levels are affected to a much greater extent by other factors such as ventilation and wall covering materials. When the total effective doses are concerned, radon level is by far the dominating contributor.

ACKNOWLEDGMENT-- The authors would like to thank the Environmental Protection Department of the Hong Kong Government for the support to the surveys.

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RADIOACTIVITY IN HOKUTOLITE FROM PEITO HOT SPRING, TAIWAN

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INTRODUCTION

Hokutolite is a radioactive rock discovered at Peito hot spring, Taiwan and is famous for high content of Ra and daughter nuclides (1,2). Hokutolite is found along the river stream flowing from the source of spring water and the main chemical composition is recognized to be the sulfate of lead and barium. Radioactive elements containing in the hot spring water are incorporated into lattice structures of hokutolite during its formation. The radioactive analysis of hokutolite is necessary to understand the relationship between the growing condition of hokutolite and radioactivity and also important for radiation protection.

EXPERIMENTAL

Sampling and sample treatment

Survey of hokutolite was carried out in October, 1993 with a NaI scintillation survey meter along the river flowing from the orifice of the spring water. Hokutolite was found at about 300 m downstream from the source and high radiation was distributed as spots on the surface of rocks exposed in the river. The high radiation spots were sampled using a burin together with the underlying rock that supports the hokutolite. The mineral specimen was cut perpendicular to the surface, along the growing direction of hokutolite, into pieces with about 2 cm in width.

Hot spring water was also collected at the source and brought back to a laboratory. After addition of Ba and Pb to the water, concentrated sulfuric acid was added to make precipitate of $BaSO_4$ and $PbSO_4$ and the precipitate was recovered by filtration.

Measurement of radioactivity with an imaging analyzer

A two-dimensional radioactive distribution of hokutolite along the growing direction was measured with an imaging analyzer, BAS 1000 (Fuji Photo Film Corp.). The cut surface including the underlying rock was contacted with an imaging plate placed on a thick lead block and the sample was covered in lead grains during exposure.

Measurement of gamma spectra of hokutolite

To examine radionuclide concentrations in hokutolite along the growing direction, hokutolite was scraped off successively from the surface to the inner part with a chisel and the activity was measured with a Ge detector (ORTEC). Activities of ^{226}Ra and ^{228}Ra were determined by ^{214}Bi (609 keV) and ^{228}Ac (911 keV).

Chemical composition analysis

Hokutolite of the whole was ground to a powder and subjected to X-ray diffraction analysis to determine a ratio between $BaSO_4$ and $PbSO_4$ from lattice constants. The powder was then dissolved in HI solution with heating at 210 °C for 3 h in a sealed decomposition vessel and concentrations of Ba and Pb were determined by ICP-AES and ^{232}Th by ICP-MS.

RESULTS AND DISCUSSION

An illustration of the cut surface of hokutolite along the growing direction is shown in

Fig. 1. The hokutolite had a whitish color and was clearly distinguished from the underlying rock. The thickness of growth was about 6 mm at the maximum. Three-dimensional distribution of the radioactivity is also shown in Fig. 1. A part of the imaging picture of the hokutolite was divided into 50 x 50 mesh screen (0.16-0.64 mm^2/mesh) and the intensity of the activity in each mesh was obtained as PSL (Photo-Stimulated Luminescence) per unit area. The resolution of the imaging analyzer is 0.2 mm and the PSL is proportional to the intensity of radioactivity. The mountain located in the center of the three-dimensional distribution is hokutolite and left side of the mountain is the surface of hokutolite contacted with river water and right side towards the underlying rock. An intrusion of hokutolite into cracks of the underlying rock was observed and the mountain showed several peaks. The imaging pictures show that non-radioactive fine particles are incorporated into hokutolite during its growth (3). The mole ratio of Ba/Pb = 1.9 was determined by X-ray diffraction analysis and 2.4 by ICP-AES, and are similar to values previously reported (4,5).

The activity ratios of $^{228}Ra/^{226}Ra$ in the hot spring water and the whole hokutolite were listed in Table 1. The hokutolite (A) is the same sample shown in Fig. 1 and was located out of the river water, while the hokutolite (B) in the water. Lower activity ratio in the whole hokutolite than the spring water is reasonable because the activity ratio decreases with time: the hokutolite near the underlying rock has lower activity ratio compared to the surface where the activity ratio would be the same as that of the spring water. The faster decay of ^{228}Ra than ^{226}Ra decreases the activity ratio in the whole hokutolite. An equilibrium $^{228}Ra/^{226}Ra$ activity ratio of 0.095 was expected for hokutolite that had been formed at a constant growth rate for a long time within the river water having $^{228}Ra/^{226}Ra$ activity ratio of 26.3. However, the observed activity ratio for the hokutolite (A) was significantly lower than the expected one and even for the hokutolite (B)

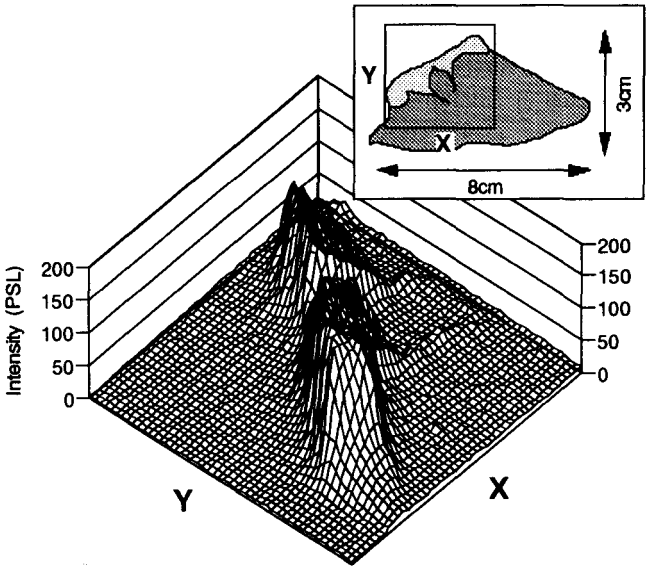

Figure 1. An illustration of the cut surface of hokutolite and its three-dimensional distribution of radioactivity measured with an imaging analyzer.

Table 1. Activity ratio in the spring water and hokutolite.

	Spring water	Hokutolite (A)	Hokutolite (B)
Activity Ratio $^{228}Ra/^{226}Ra$	26.3	0.027	0.178

that gives an unrealistic growing period within the river water, about 2,000 years. This would suggests that these hokutolite had been out of the river water for a certain period.

A change of $^{228}Ra/^{226}Ra$ activity ratio from the surface along the growing direction is shown in Fig. 2. The activity ratios were seemed to become constant at distance over 2.5 mm from the surface, suggesting an existence of ^{232}Th in the hokutolite and a radioactive equilibrium between ^{232}Th and ^{228}Ra. The average activity of ^{228}Ra from 2.5 to 5.5 mm is 1.52±0.34 Bq g^{-1} and the activity of ^{232}Th determined by ICP-MS was 1.55 Bq g^{-1}, supporting the radioactive equilibrium between ^{232}Th and ^{228}Ra over 2.5 mm. The decline in activity ratio from the surface to 2.5 mm could be attributed to decay of ^{228}Ra incorporated into hokutolite and the activity ratio at the surface would suggest an elapsed time after the hokutolite had isolated from the river water. A growth rate of the hokutolite was simulated and 0.17 mm y^{-1} gives a best fit as shown in Fig. 2. The elapsed time for the hokutolite after isolation was calculated to be about 60 years. The activities of ^{226}Ra and ^{228}Ra in hokutolite are summarized in Table 2. Higher activity ratio than the present work was reported for thin hokutolite deposited on the surface of sands and pebbles (1), indicating the older age of the present hokutolite. The concentration of ^{226}Ra in hokutolite is a few orders of magnitude higher than that in usual rocks.

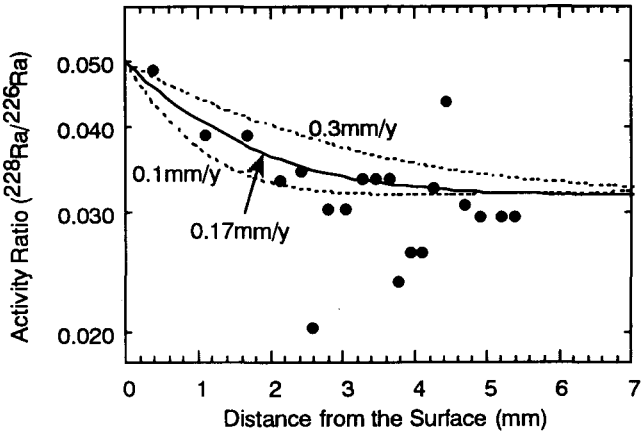

Figure 2. Activity ratios of ^{228}Ra to ^{226}Ra in hokutolite along the growing direction and results on simulation of growth rate.

Table 2. Activities* of radium isotopes and activity ratio in hokutolite.

^{228}Ra	^{226}Ra	$^{228}Ra/^{226}Ra$	Ref.
0.10	0.052	1.92	sands & pebbles (1)
0.48	0.059	8.14	sands & pebbles (1)
9.25	36.26	0.255	crystal (2)
3.46	71.3	0.0485	Present Work**

*Bq g^{-1}, ** The surface of the hokutolite

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IRPA9
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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 1.1. or 1.2.

ABSTRACT (See instructions overleaf)

This paper reports the results of environmental radiation background level monitored in Jilin Province from 1983 to 1993. The activities of U, Th, ^{226}Ra, ^{40}K are 2.3 , 0.1 , 1.1 , 0.3 ($\times 10^{-2}$ bq L^{-1}), in food they are 3.7 , 2.4 , 14, 7, 81.5 ($10^{-2} \cdot Kg^{-1}$) respectively. Rn activities indoor and outdoor are 8.7 , 5.8 (Bq m^{-3}) respectviely. Population weighted average of air-intaking dose of ionization component of universe ray is 3.2×10^{-8} Gy · h^{-1}. Air-intaking dose of γ radiation of the ground surface indoor and outdoor are 7.7 , 9.8 (10^{-8} Gy · h^{-1}) γ respectively. 1.47 msv (H) is obtained for natural background.

RADIATION EXPOSURE OF THE POPULATION IN THE VICINITY OF COAL-FIRED POWER PLANTS

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ABSTRACT

Our investigation included 4 coal-fired power plants (CPP). All 4 CPPs are located near inhabited places and the depots of ash and slag are 2-5 km away from the CPPs. The measuring of the exposition dose rate was performed using the highly sensitive ionizing chamber under high pressure "in situ" 1 m above ground and termoluminescent (TLD) dosimeters on 4 measuring points (2-5 km in front of the CPP, on the ground of the CPP, at the depots of ash and slag and 2-5 km behind the CPP). The average annual radiation dose on 5 locations in the surroundings of the CPPs is approximately 1 mSv.

INTRODUCTION

It is known that all types of coal contain at least 1-2 ppm uranium and 3-4 ppm thorium which correspond to 12.2-24.4 Bq/kg uranium-238 and 12.21-16.8 Bq/kg thorium-232 as well as the products formed by their radioactive decay (1, 2). These concentrations are somewhat less in comparison to the content of natural radionuclides in the earth's crust (3). However, some types of coal contain considerably higher amounts of natural radionuclides (hard coal). Besides this, combustion of coal in CPPs leads to the concentration of natural radionuclides, during which the enrichment factor for ash and slag can amount from 5 to 10 (4-6). It was therefore interesting to examine the contribution and influence of CPP on the environment and total irradiation dose of the population (7-9).

MATERIAL AND METHODS

Exposition dose rates of gamma radiation were measured "in situ" at 1 m above the plain surface of ground. Measurements were made at four locations (2-5 km in front of the CPPs along the most frequent wind direction, on the grounds of CPPs, at the depots of ash and slag and 2-5 km behind the CPPs along the most frequent wind direction).

Three instruments were used for measuring exposition doses of gamma radiation:
- highly sensitive, pressurized ionization chamber (0.35 MPa argone + 1.4 MPa nitrogen), with low detection limit of 0.258 nC/kgh and resolution of 0.00258 nC/kg, manufactured by "SILENA",
- termoluminescent dosimeters (TLD) from the Institute for Nuclear Science, Vinca (each box contained 3 TLD tablets).

RESULTS AND DISCUSSION

A comparison of gamma radiation exposure dose rates at four locations for all the CPPs examined is given in Table 1 and 2. Table 1 shows that the highest values for CPP 1 were detected on depots of ash and slag (11.44 E-4 nC/kgs), while the other three locations had almost identical values. Exposition doses of gamma radiation in CPP 2 showed approximately equal values in all locations, while the highest exposition dose was measured 3-5 km behind the CPP and amounted to 8.60 E-4 nC/kgs. Approximately equal values were detected in all locations measured in CPP 3. The exposition dose rates for CPP 4 ranged from 7.17 E-4 (depot) to 9.13 E-4 nC/kgs (3-5 km behind the CPP).

Table 1. Radiation doses measured by anionizing chamber in the vicinity of coal-fired power plants

Locality	Dose (nC/kg s)			
	1.	2.	3.	4.
3-5 km in front of CPP*	9.55E-4	7.17E-4	7.62E-4	7.70E-4
On the grounds of CPP	9.14E-4	7.17E-4	7.71E-4	9.10E-4
3-5 km behind CPP*	9.13E-4	8.60E-4	7.54E-4	9.13E-4
Depot of ash and slag	11.44E-4	7.51E-4	7.65E-4	7.17E-4

*along the most frequent wind direction

Table 2. Radiation doses measured with TL-dosymeters in the vicinity of coal-fired power plants

Locality	Measuring period	Dose (mSv/3 month)			
		1.	2.	3.	4.
3-5 km in front of CPP*	jan.-march	-	0.31	0.31	0.33
	apr.-june	0.16	0.16	0.17	0.19
	july-sept.	0.46	0.22	0.24	0.23
	okt.-dec.	0.23	0.20	0.24	0.17
	Annual dose:	-	0.89	0.96	0.94
On the grounds of CPP	jan.-march	0.30	0.33	0.42	0.29
	apr.-june	0.17	0.25	0.19	0.16
	july-sept.	0.22	-	0.32	0.27
	okt.-dec.	0.21	0.19	0.22	0.25
	Annual dose:	0.90	-	1.15	0.97
Close to electrofilter	jan.-march	0.29	0.27	0.25	0.20
	apr.-june	0.17	0.17	0.17	0.18
	july-sept.	0.21	0.19	0.18	-
	okt.-dec.	0.21	0.22	0.21	0.27
	Annual dose:	0.88	0.85	0.81	-
On the depot	jan.-march	0.30	0.29	0.27	0.34
	apr.-june	-	0.22	0.19	0.18
	july-sept.	-	0.20	0.19	0.29
	okt.-dec.	0.18	0.19	0.20	0.22
	Annual dose:	-	0.90	0.85	1.03

Table 2. Radiation doses measured with TL-dosymeters in the vicinity of coal-fired power plants (continued)

Locality	Measuring period	Dose (mSv/3 month)			
		1.	2.	3.	4.
3-5 km behind CPP*	jan.-mart	0.29	0.32	0.30	0.32
	apr.-juni	0.16	0.19	0.19	0.16
	juli-sept.	0.24	0.24	0.23	0.25
	okt.-dec.	0.24	0.21	0.24	0.24
	Annual dose:	0.93	0.96	0.96	0.97

* along the most frequent wind direction

An analysis of results indicates that the highest irradiation doses for all CPPs were measured during the period from January to March, while the lowest from March to June. It was obvious that the doses on depots of ash and slag as well as on the grounds and behind CPPs, in the direction of winds, were higher than at other locations. The total annual doses were highest at locations 3-5 km behind CPPs (except for CPP 3, where the highest annual dose was detected on the grounds of the CPP). The relatively lower doses detected close to the electrofilters are probably due to the absorption of radiation from metal constructions located between the dosimeters and the electrofilter contents, because it was not technically possible to set the dosimeters into the electrofilters.

Differences in values of doses detected between the spring-summer and autumn-winter seasons are most probably a consequence of a difference in the intensity of production of CPPs and repair work usually carried out in the spring-summer period. Table 2 gives a comparison of the exposition dose rates of gamma radiation at four locations measured for all the CPPs examined.

CONCLUSION

The basic problem of technologically induced increase of natural radioactivity caused by CPPs is the increase of the basic radiation rate in the surrounding area and higher exposure of the local population to radiation.

For coal with elevated or high uranium contents, a question is raised whether such depots should be considered as uranium strip mines or not. The potential for the discovery and application of coal with higher natural radioactivity is increasing as well as the need for coal. Therefore, it is necessary to devote much more attention to environmental pollution with radioactive material from coal-fired power plants in both research and scientific work as well as to legally regulate the technologically induced increase of natural radioactivity.

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NEW DATA ON NATURAL RADIOACTIVITY OF SOME ROMANIAN FRESH WATER SOURCES

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INTRODUCTION

As water is an important factor of transfer of the natural radioelements towards the human organism, the main objective of the present research is to measure the natural radioactive content of fresh water in the Moldova region. The natural radioelements of utmost interest in finding out the radioactive water content are uranium,thorium,radium-226 and potassium-40.

MATERIALS AND METHODS

During 1982-1989 there were taken, from the territory of Moldova, 767 water samples of which 348 originated from ground water sources (individual or catchments) and 419 from surface waters (1). Between 1990 and 1994 we went on with the measure of the natural radioactive content of the fresh water in Moldova region (740 samples). We wanted to have data on the possibly changing values as a result of the new water sources findings and of the modifications in water processing stations. In selecting the sampling points, i.e. wells and micropower stations, the direction of a ground water table was followed, so that the differences in value should be due only to dissolved radioelements. As far as the potable water originating in the surface sources is concerned, the samples were taken from the central municipal and town water supply systems. The taking of samples and their analysis were carried out in conformity with the current standards (1).

RESULTS AND DISCUSSION

The variation domains with water supply sources originating from surface layers and the ground water are presented in Table 1.

Table 1. Activity concentration of natural radionuclides in drinking water samples (mBq l^{-1})

Radionuclide	Water source		AMAC *
	Ground water	Surface water	
^{238}U	0.6 - 37.2	0.36 - 18.0	257
^{232}Th	0.05- 9.3	0.035- 8.0	40
^{226}Ra	1.1 - 21.0	0.75 - 9.5	88
^{40}K	15.0 -720.0	58.0 -670.0	13420

* AMAC - Admitted Maximum Activity Concentration STAS 1342-91

On should remark that neither of the analysed samples exceeded the specific activity admitted in Romania for fresh water, in any of the natural radiolements under investigation.The mean activity concentrations of natural radionuclides in fresh water samples, for the three time periods of survey,are shown in Table 2.

Table 2. Average activity concentrations (mBq l^{-1}) of natural radioactivity in drinking water samples (according to water source type)

Number of samples by water source and investigation period		Radionuclide			
Ground	Surface	^{238}U	^{232}Th	^{226}Ra	^{40}K
(1982-1985)					
I* 39	2	10.4±1.8	7.8±1.3	10.4±1.9	
II 102	155	3.9±0.8	1.3±0.2	4.8±1.2	140.4±11.1
III 17	44	<2.2	<0.22	<1.8	81.1± 7.4
(1986-1989)					
I 47	8	12.6±3.8	7.2±1.2	8.1±1.0	
II 109	157	4.0±1.4	1.5±0.3	3.9±1.4	283.2±120
III 34	53	0.6±0.2	0.06±0.02	1.2±0.3	112.2± 64
(1990-1994)					
I 66	15	11.7±3.1	7.5±1.2	8.9±1.3	
II 135	262	3.9±1.1	1.4±0.2	4.3±1.3	140.5± 22
III 119	143	0.7±0.2	0.1±0.04	0.8±0.2	52.9± 10

* - Water category

We succeeded in evidencing three categories of fresh water, according to the average content of natural radioelements, such as in Table 2.
1st category-fresh water with a high natural radioactive content, with a tenth order mean.
2nd category-fresh water with an average natural radioactive content, with unitary order mean.
3rd category-fresh water with a reduced radioactive content,with a subunitary mean.
The highest values for U, Th and Ra were mostly found in fresh water originating from ground table,about 23 % of samples being included in the 1st category. The fresh water originating from surface layers has a dimineshed natural radioelements content, due to treatment operations (floculation,settling,filtering, etc), 71% of the samples belonging to the category of average natural radioelements content and 29% to the 3rd category with the lowest natural radioactive content. During 1990-1994, the mean values for the four radionuclides were not by far different from those of 1982-1989, but the samples percentage distribution in the 2nd and 3rd categories is different. This may be explained by the rock radioactivity that the ground water passes through (2,3). In the case of the surface waters, it is possible that it might be due to the water processing improvement. The thorium concentration of drinking water is smaller than that in uranium, probably because of the much lower solubility of thorium in water. For each of the three categories the ingestion of natural radioelements per capita was assessed, considering a daily consumption of 2 litres (Table 3). We have obtained values which ranged within those reported by literature for the areas with

normal radiation background but higher than the average (4,5).

Table 3. The annual ingestion of natural radionuclides (Bq y^{-1})

Radionuclide	Drinking water <10 year old	Drinking water adults	Dietary intake in Moldova	Dietary intake in areas of "normal" background
^{238}U average	0.8	2.04	7.5	5
range	0.16-3.46	0.42-8.82	6.2-11.2	-
^{232}Th average	0.29	0.73	2.2	2
range	0.01-1.98	0.04-5.04	1.6- 3.4	-
^{228}Ra average	0.83	2.12	14.1	15
range	0.33-2.22	0.84-5.70	12.4-16.2	-

The contribution of drinking water to the total intake by ingestion varies over a wide range. Comparing the assessment values with those for food ingestion one can notice that natural radioactivity intake through water is much smaller (6).

CONCLUSIONS

1. The concentrations of uranium, thorium, radium and potassium activity in potable water in Moldova are lower than the admitted specific activity for the areas with a "normal" background.

2. Fresh water originating from ground water table has a higher natural radioelements content than that originating from surface water.

3. During 1990-1994 period the mean values for the four radionuclides were not different from the other periods, but water processing improvement and the finding of newer sources, probably influence the samples percentage distribution in those three categories.

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RADIOACTIVE POLLUTING POTENTIAL OF COAL-FIRED POWER PLANTS FROM ROMANIA

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ABSTRACT

Between 1993-1995,26 coal-fired power plants(CFP) were investigated.Water,coal,filter fly-ash,disposed ash and bottom-ash(slag) samples were collected.All samples were analysed for their natural radioactive element content (^{238}U and ^{232}Th series,and K-40) and at the sampling points the gamma dose rates were measured.

The escaping fly-ash, produced ash(fly-ash and slag) and the coal consumption per GWa were assessed as the associated activities of radionuclides.The mass radon flux density for coal,filter fly-ash,disposed ash,and slag samples were measured on lab specimens as well as the surface radon flux density for same building materials with ash content and were assessed the effective anual doses due to living is houses constructed with such materials.

INTRODUCTION

Coal is burned in furnaces operating at up to 1,800^{0}C in order to produce thermal and electrical energy,requiring $3 \cdot 10^9$ kg of coal to produce 1GWa of electrical energy (1).In the UNSCEAR 1982 Report (2) it is estimated the average concentration of ^{40}K,^{238}U and ^{232}Th in coal to be 50,20,20 Bq kg^{-1}, respectively.In a more recent paper from China (3) these concentrations are appreciably higher:104,36 and 30 Bq kg^{-1} for the same radionuclides.

Owing mainly to the elimination of the organic content of the coal,there is about an order of magnitude enhancement of the concentrations from coal to ash.So,the natural radionuclide concentrations in ash and slag from CFPs are signifinatly higher than the corresponding concentrations in earth's crust.Arithmetic averages in escaping fly-ash are 265 Bq kg^{-1},200 Bq kg^{-1},240 Bq kg^{-1} and 70 Bq kg^{-1} for ^{40}K,^{238}U,^{226}Ra and ^{232}Th,respectively (2).

METHODOLOGY

For estimation of radioactive pollutig potential of CFPs water(raw,industrial,recirculating),coal,filter (collected) fly-ash,slag and disposed ash samples were collected.The comsumption,produced ash and escaping fly-ash per GWa were assessed as well as the associated activities of radionuclides.

All samples were analysed for their radionuclide content using radiochemical or gamma spectrometric methods,respectively thermal neutron activation associated with gamma ray spectrometry for ^{238}U.

At the sampling points the gamma dose rates were measured using portable gamma rate meters having a sensivity of 0.01μSvh^{-1}(1μSv=100μR).

RESULTS AND DISSCUTION

All waters used in CFPs had U nat and ^{226}Ra concentations under the limits for drinking water(0.021 mg Unat/dm^3,0,088 Bq ^{226}Ra/dm^3(4),by themselves being not a radioactive pollution source.

The arithmetic averages of radionuclide contents of the 43 coal samples(bitumonous coal,lignite,brown coal) were at the level of reported averages in coal,being 36.3 Bq kg^{-1} for ^{238}U(range:6-112),44.4 Bq kg^{-1} for ^{226}Ra (range:2-112),19.7 Bq kg^{-1} for ^{232}Th(range:1-44) and 267 Bq kg^{-1} for ^{40}K(range:30-500).The mean ash content of coals varied between 20 and 62% and the caloric power between 1,200 and 5,500 Kcal kg^{-1}.

The activity concentrations of natural radionuclides of residual materials from CFPs are shown in Table 1.

It can be seen, than the highest activity concentrations were found in collected fly-ask and these value were taken into account for the computation of radioactive pollution potential by escaping fly-ash, although it is this know that the activity concentrations in this fraction are higher (2).

Using laboratory experiments(5) the mass radon flux density (Em) of coal, collected fly-ash, slag and disposed ash samples was measured.The results of these experiments are shown in Table 2.

The highest values for Em were measured for coal samples and the lowest ones for slag samples.

Also, were estimated the annual coal consumption, produced ash and atmospheric discharge by escaping fly-ash per GWa of electic energy generated by 19 CFPs which have had produced electric energy.The results of these estimates are shown in Table 3.

Activity concentrations of natural radionuclides in residual materals from CFPs

Table 1

ASH (no of sample)		Activity concentration ($Bq\ kg^{-1}$) ^{238}U	^{226}Ra	^{232}Th	^{40}K
Collected fly-ash	Average	100.0	129.3	49.6	421
(27)	Range	25-223	50-221	3-86	96-885
Disposed ash	Average	88.6	107.7	41.9	403
(27)	Range	17-230	21-227	5-86	83-1,097
Slag	Average	68.5	79.0	33.7	317
(20)	Range	6-161	15-159	3-84	71-852

Mass radon flux density (Em) vs. nature of sample

Table 2.

Nature of sample	No. of samples	Em (10^{-3} Bq $g^{-1}s^{-1}$) Average	Range
Coal	42	3.45	0.56 - 12.13
Collected fly-ash	26	1.59	0.11 - 4.70
Disposed ash	27	1.18	0.43 - 2.59
Slag	20	0.26	0.03 - 0.82

Estimates of annual coal consumption,ash production and atmospheric discharge per unit electrical energy generated

Table 3.

CFPs	Total installed power		Coal consumption 10^9 kg(GWa)$^{-1}$		Normalized Ash production 10^9 kg(GWa)$^{-1}$	Escaping fly-ash 10^6 kg(GWa)$^{-1}$	
19	GW						
Modern[a]	8.64	Range	4.5 - 20.4	(6)	1.56 - 8.16	4.5 - 304	(6)
		Average	11.73	3	4.33	64.2	7

a) Equipped with electrofilters

Using the values of activity concentrations of collected fly-ashes and the amounts of atmospheric discharges were assessed the activities of normalized discharges.The results of these estimates are shown in Table 4.

Estimates of the activities of annual atmospheric discharges per unit energy generated (GBq/GWa)

Table 4.

CFPs	Fly-ash removal efficiency %		^{238}U	^{226}Ra	^{232}Th	^{40}K
Modern	95 - 99.8	Average	8.4	10.0	3.65	32.2
(19)		Range	0.3-33.1	0.5-36.4	0.14-19.1	1.3-127.1

The total amount of disposed ash from ash dumps(from 26 CFPs) was estimated as being $230 \cdot 10^6$ t, and the total activity of natural radionuclides from these ones as being 21 TBq for ^{238}U, 29TBq for ^{226}Ra, 11TBq for ^{232}Th and 83 Tbq for ^{40}K and the activity of radon gas emanated in a year as being 93 TBq.

The gamma background in coal-fired power stations ranged between 0.04 and 0.12 $\mu Sv\ h^{-1}$, sometimes being slighty enhanced in comparison to the natural background of the areas.On the ash dumps the gamma dose rates ranged between 0.07 and 0.25 $\mu Sv\ h^{-1}$ being 2-3 times higher than the gamma background of the vicinities.

Two building materials with ash content in comparison with a normal one were investigated.The activity concentrations of natural radionuclides of these materials are shown in Table 5.

Activity concentrations of natural radionuclides in some building materials

Table 5.

Building material (ash content %)	^{226}Ra Bq kg^{-1}	^{232}Th Bq kg^{-1}	^{40}K Bq kg^{-1}	Density kg m^{-3}
Building material 1 (70)	123	52	141	660
Building material 2 (38)	40	13	52	1,780
Building matreial 3 (0)	27	11	242	660

Using laboratory specimens the surface radon flux density (E) was measured and were calculated the external, internal and total effective doses for a reference room build up from each material used 100%, the ventilation rate 1, residence time 4380h/y, equilibrum coefficient 0.4, conversion factor of equivalent equilibrum radon concentration in effective dose (mSv h^{-1} per Bq m^{-3}) of 9nSv and Ackers' conversion factors (6) .The results of these measurements and computation are shown in Table 6.

Annual effective dose due to bulding materials

Table 6.

Building material (ash content %)	E Bq m^{-2} h^{-1}	Annual effective dose (mSv) External	Internal	Total
Building material 1 (70)	1.14	0.397	0.024	0.421
Building material 2 (38)	0.62	0.119	0.013	0.132
Building material 3 (0)	0.80	0.118	0.017	0.135

It can be seen from Table 6 that the major contribution to the total effective dose is given by the external effective dose.At the same time, it is evident that in certain conditions building materials can be used without an additional risk for the public.

CONCLUSIONS

The main conclusions that resulted from this research are:

Coal-fired power plants are a potential radioactive polluting source for the environment due to atmospheric discharges of escaping fly-ash as well as by the presence of ash dumps, the activity concentrations of natural radionuclides in ashes being higher than in earth's crust.

Building materials with coal ash content can be used, in certain conditions, without additional risk for the public.

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WORKING LEVELS IN THE COAL FIRED POWER PLANT IN CROATIA

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INTRODUCTION

The exposure from man-made natural sources is called *technologicaly enhanced natural radiation* - TENR (1). One of the first sources of uranium and thorium which was detected not being connected with the nuclear industry, was found during energy production using fossil fuels. As the combustion of coal increases, so will the magnitude of environmental and human health hazards associated with trace elements and radionuclides mobilized by the coal fuel cycle. The large fraction of coal ash that does not find commercial application is usually dumped in the vicinity of the coal fired power plant (CFPP). The coal ash may contain enhanced levels of the natural radionuclides in the uranium and thorium series, especially fly ash. Among the decay products are the radon isotopes, ^{222}Rn, ^{220}Rn and ^{219}Rn, which are noble gases and thereby pose special problems in assessing the radiological hazard of fly ash. For that reason, investigations of the hazards were undertaken in the CFPP in Croatia, because the anthracite coal used for combustion has an average 10% sulphur and a variation of uranium.

EXPERIMENTAL PROCEDURES

In the seventies the uranium content in coal was between 500-1200 $Bqkg^{-1}$ (2). After 1980 it declined to an average 250 $Bqkg^{-1}$ due to opening of an different vein in the coal mine. This requested a thorough monitoring programme which included measurements of activity concentration of radionuclides in coal and ash samples, and measurements of working levels (WL).

WL is defined as any combination of short-lived radon daughters in one liter of air that will result in the emission of 1.3×10^5 MeV of potential alpha energy. Under conditions of secular equilibrium 3.7 $kBqm^{-3}$ (100 $pCiL^{-1}$) of ^{222}Rn produces 1 WL (3).

WLs were measured as "grab samples" using method developed by Scott (4).

RESULTS AND DISCUSSION

In the CFPP seven locations have been chosen, because of long-time occupational exposure, and five on-site in places with natural air flow. All WL values are an arithmetic mean of three measurements.

Tables 1 and 2 summarize the estimated WL values, together with occupancy time limit.

Table 1. WL OF OCCUPATIONALLY EXPOSED PERSONS INSIDE THE CFPP

Work place	mWL (1977)	Occupancy time limit	mWL (1983)	Occupancy time limit
Conveyour belt (coal)	8.0	42 h/week* unlimited	7.0	42 h/week unlimited
Conveyour belt (coal)	15.0	24-42 h/week	6.0	42 h/week unlimited
Below the automatic control (ash hooper)	80.0	21 h/week	12.0	24-42 h/week
Below the automatic control (ash hooper)	60.0	42 h/week	12.0	24-42 h/week
Waste pile fresh	80.0	21 h/week	-	-
Waste pile old	-	-	60.0	42 h/week
Bottom ash	80.0	21 h/week	20.0	24-42 h/week

* 42 h/week was taken as the occupancy time limit to comply with the US general population standards, since the workers in the CFPP were never considered as people occupationally exposed to radiation.

Table 2. WL ON-SITE IN PLACES WITH NATURAL AIR FLOW

Work place	mWL (1977)	Occupancy time limit	mWL (1983)	Occupancy time limit
Area around the steam generator building	6.0	unlimited	6.0	unlimited
Under the stack	5.0	unlimited	6.0	unlimited
Near the furnice	5.0	unlimited	6.0	unlimited
Office building (500 m from the CFPP)	3.0	unlimited	-	-
10 km fom the CFPP	3.0	unlimited	6.0	unlimited

The WLs have shown great variations between two measurements, depending on the radioactivity of the coal and combustion products present at the time of the measurements in the CFPP. The highest WL was besides the bottom ash and fresh waste pile, where even an occupancy time limit should be considered.

Table 3 summarizes the estimated WL values measured in 1990.

Table 3. WL MEASURED ON-SITE AND OFF-SITE CFPP IN 1990.

Location	mWL
Coal storehouse	6.0
Below the automatic control (ash hooper)	11.0
Area around the steam generator building	6.0
Slag and ash pile	6.0
Štrmac	6.0
Vozilići	5.0
Stepčići	5.0
Luka Plomin	4.0
Rabac	3.0

As we expected, the highest values were obtained on-site of the CFPP.

Locations 5 - 9 were at different directions and distances from the CFPP, chosen in dependency on wind-rose (Table 4).

Table 4. ALTITUDES, DISTANCES AND DIRECTIONS FROM THE CFPP

Location	Altitude (m)	Distance (km)	Direction
Štrmac	120	3	SW
Vozilići	100	5	NW
Stepčići	80	2	W
Luka Plomin	10	1	SE
Rabac	0	20	S

The most interesting case is the location Štrmac, where a hamlet was built on a ninety years old tailing site, where already the third generation of same families are dweling in the same houses. At the location Rabac, which is at the sea shore, the WL is slightly lower, due to the lower ^{226}Ra content of the sea.

CONCLUSION

This paper introduces WL measurements in the industry where TENR is present. The CFPP is a specific case with the apperance of natural radioactivity which was very similar to open pit uranium mining. The appearence of places with an increase of natural radioactivity in non-nuclear industries have left the legislator, at present without a ready solution in Croatia, how to systematize occupationally exposed workers, especially after the Chernobyl accident, when the public become sensitive to radiation of any origin.

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A FAST AND VERY SENSITIVE LSC PROCEDURE TO DETERMINE PB-210 IN ENVIRONMENTAL SAMPLES

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Introduction

Pb-210, which has a half-life of 22,3 [1] years, as a member of the Uranium-Radium decay chain causes a significant contribution to the internal radiation dose and belongs with other members of this chain to the natural occurring radionuclides which show a very high dose factor for incorporation. But unfortunately this nuclide cannot be measured sensitive and fast enough in natural samples because of its low γ- (E = 46,539 keV [2], also low transformation yield of about 4 % [2]!) and β - energies (max. energies 17 keV and 61 keV [3] respectively).
This paper describes a method with both aims: lowest detection limit and short waiting time for the results. To reach this, after a radiochemical cleaning Pb-210 is detected in a liquid scintillation counter by its beta decay.

Methods

After drying the sample material at 105° C plant samples were reduced to small pieces and ashed at 450° C. To remove the remaining carbon fractions HNO_3 must be added to the ashes as often as necessary.
Soil samples must be sifted so that only fractions smaller than 2 mm remain. For the further procedure 40 ml HNO_3 and 10 ml HF were added to samples up to 5 g ash mass and evaporated to dryness. This step must be repeated two times. The remaining material must be treated with 2 ml $HClO_4$ for three times and evaporated to dryness.
From now on plant and soil samples are treated uniformly. The next step in sample preparation is a radiochemical cleaning. Thereby the ash is dissolved under warming in 100 ml HBr and extracted with 75 ml Trioctylamine/Toluene. The organic phase must be washed with 0,1 M HBr. For the reextraction of the lead 50 ml HCl must be added and the watery phase is mixed with 50 ml $CHCl_3$. The organic phase is discarded. After adding 50 ml HNO_3 the solution is evaporated to dryness. At this point lead is completely separated from all other elements except bismuth. Therefore radiochemical cleaning must be continued [4].
For that purpose the evaporate is dissolved in 0,02 M HCl and put on an anion exchange resin (DOWEX 1X8) column (3,5 x 0,8 cm) with a velocity of 1 ml/min. The anion exchange resin must be in the chloride form. This method elutes lead, the bismuth fraction remains on the column. The eluted fluid is evaporated to dryness taken up in 10 ml of 0,1 M HNO_3 and added to 10 ml of a scintillation cocktail (Quickszint 400 from Zinsser).
The chemical yield for lead was determined for each sample by atomic absorption spectrometry (AAS).
To detect Pb-210 by its low energy beta radiation, we used a liquid scintillation counter Quantulus 1220 from the Finnish company LKB-Wallac with an active shield.

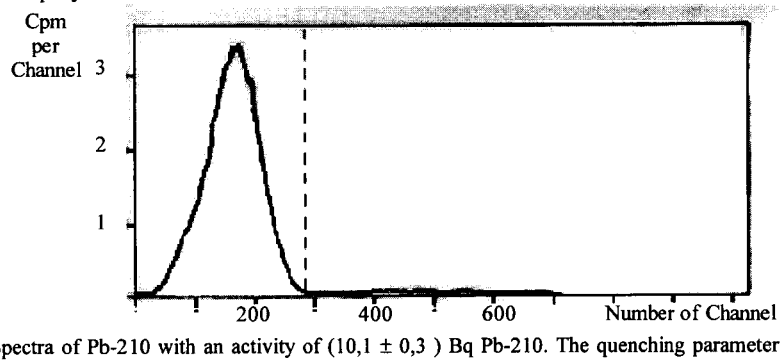

Figure 1: Spectra of Pb-210 with an activity of (10,1 ± 0,3) Bq Pb-210. The quenching parameter SQP(E) is 653. The counting window of Pb-210, which was fixed later is also drawn.

The time between separation of Pb-210 and start of the measurement was 3 h 55 min. In this period there is a build-up of Bi-210 and therefore the counts from channel 300 to 600 only come from this nuclide. Shorter waiting times are not desirable because of the effect of the chemoluminescene.

Fixation of the counting window for Pb-210 and the effect of quenching

In figure 1 the window for Pb-210 in the spectrum is marked. It reaches from channel 1 to an upper limit which is dependent to the effect of quenching. Quenching occurs when the sample is coloured, because the transmission of light pulses of the scintillator molecules to the photo cathode is weakened in the scintillator sample cocktail by absorption and so the spectra is shifted to lower energies. The quenching parameter SQP(E) which is detected by the LSC-Quantulus itself by a formula is a parameter for that effect. To get the dependence of the counting window for Pb-210 and the quenching parameter a preparation with different coloured samples was made. The result is a linear function between UC and SQP(E).

$$UC = a_1 \cdot SQP(E) + b_1 \qquad (1)$$

with: UC: upper channel of the window of Pb-210
$a_1 = 0{,}661$
$b_1 = -147{,}3$

With each SQP(E) given by the LSC counter the suitable upper channel of the counting window can be calculated.

The role of Bi-210 and Po-210 in the spectrum

Immediately after separation, Bi-210 grows up again and disturbs the detection of Pb-210. Also Po-210 grows up, but this nuclide with its great half-life in comparison of the time after separation can be neglected.
To correct the spectrum of Pb-210 in the counting window from the counts which belong to Bi-210 it must be shown, that the ratio from counts of Bi-210 inside the window to those outside of it is constant with time.
After the measurement of a pure spectrum of Bi-210 this ratio could be fixed as the constant value of $0{,}2958 \pm 0{,}0009$.
When Pb-210 in a sample material is now detected the counts outside the detection window of Pb-210, which all belong to Bi-210 can be measured. With the constant ratio mentioned above it is possible to subtract this fraction from the counts in the detection window.

Counting efficiency for Pb-210

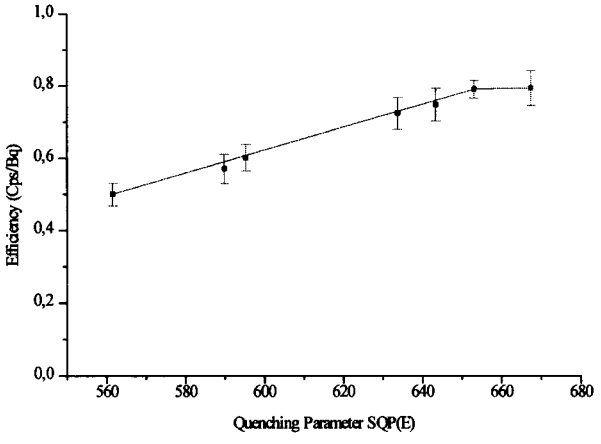

Figure 2: Relation of the counting efficiency for Pb-210 and the quenching parameter SQP(E) (1σ confidence level).
The quenching parameters normally are higher than 653, so that the counting efficiency is constant and $\eta = (0{,}80 \pm 0{,}04)$ Ips/Bq.

Influence of Pb-212 on the spectrum of Pb-210

Pb-212 from the thorium decay chain and its daughters cover a wide range of the measured spectrum by determination in LSC Quantulus. Pb-212 with a half-life of 10,64 h [3], Bi-212 ($T_{1/2}$= 60,6 min), Po-212 ($T_{1/2}$= 305 ns) and Tl-208 ($T_{1/2}$= 3,07 min) therefore also cause a counting rate in the window of Pb-210. But if the ratio of counts inside the window of Pb-210 belonging to Pb-212 and its daughters, to that outside the window is regarded, a constant value over time is obtained. This value, however, is only constant for times after separation which are greater than 4 hours. Then it amounts to 0,349 ± 0,002. This time is also sufficient to eliminate the possible presence of Pb-214 ($T_{1/2}$ = 26,8 min)
With known ratios of counting rates from Bi-210 and Pb-212 (and daughters) in- and outside the window of Pb-210, chemical yield and efficiency for Pb-210 the activity of Pb-210 in the sample can be calculated.

Detection limits and comparison with other possibilities of the measurement of Pb-210

To compare the single methods, detection limits are calculated for a 1000 min measurement and Pb-210 in watery solution. The formulas for detection limit calculate the 3σ confidence level.

Table 1: Comparison of detection limits of the different detector systems for Pb-210 in watery solution; time of measurement: 1000 min, no Pb-212 in the sample material.

		Detection limit (mBq/sample)
high purity Ge detector	100 ml Dose (content 100 ml)	662
	LSC-vial (content 8 ml)	234
NaJ detector	LSC-vial (content 20 ml)	340
LSC Quantulus		**7,4**
gasfilled counter	detection via Bi-210	3,2
α-spectrometry	detection via Po-210	0,3

Table 2: Comparison of sample preparation time

	duration of time in days for			
method	sample preparation	radiochemical cleaning	time between end of analysis and begin of measurement	sum
high purity Ge det.				
not ashed	0	0	0	0
ashed	2	0	0	2
NaJ, ashed	2	0	0	2
gasfilled counter	2	2	5	9
LSC Quantulus	**2**	**2**	**0**	**4**
α-spectrometry	altogether ca. 120 days			

Taking into account also the detection limits mentioned in table 1, the LSC Quantulus has a very low detection limit and the time between drying of the sample and begin of detection for Pb-210 is short.
To compare the results of this method with a high purity Ge detector, soil samples have been measured and the results harmonise within the error bars.

Conclusions

A new method for the detection of Pb-210 has been presented. This nuclide is measured by an LSC Quantulus, which has an extremely low underground counting rate. Therefore very low detection limits can be reached. Pb-210 can be detected after a waiting time of only 4 hours after separation from the daughter product Bi-210. This time is necessary because of the chemoluminescene and the possible presence of disturbing Pb-212.

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THE EVOLUTION OF THE EARTH'S BACKGROUND RADIATION LEVEL OVER GEOLOGIC TIME

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INTRODUCTION

The earth currently has a background gamma radiation level of 0.28 mSv/yr due to radionuclides such as U, Th, Ra, and ^{40}K (1), and humans are exposed to approximately 0.40 mSv/yr from ^{40}K in their bodies. An additional dose of 2.90 mSv/yr is derived from radon (2.0 mSv/yr), building materials (0.28 mSv/yr), cosmic radiation (0.27 mSv/yr) and various anthropogenic sources (1). Owing to the low penetrating power of alpha radiation, radon does not expose the whole body to ionizing radiation. Of the natural sources of gamma radiation the nuclides K, U+progeny, Th+progeny are Large Ion Lithophiles (LILs) and are preferentially partitioned into the continental crust. Since the amount of crust has increased with time, LIL abundance at the earth's surface has increased over geologic time. These elements are all radioactive with half-lives of less than the age of the Earth (except for ^{232}Th), and are now present in lower activity concentrations in the continental crust than at any time in the geologic past.

Prokaryotic life is thought to have evolved between 3.5 and 4 billion years ago (Ga), giving rise to eukaryotic life as early as 2.1 Ga (2) . It is further thought that primitive organisms resembled current life in terms of radiation repair mechanisms, potassium content, and basic biochemistry, based on remarkable similarities in basic chemical compositions of modern primitive and advanced life forms (3).

This paper examines the evolution of the background radiation field in which primitive organisms evolved in terms of internal dose (from internal ^{40}K) and external geologic dose (from gamma emitters in the crust of the earth), and how this may relate to dose-response in modern organisms.

GEOLOGIC AND INTERNAL DOSE

Crustal evolution was not uniform throughout time. Taylor and McClennan (4) suggested that the major epoch of crust formation occurred between 2.8 and 2.6 Ga. The oldest crust known dates from about 3.8 Ga. Ocean chemistry at this time is suggested to be similar to that of today's oceans (5).

LILs are partitioned into the molten phase of magma and discharged into the crust in magma intrusions or erupted as lava. These elements concurrently decay towards stable end products. Since the surface gamma radiation field from these elements is proportional to their activity concentration (6), competing mechanisms determine surface radiation levels from geologic sources (Fig. 1). Using Taylor and McClennan's (4) model for crust formation and assuming that LIL concentration was equal to the percentage of crust formed (i.e. LILs had one half of their current concentrations when the crust was 50% formed) then, at the time that life evolved, surface gamma radiation levels due to contact with the continental crust were similar to those at present, rising to approximately twice current levels at the time that eukaryotic life evolved. Therefore, life would have been exposed to up to four times current levels of external gamma radiation due to contact with the continental crust. In addition, based on a current ^{40}K concentration in seawater of 11 Bq/gm (7), organisms in the water column at that time would have received comparable doses from ^{40}K beta and gamma radiation.

Potassium is essential to life and is not evenly distributed in organisms. Most cells selectively maintain a K concentration approximately one order of magnitude greater than that found in extracellular fluids (8,9). Average K content in human muscle is approximately 155 moles/m^3, compared to an average interstitial fluid content of 4 mol/m^3 (9). Single-celled organisms appear to maintain a similar concentration factor, most at higher internal concentrations. For example, *Acetabularia mediterranea* (a single cell plant) contains 400 mol/m^3 K in its cytoplasm, 355 mol/m^3 in vacuoles, and 10 mol/m^3 in external solution (9). Our model uses the K content of human muscle, which appears to be a conservative estimate, as the basis for internal dose estimates through geologic time.

Radiological dose is measured in terms of energy deposited per unit of mass. Although a single-celled organism has much less potassium than a human, it also has far less mass. Therefore, we assume that the dose due to internal potassium activity remains constant at all scales. Many single-celled organisms are smaller than the mean free path of photons or beta particles in the cell. However, many early organisms were colonial (stromatolites, algal mats, etc). We assume that the thickness of these colonies is greater than the stopping distance of ^{40}K betas in water (approximately 0.7 cm) and that, as a result, the beta decay energy is deposited in the colony with each member receiving an approximately equal share.

Humans currently receive approximately 0.40 mSv/yr from internal ^{40}K. The majority of this dose is due to the 1.33 MeV beta decay (~89%) with the balance resulting from the 1.46 MeV gamma (~11%). The specific

activity of potassium currently is 31.6 Bq/gm. Three half-lives ago (approximately 4 Ga), K had a specific activity of 253 Bq/gm giving a yearly dose of 3.20 mSv.

There is currently a component of dissolved uranium to which we are exposed via drinking water. The short range of an alpha particle results in the deposition of a high percentage of its energy within the cell. However, uranium is insoluble under the reducing conditions which existed during the Archaean and is not likely to have contributed much dose to organisms at this time.

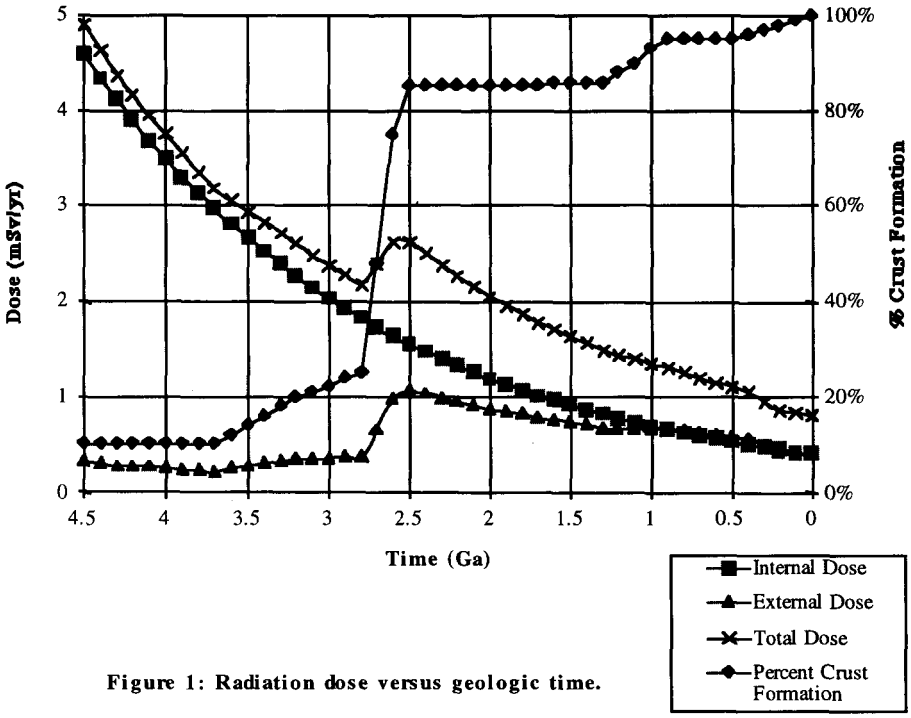

Figure 1: Radiation dose versus geologic time.

ULTRAVIOLET RADIATION

Kasting and others suggested (10) ozone did not exist during the Archaean due to a lack of free oxygen and could not have shielded the earth from ionizing UV radiation. The portion of the UV spectrum most likely to penetrate to the surface is the most energetic (10), UVC (200-280 nm). A high flux of UVC is thought to have sterilized the surface of the earth until a protective ozone layer developed. In addition, penetration of UVC through shallow pools of water or into shallow portions of the ocean may have further increased the ionizing radiation dose to organisms. During the Archaean, solar luminosity was approximately 70% present levels (11). This corresponds to a lower solar surface temperature, giving a UVC flux of 50-70% present levels. Therefore, the UVC flux reaching the earth at this point in time was much less than at present.

While free oxygen did not exist at this time, the earth's atmosphere was considerably thicker than at present, consisting of up to 10 bars of CO_2 with nitrogen, water vapor, nitrous oxides, and trace gases (11). Due to high CO_2 levels, mean surface temperatures were likely between 20° and 50° C, with a corresponding increase in water content and cloud cover (11). In addition, increased volcanic activity would have increased atmospheric loading of aerosols, including sulfur compounds, many of which absorb strongly in the ultraviolet (12). Because of this we feel that, while UVC levels may have been higher prior to formation of the ozone layer, they would not have added significantly to the radiation levels to which early organisms were exposed. We do not know if life evolved at the surface, in the sediments, in shallow water, or in deep water. We do know, however, that the presence of life today indicates that life evolved in an area in which UV radiation was not lethal to it.

DISCUSSION

It seems likely that life was exposed to much higher radiation levels throughout much of its history than exist at present. We estimate the dose due to internal ^{40}K activity to have ranged from four to eight times that at present throughout the time that only prokaryotic and eukaryotic life existed. At the time that metazoan life arose (0.7-1.0 Ga), internal radiation dose was approximately twice that at present.

The dose from geologic materials has changed in the past, although not as monotonically as has internal dose. As the continents formed, the dose from geologic materials increased to four times what we now experience, decreasing due to radioactive decay and, lately, due to the formation of organic soils.

Mutation rates are proportional to radiation dose and dose rate (13). If radiation levels have changed over time, then mutation rates must also have changed over time. The molecular clock, then, may have run faster in the past. In fact, when we modify Runnegar's (13) plot of amino acid substitution versus time (=molecular clock) to incorporate changing ^{40}K activity levels over time, the predictions match the fossil record and the results of Valentine's (15) cell-type model much better. We are presently exploring the significance of ^{40}K activity levels though time on evolution and molecular clock theory.

Cells have mechanisms to repair damage caused by exposure to ionizing radiation. These mechanisms appear to have evolved to contend with radiation levels that were nearly an order of magnitude higher than today. Has this ability been retained by life through geologic time? Even if in the geologic past there was no threshold for "safe" exposure to radiation, falling radiation exposure levels over geologic time suggest that there should be a threshold at the present, as suggested by the recent work of Kondo (16). Is it not possible, if not probable, that the threshold is equal to the background radiation levels that existed at the time when this repair mechanism evolved?

WORK IN PROGRESS

We are currently working to resolve several issues. Among these are the composition and opacity of the early atmosphere to UVC radiation, the effects of increased radiation levels and dose in the geologic past on current ideas in evolution and the molecular clock, the chemical composition of Archaean seawater, and the concentration of U, Th, and K in the earth's crust during its formation.

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IRPA9
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PAPER TITLE MEASUREMENTS OF THE COSMIC RADIATION FIELD IN A JET AIRCRAFT AT COMMERCIAL AVIATION ALTITUDES

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MAJOR SCIENTIFIC TOPIC NUMBER 1.3 (see page 7)

ABSTRACT (See instructions overleaf)

Crews working on jet aircraft receive one of the highest average equivalent doses of any occupationally exposed group, but because of the complex nature of the atmospheric cosmic radiation field, there are large uncertainties in aircrew exposures and associated health risks. Neutrons contribute roughly half of the equivalent dose, and the large uncertainty in the neutron spectrum makes the equivalent dose uncertain by a factor of 2.

To reduce these uncertainties and evaluate instruments for monitoring aircraft radiation, we have measured various components of the radiation field aboard a Canadian Forces Boeing 707 aircraft during 3 flights covering 5 altitudes from 10 to 12.5 km (33,000 to 41,000 ft.) and 52°-70° N geomagnetic latitude. Neutron energy spectra were measured using a high-sensitivity multisphere spectrometer. The neutron dose equivalent was measured using a remmeter, a lead-covered remmeter, and superheated-drop/bubble detectors. Gamma-ray and charged-particle energy spectra were measured using BGO, NaI(Tl), and plastic scintillators with anticoincidence shells. The dose rate from a pressurized argon ionization chamber was recorded every minute, and the dose for each flight was measured using three types of TLDs. The TLDs and bubble detectors were located within and on an anthropomorphic phantom and throughout the aircraft.

We will present results for the ionization dose rate, neutron dose equivalent and gamma-ray spectrum as a function of altitude and geomagnetic latitude, and doses in the phantom for each flight. Methods for determining the high-energy response of the neutron detectors will be discussed.

COSMIC RADIATION DOSES IN COMMERCIAL AIR TRAVEL

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INTRODUCTION

Cosmic radiation doses received in commercial air travel are considered important because of the large number of passengers carried, the long distance traveled, and the high geomagnetic latitude at some flights occurred.[1] In particular, the crew members of pilot and flight attendant serve more than 720 flight-hours each year , exposing to the cosmic radiation much more than those received by average passengers. The in-flight cosmic radiations are measured using pocket-sized solid state detector for electromagnetic radiation and the bubble detector for fast neutrons. Both domestic and international flight routes of airlines homebased in Taiwan are chosen for in-flight measurements, as illustrated in Fig.1

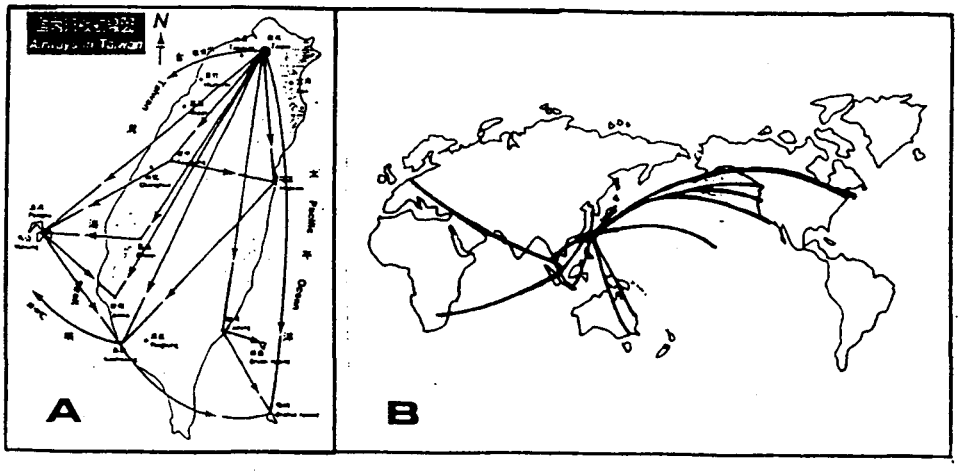

Fig. 1. Domestic and international flight routes of Taiwanese air-liners.

RESULTS

Sensitivity of both detectors are 0.01 μSv/h for gamma rays and 3 bubbles /μSv for neutrons, satisfactory for the in-flight measurements. Calibration of both detectors are conducted on ground using reference sources of high-energy gamma and neutrons. The measurements are carried out in all commercial flights served by the Eva Air and China Airline; the flight patern of initial and final ascending/descending, crusing, and landing approach are logged on-line, while the in-flight doses are recorded in cumulative modes. Typical result is 42 ± 3 μSv for Taipei-NYC flight route, to which the neutrons contribute more than 50% of the total cosmic doses. Annual dose for crew members can be assessed by the measured data and the information of their flight-hours, aircraft types, flight routes, and flight patern. The annual doses for crewmembers are 108 μSv for domestic service and up to 7200 μSv for international service, as summarized in Table 1.

Table 1. Air travel statistics and in-flight dose assessment for civil aviation in Taiwan

Statistics	Flight routes Domestic	International	Evaluation	Flight routes Domestic	International
Average crusing altitude (ft)	14,000	37,000	In-flight dose rate (μSv/hr)	0.15-0.80	3-10
Average distance per flight (km)	135	2,613	Dose per flight (μSv)	0.06-0.34	9.8-32.5
Average air borne time per flight (hr)	0.42	3.25	Annual collective dose(man-Sv)	0.82-4.41	82.2-274
Flights per day[1]	811	81	Annual dose (μSv)	0.12- 0.68	19.6-65
Flights per year	296,056	29,429	Contribution to population (μSv/yr)[3]	0.04-0.23	4.13-13.76
Seats per flight[2]	81	287			
Seats per year (millions)	13.12	8.43			
Seat-kilometers per year(giga person-km)	1.77	22.03			

Note 1. Only commercial flights landing to and taking-off from airports in Taiwan are accounted for.
2. Only citizens of Taiwan who travel on board are accounted for, foreign travelers are excluded.
3. The population in Taiwan is 20,752,000 in 1992; the in-flight portion (4.14-13.99 μSv/yr) is far less than the ground radiation background caused by cosmic origin (around 280 μSv/yr).

This work is financially supported by the National Science Council of the Republic of China under contract number NSC85-2212-E-007-053.

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ACTIVE DOSIMETER FOR ON BOARD AIR - CREW - EXPOSURE

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International Commission on Radiation Protection ICRP60 recommend that the cosmic radiation exposure in civil aviation should be taken into account as occupational exposure. Recent measurements at civil airlines show a radiation exposure in the range of 3 - 10 mSv / year. An economic and simple active method to assess the radiation exposure in aviation altitudes is described in this paper. This method takes account of the contribution of solar particle events, changing flight pattern and altitude. First experimental data by in-flight measurement are presented.

INTRODUCTION

In 1990 ICRP published its report ICRP 60 with updated excess cancer risk estimates, which led to significantly higher risk coefficients for some radiation qualities. An increase of the radiation weighting factors for mean energy neutron radiation increases the contribution for the neutron component to the equivalent dose by about 60%, as compared to the earlier values of ICRP26. This higher risk coefficients lead to the recommendation of the ICRP, that cosmic radiation exposure in civil aviation should be taken into account as occupational exposure. Numerous recent exposure measurements at civil airliners in Germany, Sweden, USA, and Russia show exposure levels in the range of 3 - 10 mSv / year. This is significantly more than the average annual dose of radiation workers (in Austria about 1.5 mSv / year). Up to now no practicable and economic radiation monitoring system for routine application on board exits. During the last year the Seibersdorf group for radiation protection studied an active dose assessment method for aircrew [1].

ACREM CONCEPT

The exposure of air crew can be assessed by different methods [1], basically by using active or passive dosimeters. Detailed in-flight dose investigations on different positions at aircraft has shown that individual dosimetry for each crew member is unnecessary, since the personal and area dose equivalent are essentially independent of the location at aircraft.

Several cosmic ray transport codes exist which are able to calculate the radiation exposure at aviation altitudes. The well-known CARI code has been used for some years to calculate the equivalent dose between two flight destinations and was up-dated last year by CARI 3. Basis for the CARI 3 calculation is the more detailed transport code called LUIN94 [2]. Even though the calculations are in agreement with measurement results, for dose assessment they require knowledge of the actual flight pattern and can not account for unexpected solar particle events. To solve this problem active dosimetry is necessary.

An economical and simple active method to assess the radiation exposure, presently being studied in our laboratory, is realised with ACREM (Air Crew Radiation Exposure Measuring) system. ACREM is based on a combination of measurement of the ionising component and calculation of the neutron component. Up to know we use a GM-counter with an energy range of 50 keV to 1.3 MeV for dose measurements. The instrument is based on the Seibersdorf Radiation Protection Survey Meter SSM-1, designed for Austrian military and civil protection. Two GM-detectors and microprocessor controlled electronic, are used for real-time, continuous measurement of the equivalent dose (rate) at aircraft.

For calculations the transport code LUIN94 is used. LUIN94 provides the neutron component, as well as the ionising component of equivalent dose caused by cosmic ray for different flight patterns and altitudes. The sum of both components gives the total equivalent dose. The relation of the total calculated equivalent dose to the calculated ionising component gives a conversion factor for a certain aircraft position. This factor is tabulated for suitable values in altitudes and grid distances and is stored in a database of a micro computer system. This provides fast data acquisition. The ionising component measured by a GM-counter, is weighted with the

1 Developed under Research Contract by Allgemeine Unfallversicherungsanstalt, Vienna; Patents pending

calculated conversion factor. The result gives an estimation of the equivalent dose (rate) in typical time intervals of some 10 seconds, for a momentary flight position. The dose value is stored in real-time mode and direct readable on a display. A warning system tells about unusual high dose rates. Dose assessment to crew members is achieved by personal chip card. The concept of the Air Crew Radiation Monitor is shown in fig.1.

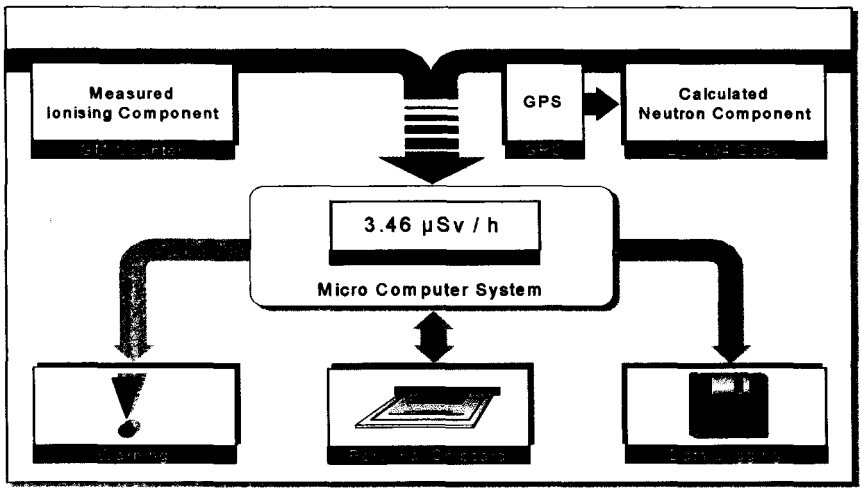

Fig. 1: The concept of the active aircrew dose assessment system - ACREM

PRELIMINARY RESULTS

Recently first in-flight measurements have been performed in co-operation with Universität der Bundeswehr, Munich, Germany. The experiments were performed on a Boeing 707 of the German Airforce during a flight from Cologne to El Paso with an intermediate landing in Washington. A low pressure tissue-equivalent proportional counter (TEPC) serves as reference instrument. The used TEPC system with an internal ^{244}Cm calibration source, has been described in greater detail in [3]. Additionally a Reuter-Stokes, high pressure ionisation chamber was on-board to compare results with the low-LET component of TEPC measurements. All detectors were calibrated for ^{137}Cs at the radiation calibration facility of Seibersdorf. The overall uncertainty for measurements by the GM-counter as well as by the ionisation chamber is about 20 %. The uncertainty for TEPC measurements is nearly the same.

For first data analysis the ionising component of cosmic rays and the low-LET measurements by TEPC will be discussed. Fig.2 shows the dose equivalent rate in µSv/h provided by measurements with the GM-counter (SSM-1), the ionisation chamber (Reuter Stokes RS 111) and the TEPC (low-LET) at the flight route Washington - El Paso. Additionally the dose equivalent rate for the ionising component, calculated by LUIN94, is included for several flight-positions. The integration time for GM-counter and ionisation chamber data points was about 30 seconds. The bar of TEPC data points indicates a measurement time of 1000 seconds. Fig.3 shows the same measurement situation like Fig.2 but for the flight route Washington - Cologne.

During the Washington- El Paso flight mostly the same altitude was used: 39.000 feet. The latitude decrease from 39° north to less than 32° north rather directly. All measurements and the calculations describe this well known fact by reduction of the dose equivalent value, caused by interference of cosmic rays and earth magnetic field. The largest dose value of 4.5 µSv/h was measured by the GM-counter at the most northern position of this flight, for measurements by the Reuter Stokes chamber 3.7 µSv/h at the same position. The TEPC low-LET data and the LUIN94 data fit the Reuter Stokes data very well.

At the flight from Washington to Cologne several different altitudes have been used. All instruments describe the change of altitudes very precisely. The latitude increase from 39° north to 55° north (about 3:00 UTC) and decrease at Cologne to 50° north. Since the changing of altitude and latitude goes in parallel a significantly observations of the "latitude-effect" like at the Washington - El Paso flight can not be expected. More detailed analysis would be necessary. Dose equivalent values range from 2.1 µSv/h at 29.000 feet, 3 µSv/h at 33.000 feet to about 3.5 µSv/h at 35.000 feet.

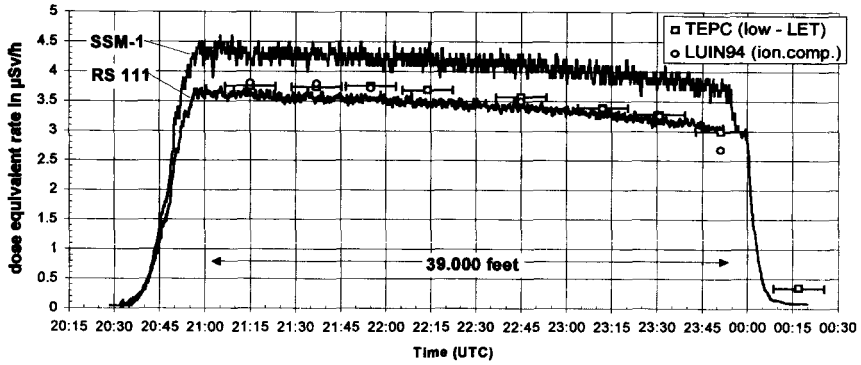

Fig. 2 : Dose equivalent rate determined by GM-counter (SSM-1), Ionisation Chamber (RS 111), TEPC (low LET component) and LUIN94 (ionising component) at the flight route: Washington - El Paso.

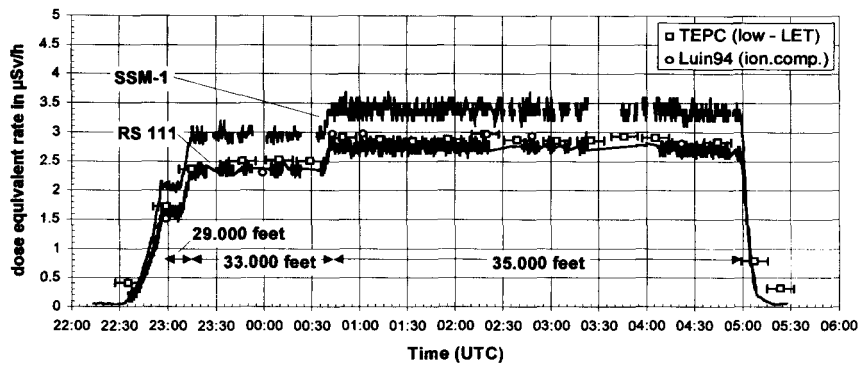

Fig. 3: Dose equivalent rate determined by GM-counter (SSM-1), Ionisation Chamber (RS 111), TEPC (low LET component) and LUIN94 (ionising component) at the flight route: Washington - Cologne.

CONCLUSION

Taking measurement uncertainties into consideration no significant difference between used instruments and calculations could be observed.

ACREM, if eventually verified under routine flight conditions, can be a simple and economical solution for active airborne dosimetry. This method can also account for the contribution of solar particle events, changing flight pattern and altitude.

ACKNOWLEDGEMENTS

The financial support by the Austrian Allgemeine Unfallversicherungsanstalt is gratefully acknowledged.

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DETERMINATION OF THE AVERAGE LET AND THE EQUIVALENT DOSE IN AIRCRAFTS USING THE HTR-METHOD

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ABSTRACT

Measurements of the average LET and the equvialent doserate in high altitude aircrafts were performed with TLDs using absorber material of varying thickness. Besides these TLD-packets Bonner spheres were used.
The determination of the equivalent doserate was carried out with two methods. With the Bonner spheres calibrated for their neutron energy dependent response the equivalent doserate induced by neutrons and gammas is measured with TLD-600 and TLD-700 in pair use.
In addition the high temperature region of the TLD-700 is analysed. Using a LET calibration of the HTR (high temperature ratio) the quality factor and the average equivalent dose can be determined in dependence on the absorber thickness. The deviation in the equivalent doserate of the two methods is discussed. Furthermore the results of two different flight periods are shown.

INTRODUCTION

Interest in radiation protection measurements in mixed (n,γ)-fields as they occure in high altitude aircrafts has increased with improved knowledge of the biological effectiveness of some of the components. The mixed radiation field induced by solar and galactic cosmic radiation consists of a broad spectrum of different particles (protons, neutrons, electrons, gamma-quants,..) with varying energy and LET. Beside absorbed dose measurements particular investigations for the evaluation of the LET-spectrum of the mixed radiation field have been made using LET-spectrometers (1). Furthermore increased efforts are made concerning neutron dosimetry in aircrafts (2) because about half of the equivalent dose is caused by neutrons of various energies. Since most of the applied systems are very complex, unwieldy and heavy, especially for radiation protection purpose investigations based on TLDs are of particular interest.
TLDs are mainly used to obtain the absorbed dose of the ionising component of the mixed radiation field (3, 4). Analysing the high temperature region (5) the average LET can be evaluated in addition. Beside determination of the equivalent dose using the high temperature ratio (HTR), the neutron induced equivalent dose was measured with a calibrated neutron detector.

MEASUREMENTS IN AIRCRAFTS

Measurements in aircrafts were performed during two different flight periods. Flight period 1 covers three flights Köln - Washington - Köln (exposure time: 60.25 h), during flight period 2 the dosemeters were exposed on four flights the same distances (exposure time: 108 h). For both periods the latitude was between 32 - 52°N in 10 - 11 km altitude.
For flight 1 a standard packet (3mm poystyrol), a standard packet with 20 mm polystyrol in addition and a Bonner sphere with 25 cm diameter was used. Since a dependence of the equivalent dose on the absorbing layer was measured, for flight 2 a second Bonner sphere with 12 cm diameter was used to evaluate the depth dependence in more detail. Each packet included TLD-600, TLD-700 and TLD-200, the two spheres only TLD-600 and TLD-700. During flight 2 labor-made single crystals with different magnesium content were exposed in one of the standard packets.

DETERMINATION OF THE EQUIVALENT DOSERATE CAUSED BY NEUTRONS

For tissue equivalent dose measurements of the neutron dose a neutron detector consisting of TLDs embeded in the centre of a Bonner sphere with 25 cm diameter was used. This sphere enables nearly neutron energy independent measurements in the energy region between approximately 0.025 eV and 10 MeV. In the centre of the moderator sphere a special inset containing the TLDs was developed to improve neutron shielding of the TLD-700 particular against thermal neutrons: TLD-700 situated in the centre of the compartment are surrounded by TLD-600.
To determine the equivalent dose caused by neutrons in mixed (n,γ)-fields the pair method is applied to discriminate between the neutron- and the γ-component (6). TLD-700 were used to measure the γ-dose. The TLD-600 glowcurve is caused by composition of the γ- and the thermal neutron component. Therefore the glowcurve caused by neutrons can be obtained by subtraction of the γ-fraction of the total glowcurve. The calibration of the neutron detector is described in more detail elsewhere (7). Analysing the peak 5 maximum of this neutron glowcurve and using the energy dependent calibration the neutron induced equivalent dose can be determined.

The neutron equivalent doserate measured in the aircraft with the calibrated neutron detector was 3.7 ± 0.5 μSv/h for flight 1 and 2.6 ± 0.4 μSv/h for flight 2. For the calculation a mean calibration factor of 1473 ± 221 cts/mSv was used. The total equivalent doserate is obtained by addition of the absorbed doserate measured with TLD-700. This gives a total equvialent doserate of 5.7 ± 0.5 μSv/h (flight 1) and 3.9 ± 0.4 μSv/h (flight 2).

DETERMINATION OF THE TOTAL EQUIVALENT DOSERATE

The absorbed dose in the aircraft was measured with TLD-700 and TLD-200. Table 1 shows the dependence of the absorbed doserate on the layer thickness. Measuerments of period 1 indicated that the dose increases with absorber thickness. The more detailed measuements of period 2 however shows a maximum at 6 cm depth. This function of the dose is the result of the production of secondary protons by neutrons penetrating the absorbing material. This is also indicated by the dependence of the neutron flux on the absorbing layer (table 1). For flight 2 two identical standard packets were used which show a discrepance in the absorbed doserate. This can be due to a different positions of the packets with different shielding thickness or material in the aircraft.
Furthermore the absorbed doserate was measured with labor made single crystal exposed in packet 1 during flight 2. Crystal E (280 ppm Mg, 11.5 ppm Ti) gives a doserate of 1.83 ± 0.07 μGy/h, crystal F (150 ppm Mg, 11.5 ppm Ti) 1.73 ± 0.11 μGy/h and crystal I (410 ppm Mg, 28.8 ppm Ti) 2.06 ± 0.05 μGy/h. These values are in good agreement with the doserate obtained with TLD-700.

	absorbed doserate [μGy/h]		thermal n-fluence [n/cm²s]	
	flight 1	flight 2	flight 1	flight 2
packet 1	1.50 ± 0.09	1.32 ± 0.07 2.20 ± 0.06	0.61 ± 0.20	0.35 ± 0.08 0.23 ± 0.05
packet 2	1.69 ± 0.04	-	0.56 ± 0.05	-
sphere 1	-	1.50 ± 0.10	-	0.71 ± 0.02
sphere 2	2.00 ± 0.03	1.09 ± 0.05	0.44 ± 0.05 0.49 ± 0.03	0.35 ± 0.06

table 1: Dependence of the γ-doserate (measured with TLD-700) and the thermal neutron flux on the absorbing layer. For calculation of the neutron flux the peak 5 maximum was analysed after subtraction of the γ-component, the calibration factor is $3.03 \cdot 10^{-3}$ cts/ncm².

In addition the total equivalent doserate was evaluated applying the HTR-method. This method compares the high temperature region of the glowcurves with a Co-60 irradiation, normalized on the peak 5 maximum. Since this ratio correlates with the distributaion of the energy deposition, a LET-calibration allows the determination of the average LET of unknown mixed radiation fields (5). The method was used to evaluate the average LET of the measurements in the aircraft.
The increase of the high temperature region of TLD-700 is shown in figure 1. According to the measurements of the neutron flux, an increase of the LET respectively the equivalent dose with absorber thickness was obtained for both periods (table 2).

	HTR		$LET_{average}$ ICRP 26 [keV/μm]		$Q_{average}$		H [μSv/h]	
	flight 1	flight 2	flight 1	flight 2	flight 1	flight 2	flight 1	flight 2
packet 1	4.4 ± 0.8	1.57 ± 0.1	18.0 ± 3.6	8.7 ± 0.3	4.1 ± 0.4	2.4 ± 0.2	8.3 ± 1.6	3.1 ± 0.3
packet 2	3.2 ± 0.6	-	15.0 ± 2.3	-	3.5 ± 0.8	-	7.4 ± 1.5	-
sphere 1	-	2.81 ± 0.3	-	16.3 ± 1.7	-	3.7 ± 0.6	-	5.6 ± 0.9
sphere 2	6.6 ± 1.8	1.71 ± 0.1	25.4 ± 7.1	9.5 ± 0.6	5.4 ± 1.3	2.5 ± 0.3	15.4 ± 4.6	2.7 ± 0.3

table 2: Evaluation of the average LET and the equivalent doserate using the HTR-method. If measurements were made using two equal packets for one flight the mean value is given. The increase of the high temperature region of the TLD-700 glowcurves is shown for both flights in figure 1. $LET_{average}$ is only meaningfull using ICRP 26 because ICRP 60 demands the assessment of the various components of the mixed radiation field.

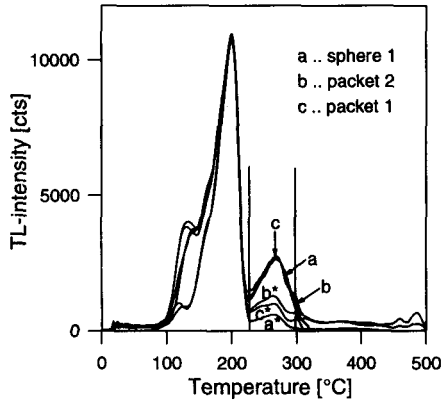

figure 2: Comparision of the TLD-700 glowcurves obtained during flight 1 (a, b, c) with the corresponding Co-60 calibration (a*, b*, c*). Glowcurves are normalized on peak 5 maximum.

DISCUSSION

Comparision of the neutron equivalent doserate of the two flights shows that flight 2 gives only about 70% of the doserate obtained on flight 1. This situation can also be seen by comparing the results of the HTR-method. The values differ between flight 1 and 2 about a factor of 2 to 5 depending on the absorber thickness. Part of this discrepance can be explained by the lower neutron flux during flight 2 (table 1). This indicates that the particle spectrum must have changed quite significantly in this short period between flight 1 and 2. A further indication of this variance of the particle specturm is given by analysis of the dependence of the equivalent dose on the absorbing layer. Flight 1 shows an increase of the equivalent doserate with absorber thickness whereas on flight 2 packet 1 with 1.5 mm absorber and sphere 2 with 12 cm absorber give the same equivalent doserate, only at 6 cm depth a maximum occurs. This discrepancy can be caused by a differing neutron and proton spectrum during each flight. A direct correlation of the results obtained with the HTR-method and the variation of the solar activity was found comparing measurements performed in 1989 and the flights in 1995. In 1989 (route: Köln - Aschchabad, altitude: 10 - 11 km, latitude: 40 - 55 °N) an absorbed doserate of 2.3 ± 0.2 µGy/h, a neutron flux of 0.17 ± 0.05 n/cm²s and an equivalent doserate of 5.2 ± 0.22 µSv/h were measured. If these values are compared with the results of the two flights in 1995 a decreased absorbed doserate and an increased neutron flux is determined. In correlation to these results in 1989 a solar maximum reduced the primary proton and neutron component compared with the situation of 1995 (minimum solar acitivity).

CONCLUSION

The deviation of all the investigated parameters compared with former measurements correlates with the variation of the solar activity. Furthermore the differences of the parameters between flight 1 and 2 explained by a variing particle spectrum shows the sensitivity of the HTR-method. It can be seen that the composition of the mixed radiation field in aircrafts changes significantly with the solar acitivity. Since these variations cannot be predicted continous measurements of the absorbed dose and the LET are proposed.

Using TLD-700 not only the absorbed doserate can be determined but also the evaluation of the average LET is possible. The differences in the equivalent doserate dependence on the absorber thickness for the two flights show that further efforts have to be made to obtain the depth dose in more detail. Measuerments with more Bonner spheres of different diameters are planned.

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CONTRIBUTION OF SECONDARY HIGH LET PARTICLES TO THE ENERGY TRANSFER FROM HIGH ENERGY PROTONS

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INTRODUCTION

The most of dose absorbed in the human tissue from cosmic radiation outside of the Earths' atmosphere is delivered through high energy proton energy transfer (1). There are principally two basic processes contributing to this transfer:

- ionization losses, and
- the energy transfer due to high energy protons interactions with tissue constituting nuclei.

This paper presents the results of both theoretical and experimental studies of the energy transfer from high energy protons to tissue through the second process mentioned above, i.e. via secondary, mostly high LET (LET - linear energy transfer), charged particles. Available cross sections data were used to characterize the doses due to these secondary particles theoretically. Theoretical results are compared with experimental data obtained through the response of track etch detectors to high energy protons.

THEORETICAL CALCULATIONS

Let a broad beam of protons of the energy E_p^0 impinge on a slab of the thickness x_0. The number of secondary particles leaving such slab can be calculated through the procedure described in more detail preciously (2). The calculations have been performed for the slab thickness x_0 equal to 0.1; 0.3; 0.5, and 1 g.cm^{-2} of the tissue, and for the initial proton energies equal to 100; 200; 350 and 660 MeV. Following types of secondary charged particles have been taken into account:

- protons created through the cascade processes, i.e. with LET up to 30 MeV.cm^{-1};
- particles created through evaporation processes, i.e.:
  - protons with LET between 30 and 145 MeV.cm^{-1};
  - α-particles with LET between 150 and 1900 MeV.cm^{-1}; and
  - heavier fragments (z~4) with LET between 1200 and 5200 MeV.cm^{-1};and
  - recoil particles (z~7) with LET between 6000 and 10000 MeV.cm^{-1}.

Some simplifications and additional hypothesis have been admitted:

- the independence of the inelastic processes cross section within the slab thickness;
- the independence of the differential evaporation cross section on the proton energy between 100 and 660 MeV;
- only first generation of nuclear processes has been considered
- the ratio of forward and backward evaporation cross sections is equal to about 2;
- the form of spectra of any secondary particle in a group mentioned above is the same.

The results of calculations showed that:

- the integral LET spectra of secondary charged particles do not depend on the proton energy between 100 and 660 MeV;
- absolute contribution of particles is the same, relatively it increases with decreasing ionization losses, i.e. with increasing proton energy;
- the thickness x_0 starts to be important only at the lowest LET values.

The integral LET spectra of secondary charged particles obtained are presented in Fig. 1.

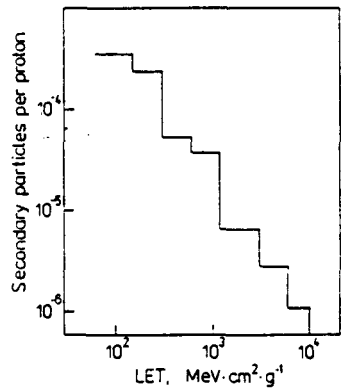

Figure 1. Calculated integral LET spectra of secondary charged particles (x_0 = 0.3÷1g.cm^{-2}).

EXPERIMENTAL RESULTS

Experimental data on LET spectra of secondary charged particles have been obtained by means of track etch detectors (TED), devices, response of which is directly related to the LET (3). There are principally two possibilities how to determine LET spectra of charged particles in a TED:

- to use several TED with different LET thresholds; or
- to determine LET spectra through the analysis of parameters of an individual chrged particle track (5).

Both approaches have been used in our studies. Responses of several different TED to high energy protons have been recieved in our recent publication (4). It was found out that numbers of registered tracks do not depend on the proton energy above about 100 MeV. Their numbers are equal to about $4x10^{-5}$ per perpendiculary incident proton in the use of electrochemically etched CR 39 (LET threshold about 300 MeV.cm^{-1}), about $1.5x10^{-5}$ for cellulose nitrate LR 115 (threshold about 800 MeV.cm^{-1}), respectively about $1x10^{-5}$ in the case of electrochemically etched MELINEX 0 (threshold about 7000 MeV.cm^{-1}). All these figures are close to the theoretically calculated values (see Fig.1).

Much more straightforward comparison can be obtained through the determination of full LET spectra of secondary particles registered in detectors irradiated with high energy protons. Typical LET spectra in CR 39 TED after the irradiation with high energy protons are presented in Figure 2. One can see there that the general shapes of curves in Figures 1 and 2 are very similar.

Both theoretically calculated and experimentally measured LET spectra of secondary particles permit to estimate the dose absorbed in a thin layer of a material due to these particles. The dose is calculated through the equation:

$$D = \int(dN/dL) \times L \times dL, \quad \text{where}$$

dN/dL is the number of tracks in a LET interval; and
L is the value of LET.

When only particles with LET higher than 100 MeV.cm^2.g^{-1} (threshold of registration by CR 39 TED) are taken into account, the doses absorbed in a layer of the thickness of 0.1 g.cm^{-2} can be obtained from both theoretical and experimental data. The results obtained are presented in Table 1.

One can see there that both, quite different ways of the estimation of the secondary high LET particles contribution to the dose from high energy protons give quite comparable results, some

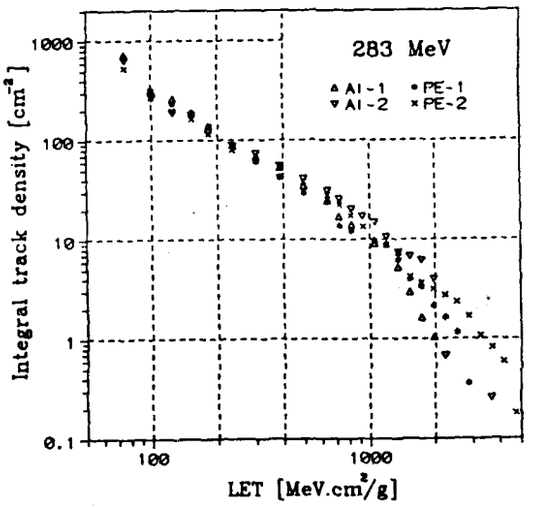

Figure 2. LET spectra of secondary charged particles as measured in CR 39 TED irradiated with high energy protons (Al, PE - covered with aluminium or polyethylene).

Table 1: Dose conversion factors in small layer of tissue-like material.

Initial proton energy [MeV]	Dose conversion factor, Gy.m²; due to					
	Secondary particles with LET higher than 100 MeV.cm².g⁻¹				Ionization losses [5]	
	Theory		Experiment			
100	1.71	-11	1.80	-11	1.15	-9
200	1.68	-11	1.88	-11[1]	7.08	-10
350	1.63	-11	1.16	-11[2]	5.11	-10
660	1.56	-11	-		3.89	-10

[1] For 160 MeV; [2] For 238 MeV

differences observed will be further studied. The data obtained show that the relative contribution of such particles is relatively low at energies about 100 MeV (and less). At 660 MeV, however, this contribution increases substantially, when higher quality factor for these secondary particles is taken into account the relative contribution to dose equivalent values exceed 10%. Further studies of this topic are in progress.

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MEASUREMENTS OF ABSORBED DOSE AND AVERAGE LET IN THE SPACE STATION MIR AND ON SPACE SHUTTLES **

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ABSTRACT

For determination of the absorbed dose and the equivalent dose in complex mixed radiation fields, a new method was developed. The method utilizes the changes of peak height ratios in thermoluminescence glowcurves. Peak height ratios depend on the linear energy transfer (LET) of absorbed radiation. This effect was calibrated in different radiation fields (alpha-, beta-, gamma-, neutron fields and heavy charged particle beams). The method was approached for dose measurements in space. Measurements were carried out on space station MIR during the mission AUSTROMIR and during the Russian Long Term Flight RLF experiment ADLET. During this experiment dosemeter packets were exposed under different shielding conditions in order to measure the variation of absorbed dose as well as the variation of average LET of absorbed radiation within the habitable part of space station MIR. The results are compared with calculations based on models for trapped belt radiation (AP8). Additional measurements were carried out within space shuttles during the missions STS-60 and STS-63. Our results are compared with results obtained by Russian specialists (IMBP) and by the Space Radiation Analyses Group (NASA).

INTRODUCTION

In space respectively in a space station humans and matter are exposed to charged particles as well as to electromagnetic ionizing radiation (X-rays and gamma radiation) with various energies. The radiation environment in space results from three main components: galactic cosmic radiation, solar particle-event radiation and geomagnetically trapped radiation. Interaction processes with matter transform this primary space radiation into secondary radiation consisting of charged particles, neutrons, gamma - and X-rays. A considerable part of the resulting radiation is characterized by a high linear energy transfer (LET). For determination of the biologically relevant equivalent dose, beside the knowledge of the absorbed dose, information about the LET-spectrum is required. The doserate and the LET-spectrum within a spacecraft depend on the orbit parameters, the time (solar flares) and the shielding conditions.

SCIENTIFIC METHODS

In standard TL-dosimetry with LiF dosemeters light emission at temperatures below about 240 °C is analysed. The intensity of the dominant peak (peak 5) in the glowcurve appearing at about 200 °C is linear over a wide range of absorbed dose. For evaluation of absorbed dose either peak amplitude or peak integral is used. Peak ratios are more or less constant at low doses after beta or gamma iradiation. After absorption of radiation with higher LET, peak ratios change significantly. In general high temperature peaks arising from recombination of thermally more stable thermoluminescence centers are enhanced after absorption of high LET radiation. These peaks appear in the temperature range from 240 to 350 °C. In our former investigations we calibrated this effect with test irradiations of different LET (1.2.3.and 9). As parameter for the average LET of absorbed radiation we used the ratio of the intensity of TL-light in the high temperature region (HTR) to the intensity of emission in the same temperature range after Co-60 (gamma) irradiation. Intensity of the dominant peak 5 was normalized before calculation of the HTR-ratio.

EXPERIMENTAL

The experiment ADLET was carried out on space station MIR during the Russian-Austrian joint project RLF (Russian Long Duration Flight). Six dosemeter packets designed for a former experiment (2) were used for measurements. Absorbed dose and average LET was recorded over three periods (RLF-1: 8.1.1994 - 9.7.1994, duration 183 d; RLF-2: 8.1.1994 - 4.11.1994, duration 300 d and RLF-3: 8.1.1994 - 22.3.1995, duration 437 d). During each period packets were exposed in the cabin of the board engineer, position 1 (minimum shielding, packets no.9,11,12); and in the central working area, position 2 (maximum shielding, packets no.10,13,14), in order to measure the variations of doserate and average LET respectively quality factor within the habitable area of the space station MIR. The positions of the dosemeter packets are indicated in the scheme of space station MIR (fig.1).
During the shuttle missions STS-60 and STS-63 we could take part in an US-Russian dose measurement intercomparison experiment. So our data can be compared with Russian and NASA dosimetry results, and with data gathered with a TEPC by NASA. All our TL-dosemeters were read out with the labor-made read-out system TL-DAT-II using a Thorn Emi 9635QB photomultiplier and evaluated by a specifically developed software (4).

** **Work supported by the Austrian Federal Ministry for Science, Research and Arts and the Austrian Society for Aerospace Medicine, ASM**

RESULTS

For the measurement of the absorbed dose the amplitude of peak 5 in LiF-dosemeters was used. All dosemeters were calibrated with Co-60 gamma radiation. Gathered data show that the measured average doserate depends very strongly from position. During period RLF-1 the ratio of measured absorbed dose in the two positions was 1.48. This difference depends on exposure period in space. During period RLF-2 the ratio was only 1.27. The absorbed doserate varied from 0.205±0.01 mGy/d (RLF-1 pos.2) to 0.302±0.01 mGy/d (RLF-1 pos.1). Austrian and Russian results of absorbed dose are in good agreement (see table 1: Dose-ATI: Austrian results, Dose-IMBP: Russian results). During STS-60 mission we measured an absorbed dose of 1.37±0.09 mGy with TLD-600 and 1.24±0.08 mGy with TLD-700. These results are in excellent agreement with NASA results from TLD-700 in compareable position: 1.24 ± 0.03 mGy.

For evaluation of average LET of absorbed radiation the method of HTR was approached (1,2,3,9). Glowcurves from LiF-TLDs exposed in space show a significantly enhanced high temperature TL-emission compared to glowcurves after Co-60 irradiation (fig.2). According to LET-calibration a maximum variation of average LET from 5.8±0.2 keV/µm (RLF-2 pos.1) to 7.1±0.1 keV/µm in tissue (RLF-1 pos.2) was measured. A summary of results from space station MIR is shown in table 1.

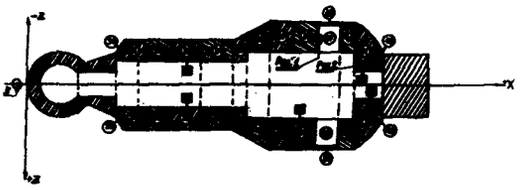

Fig.1: Scheme of space station MIR used for calculation of mass distribution. The positions of dosemeters are indicated.

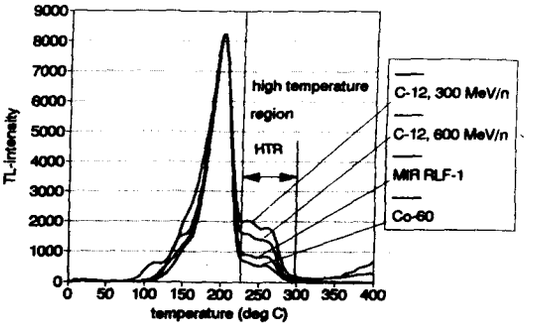

Fig.2: Glow curve from TLD-600 exposed on space station MIR compared to glow curves after irradiation with C-12 ions and Co-60 gamma radiation

Table 1: Summary of results from dose measurements on space station MIR

Parameters	DOSIMIR Ph.1	DOSIMIR Ph.2	RLF Phase 1	
Duration (d)	145	8	182	
	Phase 1	Phase 2	Position 1	Position 2
Dose -ATI (mGy tissue)	34.2 ± 1.1	1.61 ± 0.02	55 ± 1.8	37.3 ± 1.7
Dose -IMBP (mGy tissue)	-	1.74 ± 0.08	59.1 ± 5.0	37.15 ± 3.1
Doserate -ATI (mGy/d)	0.240 ± 0.008	0.201 ± 0.003	0.302 ± 0.01	0.205 ± 0.01
av. LET (keV/µm)/tissue	6.2 ± 0.3	6.5 ± 0.3	7 ± 0.1	7.1 ± 0.1
Quality Factor	**1.8**	**1.9**	**2**	**2.1**
Equivalent Dose (mSv)	61.6 ± 2	3.06 ± 0.04	110 ± 3.7	78.2 ± 3.7
Equiv. Doserate (mSv/d)	0.43 ± 0.01	0.38 ± 0.01	0.60 ± 0.02	0.43 ± 0.02

Parameters	RLF Phase 2		RLF Phase 3	
Duration (d)	300		437	
	Position 1	Position 2	Position 1	Position 2
Dose -ATI (mGy tissue)	90.3 ± 3.4	70.8 ± 4.7	125.7 ± 4.5	96.0 ± 4.3
Dose -IMBP (mGy tissue)	83.5 ± 1.9	69.6 ± 4.3	130.5 ± 9.0	100.2 ± 4.4
Doserate (mGy/d)	0.301 ± 0.01	0.236 ± 0.02	0.288 ± 0.01	0.220 ± 0.01
av. LET (keV/µm)/tissue	5.8 ± 0.2	6.8 ± 0.4	6.4 ± 0.1	6.7± 0.2
Quality Factor	**1.65**	**1.9**	**1.8**	**1.9**
Equiv. Dose (mSv)	149 ± 5.7	135 ± 8.9	226 ± 8.1	182 ± 8.2
Equiv. Doserate(mSv/d)	0.5 ± 0.02	0.45 ± 0.03	0.52 ± 0.02	0.42 ± 0.02

The values measured on MIR-station were obtained from TLD-100. In general average LET is lower at the position with weaker shielding and higher behind stronger shielding. The values obtained by TLD-600 are higher because of the

evidence of thermal neutrons. This difference is more significant in the results from space shuttles. Due to the havier shielding of the space shuttle compared to space station MIR the contribution of secondary neutrons is higher. The value of average LET obtained during STS-60 mission was 10.6±0.6 keV/µm in tissue (TLD-600) and 6.5 ± 0.4 (TLD-700). The quality factor was 2.8 (TLD-600, overestimating thermal neutrons) and 1.8 (TLD-700, neglecting thermal neutrons). The quality factor obtained from NASA TEPC was 2.55. During STS-63 mission the contribution of secondary neutrons was less. The results from space shuttle missions STS-60 and STS-63 are resumed in table 2.
The errors given with the results are only statistical. The total error for dose measurements is assumed to be less than 5% and the error for LET measurement less than 10%.
For the evaluation of the quality factor the relation between LET and Q proposed in ICRP-26 was used because „average LET" is only meaningful using ICRP 26. Contrarely ICRP 60 demands assessment of the various components of the radiation field.

Table 2: Summary of results from dose measurements during space shuttle missions STS-60 and STS-63

Parameters	**STS-60**		**STS-63**	
Duration (d)	8.3		8.2	
	TLD-600	TLD-700	TLD-600	TLD-700
Dose -ATI (mGy tissue)	1.37 ± 0.09	1.24 ± 0.08	2.59 ± 0.17	2.49 ± 0.09
Doserate -ATI (mGy/d)	0.165 ± 0.012	0.149 ± 0.01	0.316 ± 0.021	0.303 ± 0.011
av. LET (keV/µm)/tissue	10.6 ± 0.6	6.5 ± 0.4	10.2 ± 0.4	8.8 ± 0.6
Quality Factor	**2.8**	**1.8**	**2.6**	**2.3**
Equiv. Dose (mSv)	3.84 ± 0.25	2.23 ± 0.15	6.73 ± 0.44	5.73 ± 0.21
Equiv. Doserate(mSv/d)	0.46 ± 0.03	0.27 ± 0.02	0.82 ± 0.05	0.70 ± 0.03

ANALYSES OF MEASURED DATA IN COMPARISON WITH DOSE CALCULATIONS

Measurements of absorbed dose have been routinely made on all space missions. A comparison of measured dose rates with results obtained by trapped radiation belt models (AP8) (5) have shown that dose is underestimated using AP8 min (calculation for solar minimum) and overestimated using AP8 max (solar maximum) by about 30%. The anisotropy of proton fluxes is a well understood phenomenon but is not included in the models up to now (6). Studies have also indicated that the most propable energy of protons predicted by the model differs from the measured energy by about 30 MeV (7).
For the STS-60 mission dose calculations for the positions of the dosemeters were carried out using the AP8min-model from IMBP and the NASA. Although the predicted doses on the one hand and the measured dosea on the other hand are in excellent agreement, there is an average ratio of the calculated doses to the measured doses of 1.8. Therfore there is a great international interest to improve the models of trapped belt radiation. Data gathered during these experiments may contribute to the improvement of calculation models. More detailed analyses of our data are in progress.

DISCUSSION

The method of LET-determination with TLDs on space station MIR was very successful. Absorbed dose rate and average LET respectively quality factor within the space station depend strongly from the mass distribution around the position. The small detectors allow the measurement of dose rate distribution as well as the position sensitve determination of average LET. This opens a wide field of applications. The results regarding average LET respectively quality factor are in excellent agreement with former measurements with LET spectrometers based on proportional counters (8). Data gathered during this experiment might be important to improve radiation models in low earth orbits.

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AIRCREW RADIATION EXPOSURE ASSESSMENT FOR YUGOSLAV AIRLINES

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INTRODUCTION

Cosmic-ray neutrons are produced by the interaction of galactic cosmic radiation with magnetic-field-bearing plasma cloud emitted from the sun (solar wind) and with the atmosphere. This neutron radiation varies with altitude, geomagnetic latitude and in different solar events. Commercial aircraft's have optimum cruising speed of 800 -900 km/h and the cruising altitude near 13 km. The flight paths are assigned according to airway corridors and safety requirements. The relatively high dose-equivalent rates at cruising altitudes near 13 km (about 1-2 μSv/h) cause exposures greater than 5 mSv/y (the maximum permissible dose for the general population) for a crew with full-time flight (500 - 600 h/y). The radiation exposure of the crew in commercial air traffic has been studied for the associations of the air crews and airline management and published, [1-5], and regulatory authorities are slowly accepting the fact that there indeed is a problem which needs investigations and protective regulation, [6-8].

The radiation exposure assessments of the crews in Yugoslav Airlines (JAT), are presented in this paper. Yugoslav Airlines (JAT) has the fleet of 30 aircraft's and 65 years of experience in commercial traffic (3.500.000 passengers per year and 35th on the IATA list before the war in Yugoslavia).

ANNUAL DOSE ASSESSMENT FOR YUGOSLAV AIRLINES CREWS

The flights of selected groups of crews (from each work category group of pilots and cabin crew members) have been studied during one year. All flights (about 17000) of selected 34 pilots and co-pilots and 41 cabin crew members have been analyzed. Each work category group of pilots and cabin crew members has mostly full-time block hours during 1989 (the last year of normal traffic for JAT). Selected pilots make four groups, for four aircraft types in use by JAT (DC-9, DC-10, B-727, B-737), and two groups of the co-pilots ("flight engineers" for B-727 and DC-10). Cabin crew members make three groups of pursers and two groups of STW/STD (they include both male and female workers).

Pilot (aircraft)	Total time of flights [h:min]	Annual effective dose [mSv]
1,2,3,4,	423:01 297:48 428:23 484:40	0,295 0,210 0,316 0,343
5 (DC-9)	324:29	0,316
6,7,8,9(B-727)	548:33 512:54 497:15 473:19	0,799 0,628 0,674 0,675
10,11,12,13(DC-10)	508:57 499:37 505:49 489:26	1,748 1,712 1,724 1,722
14,15,16,17(B-737)	467:49 507:51 440:20 419:28	0,523 0,448 0,464 0,392
Co-Pilot (aircraft)	**Total time of flights [h:min]**	**Annual effective dose [mSv]**
18, 19, 20, 21,	492:40 529:43 512:29 497:15	0,728 0,820 0,743 0,674
22, 23, 24(B-727)	508:32 489:02 467:15	0,716 0,667 0,688
25, 26, 27, 28,	25:08 366:04 428:57 447:24	0,260 1,221 1,467 1,540
29, 30, 31, 32,	360:39 452:28 387:10 416:51	1,247 1,562 1,326 1,421
33, 34 (DC-10)	461:14 430:20	1,528 1,491

Table 1. Aircraft types, total times of flight during one year and assessed annual effective dose for four selected groups of pilots and two groups of co-pilots ("flight engineers")

Model for calculations of the annual absorbed doses and effective doses uses data for annual cosmic-ray dosimetric profile in the lower atmosphere for solar minimum, taken from literature, [2]. The flight paths of aircraft's during various phases of the flight are approximated by different vertical profiles according to the

operational flight plans, with constant climbing and descending speed and the flights at constant altitude. The average times spent during climbs and descents were assumed using a set of the data for the flights.

Cabin crew (Pursers 35-57) (STW-STD 58-75)	Total time of flights [h:min]	Aircraft (in % of the total time of flight)					Annual effective dose [mSv]
		DC-9	DC-10	B-727	B-737	ATR-42	
35	420:37	5	95				1,38
36	563:33	2	82	6	10		1,77
37	487:23		66	34			1,38
38,39	458:58 452:33		100				1,66 1,57
40,41	449:19 163:57						1,60 0,57
42	397:57	10	87	3			1,25
43	505:11	11	56	27	6		1,25
44	429:44	10	74	5	11		1,25
45	511:21	3	67	26	4		1,37
46	603:11	3	57	39	1		1,51
47	286:15	22	47	28	3		0,58
48	166:54		84	16			0,54
49	284:58		76	24			0,88
50	545:16	1	56	43			1,40
51	379:10	25	69	5	1		1,03
52	474:25	1	55	43	1		1,18
53	485:01	34	62	3	1		1,22
54	633:34	42	48	5	4	1	1,28
55	492:09	5	60	35			1,27
56	525:12	4	59	4	17	16	1,25
57	385,16	1	62	37			0,99
58	510:30	2	63	1	34		1,27
59	350:10	26	5	69			0,47
60	348:02	1	11	3	67	18	0,41
61	501:19	1	77	2	20		1,48
62	507:55	1	58	4	19	18	1,19
63	430:09	27	70	1	2		1,14
64	449:23	20	78	2			1,33
65	470:39	1	52	38	2	7	1,16
66	424:09		71	24		5	1,14
67	521:45	2	4	92	2		0,77
68	131:12	3		9	88		0,13
69	243,19	2	56	41		1	0,60
70	526:05	9	78	12	1		1,55
71	473:17	1	53	5	21	20	1,07
72	461:57		69	1	15	15	1,20
73	449:20		73	8	8	11	1,23
74	348:43		56	2	23	19	0,79
75	295:53		74	1	18	7	0,85

Table 2. Aircraft types, total times of flight during one year and assessed annual effective doses for three selected groups of pursers ("cabin chiefs" 35-41, "check pursers" 42-49, "pursers" 50-57) and two selected groups of STW/STD ("seniors" 58-66, "STW/STD" 67-75)

About 17000 flights have been simulated and annual effective doses for each of analyzed pilots and cabin crew members have been assessed. The calculated annual effective doses for each selected pilot and co-pilot are presented in Table 1. and for each selected cabin crew member in Table 2.

The results show that the doses for the long-distance flights (DC-10 aircraft) are higher than the doses for domestic and continental flights. This demonstrate an assessment of the effective dose during the flight Belgrade - Beijing - Belgrade (1,22 mSv, [10]). Exposure of air crews can be up to 5 times higher than the average exposure of the ground based occupational workers. The assessed annual effective doses (1-2 mSv) are smaller than the maximum permissible dose for the general population (5 mSv), but they are comparable with the values for some radiation workers. The model for the calculations includes minimized influence of the cosmic neutrons. The newly reevaluation of health effects of the radiation (ICRP 1990.) changes proposed limits for the general public and professional radiation workers. Inflight radiation exposures of air crews can be higher then the newly proposed limits for the general public, [7]. The exposure rates during solar flares can rise up 3-4 times (many times in extremely rare cases), lasting for several hours up to several days, [7].

The calculated values for Yugoslav Airlines (JAT) are generally less than doses from similar studies, [7-9], because of the structure of the intercontinental flight (only two aircrafts DC-10 in use in 1989.), but the problem is interest for future studies according to the newly proposed limits of ICRP.

Assessed doses for cabin crew members have the variations for similar block hours. The reason for this is that cabin crew members function in various aircraft types.

Some discussions and comparisons with experimental results and the results from literature are also realized. The validity of the flight path approximation in calculations is analyzed, [10]. The comparison shows the small difference (1%-5%). An example for the flights Belgrade - Athens and Athens - Belgrade-Belgrade (two types of the aircraft) shows the difference less than 1% (the calculated effective dose according to the operational flight plan is 2, 157 μSv, and these value by using the approximation of the flight path is 2.039 μSv), [10]. Similar results gives the calculation for the flight Frankfurt - Belgrade (2, 645 and 2.624 μSv, [10]).

A dosimetric experiment has been realized during a routine flight (the flight Belgrade - Podgorica - Belgrade, February 24, 1994). Dosimeter 6150 AD 2 - Automess GmbH (D-6802 Ladenburg), S/N 57787 has been used. Dosimeter has measuring range 0,1 μSv/h -10 mSv/h, energy range of gamma radiation 60 keV - 3 MeV. The measured dosimetric profile (altitude vs. effective dose rate) is compared with the theoretical values, [2]. The results confirm the validity of the used model for calculations, [11].

The calculated values can have an uncertainty of about 20%: the cosmic rays vary with solar cycle (more than 10%) and geomagnetic latitude (about 2%) and the approximation of the real flight paths includes an additional uncertainty (1-5%).

CONCLUSIONS

The presented study shows that the crews of the intercontinental flights can receive significant annual effective doses (1,5 -2,0 mSv). The exposure of the crews is comparable with natural radiation level on the ground level (it can be up to 5 times higher for some air crew members in the intercontinental flights), but smaller than maximum permissible dose for general population. The annual exposures of the passengers are generally smaller than the exposures of the air crews, because the passengers have a limited number of flights per year compared with the members of the air crews.

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COMMERCIAL AIRLINE CREWS - A NEW GROUP OF OCCUPATIONAL RADIATION WORKERS

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Introduction

It has been recognised for a long time that aircrews who spend significant periods flying at high altitudes experience background radiation exposures considerably larger than those arising at ground level. The problem is most extreme in the case of astronauts, and in the U.S.A. the NASA has conducted or sponsored a number of well document studies of this problem. In the case of commercial airline operations, the exposures are most significant for the crews of supersonic aircraft which generally travel at much higher altitudes than regular flights, and such aircraft are provided with radiation alarm systems designed to indicate when background radiation levels temporarily become much higher than normal due to solar flares or other sources of abnormally high cosmic radiation. The pilots can then reduce altitude to regain additional protection from greater atmospheric shielding.

Aircrews on regular commercial flights are not generally regarded as occupationally exposed radiation workers. Although several studies have indicated that they may receive radiation doses in excess of the ICRP recommended limit for members of the public, the same is true of individuals living at ground level in one of the areas of abnormally high background gamma radiation; and in the past it has been general practice to avoid regarding either of these situations as coming within the radiation dose control practices of the various regulatory agencies. Furthermore it must be recognised that in areas prone to high background radon levels, even greater unregulated radiation exposures of many individuals take place. ICRP 65 recommends an action level for correcting high radon concentrations in dwellings which comes in the range corresponding to effective doses of between 3 and 10 mSv/y but experience has shown that few homeowners voluntarily carry out remedial work even at these levels. An aircrew member living in a home with abnormally high radon levels may well experience lower effective dose rates whilst in flight than whilst relaxing at home.

Dose Rates

The outdoor gamma dose rate at ground level in many areas is below 0.1 μSv/h, and of this only about half results from cosmic radiation passing through the atmosphere, the remainder results from gamma emitters in rocks and soil. In middle latitudes, at an altitude of about 4 km the dose rate from cosmic radiation increases to about 0.2 μSv/h, and above this height it increases steadily to an average value of about 12 μSv/h at the altitude at which a supersonic airliner would be likely to fly (16km). For most commercial jet aircraft the typical flight altitudes are about 12 km where dose rates of about 7 μSv/h are likely to be experienced. Transpolar routes involve significantly greater dose rates than those which take place at lower latitudes. It follows from this that a working year of 2,000 hours, mainly spent in flight, could be expected to lead to the typical crew member receiving a radiation dose of about 10 mSv. These estimates, based on measurements of cosmic ray intensities carried out at many centres, over many years, have been supported by several direct radiation dose measurements carried out in co-operation with airline operators, mainly in the last 5 years. For example in 1992 a study made at Tokyo University concluded that the annual doses received by typical flight crew personnel flying for about 900 hours per year were about 4 mSv; and a similar study made at Giessen University based on North Atlantic routes concluded that aircrew doses were higher then those actually received by nuclear plant operators in Germany. In 1990, at Oak Ridge, Fry carried out a theoretical study based on NASA satellite data and concluded that aircrew doses come somewhere between the recommended limits for occupationally exposed workers and those for members of the public. His data suggested that dose rates on trans-polar routes are about 2.5 times greater than those on low latitude flights.

Epidemiological studies have also been carried out. In 1989 the British Columbia Cancer Control Agency published a study of cancer incidence among 900 Canadian Airlines pilots. This showed statistically elevated incidences of lymphoma, Hodgkins disease and non-fatal skin cancers, but the numbers studied were not high enough for these results to be of high statistical significance and the study group were of course exposed to other

potential carcinogens in addition to cosmic radiation. Further studies are in progress and additional data is expected to gradually become available.

The nature of air transport is such that no abnormally high annual doses are likely to occur, as regulatory requirements designed to minimising the fatigue of aircrew members will also serve to limit their maximum radiation dose. Equally little can be done to further reduce such doses, except by planning flight schedules so that a reasonable proportion of the flight time of all aircrew personnel is spent operating short range services which generally take place at lower altitudes and therefore with lower dose rates. The questions that must be addressed are whether existing doses should be regarded as excessive and whether future developments in air transportation may introduce any factors which would cause them to increase. In the latter context it can be recognised that there has been an ongoing tendency for the average cruising altitudes of new aircraft to increase, partly in the interests of fuel efficiency. It may also be significant that, whilst high altitude ionizing radiation levels appear to be fairly consistent, UV levels are rising rapidly as a result of depletion of the ozone layer. It is important to ensure that the acrylic windscreens on commercial aircraft do not permit significant UV exposures of cockpit crews.

ICRP Recommendations

Since ICRP 60 recommended lowering the radiation dose limit for non-occupationally exposed members of the public to 1 mSv/y, the radiation exposure of aircrews has received a great deal more attention than in the past. It is clear that aircrew personnel may in the past have exceeded the old ICRP recommended dose limit of 5 mSv/y for members of the public, but when the recommended dose limits were higher it was easier to justify regarding aircrew doses as an example of high background doses which are not amenable to ready control. Recently this has become one of a number of areas in which attention is currently being directed towards more effective control of doses arising from various forms of what is commonly referred to as "technologically enhanced background radiation", and in the USA the FAA now recommends the classification of aircrews as occupationally exposed radiation workers. The increasing tendency for pregnant radiation workers to continue in employment until very close to their date of delivery is another important factor which has led to greater recognition of the need to reduce doses to all pregnant workers. A significant proportion of airline flightcrew personnel are women of childbearing age, and pregnancies among them arise frequently. The high altitude background cosmic ray dose to which these workers are exposed is not directly comparable with the corresponding background doses received at sea level as it will not consist exclusively of soft gamma radiation but will also include more penetrating meson and neutron components. The ratio of foetal dose to skin dose will therefore be greater than at ground level.

All these factors have led to a situation where aircrew personnel have themselves begun to recognise radiation exposures as a potential hazard of flight, and to demand that it receives appropriate attention from airline operators. The international aspect of the air transportation industry means that there would be likely to be intense pressure from all workers in the industry to see that any dose control procedures which might be introduced by one airline operating in one country, were quickly and generally adopted throughout the world. Nevertheless the complexity of establishing common standards in countries with very different legal codes and approaches to ionizing radiation control is very great. In the case of sea transport, much of the world's merchant shipping operates under "flags of convenience" which often simplify the regulatory requirements that individual operators would have to meet if their fleet was registered under the flag of countries with a more highly developed regulatory approach. It would be unfortunate if a similar situation was to develop among major airlines, and it is therefore very desirable that a uniform and carefully considered dose control policy should be introduced by international consensus and adopted by airline operators registered in all countries.

The Legislative Approach

It must also be recognised that regulatory control of the radiation doses received by aircrews may not be easily implemented under the legislative systems developed by some jurisdictions. These doses do not arise from either machine generated radiation or artificial or natural radioactive sources. They cannot even be classified as arising from technologically enhanced background radiation, they simply represent normal background radiation doses that have been received as a direct consequence of technological activities which have led to an increased potential for exposure to abnormally high background radiation levels. In both the U.S.A. and Canada the control

of sources of machine generated radiation is the responsibility of provincial or state authorities, control of doses received from artificially produced radionuclides is a national responsibility. The radiation doses received by aircrews do not readily fit into either of these categories. Occupational health regulations applicable to most Canadian workers are generated provincially and differ from one province to the next - a similar situation applies in the U.S.A. Nevertheless certain categories of workers are deemed to be employed nationally and for such workers national health and safety legislative standards apply. In Canada these categories include workers employed in national or international transportation activities (i.e. intra-provincial travel but not inter-provincial travel.) Such workers, which will include most of the staff attached to registered Canadian airlines, are subject to occupational health regulations developed by Labour Canada rather than to the corresponding provincial regulations. Until recently Labour Canada has had no regulations relating to radiation dose limits for workers, but last year proposed new regulations were introduced and these were published in Canada Gazette Part I on June 24th 1995. This is an intermediate stage in the introduction of legally binding regulations. New regulations generally become legally enforceable shortly after publication in Canada Gazette Part 1, unless they have aroused concerns which necessitate a further review. In this case it appears possible that some concerns about these proposed new Labour Canada regulations may lead to certain revisions taking place before they become legally enforceable. Should they be implemented as published, they would have a considerable impact on the operation of commercial airlines in Canada. Although the proposed new dose limits will not apply to the doses received by employees of organizations licenced by the federal Atomic Energy Control Board, they will be applicable to all federal workers whose employers do not require licencing for the use of radioactive sources. Employees of commercial airlines come into this category, and if these proposed new regulations are eventually introduced in their present form, they will require that the doses received by such workers are kept below both the current ICRP recommended dose limits and the dose limits currently enforced by the Atomic Energy Control Board for the employees of organizations licenced to use radioactive controlled substances. Since the proposed new regulations make no provision for registering certain employees as occupationally exposed workers, it appears implicit that the appropriate limits will be those applicable to members of the public rather than those applicable to registered atomic radiation workers; and it is clear from much of the data above that this could lead to considerable difficulties in the case of airline operators.

A Possible Role for IATA

Similar legal initiatives are likely to arise in other jurisdictions, and in some extreme cases there might appear to be advantages to be gained by commercial airline operators if they restructure their companies so that these are registered in countries where less restrictive legal requirements are in force. Ultimately there would be no advantage to be gained by either operators or crew members in developing a network of partially incompatible standards based on different jurisdictions and it appears desirable to develop an international approach to this problem which would avoid such problems arising. One way in which this could be done would be through an organization such as IATA which could establish a working group, incorporating representatives of both airline operators and crew members, that was charged with the responsibility for developing an agreed Code of Practice relating to dose monitoring procedures and, in circumstances where this might be necessary, procedures for modifying flight schedules to avoid occasional excessive doses to any crew members from arising. It would also be important for this Code of Practice to provide definitive guidance, based on authoritative medical opinion, on procedures for dose limitation in the case of pregnant employees. Once such a Code of Practice had been generally accepted by most major airlines and their employees, it could be enforced through IATA as a formal requirement for all IATA member organizations. In this way there would be a uniform standard of radiation protection for aircrews which would be enforced by airlines in all countries regardless of the legislative requirements of the country in which the company was registered. Once such a scheme was introduced there would be little point in individual countries introducing any legislation other than a regulation requiring all national carriers to adhere to the IATA standards.

It would also be easy for such a Code of Practice to be updated on an ongoing basis as additional scientific data became available. This is contrary to the situation which often occurs when updating formal legislative requirements in cases where the reasons for the proposed changes are highly technical and are not readily understood by members of the legislative body responsible for making the changes.

MEASUREMENTS OF COSMIC-RAY DOSES IN COMMERCIAL AIRLINE CABINS

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INTRODUCTION

Cosmic radiation doses which aircrew and air passengers receive in airplanes have been calling attention in many countries especially in the last decade. In this relation, various types of information had been reported on cosmic radiation intensity. In Japan, the cosmic radiation intensity had been measured in commercial airline cabins as well as chartered flights(1-3). While the intensity depends on altitude, geomagnetic latitude(or cutoff rigidity), and temporal variation of the solar activity, their doses are often speculated based on paper records on airflights combined with the intensty-altitude relationship(4). In this study, however, efforts were made to estimate more realistic integrated doses in airline cabins based on actual on-board measurements which had been conducted several dozens of times in each year(e.g., 45 times in 1994 and 27times in 1995).

MEASUREMENTS AND EMPIRICAL FORMULA

The measurements were done by the uses of several kinds of spherical scintillation spectrometers, an ionization chamber, neutron detectors and personal dosimeters. The main device was a scintillation spectrometer which accomodated a 3" ϕ NaI(Tl) spherical detector, and signal components above 3MeV were regarded as cosmic contribution. The neutron contribution was estimated based on intercomparison tests between an ionization chamber and Bonner sphere neutron detectors on board a DC-8 chartered flight(2-3). The results showed that the neutron dose is about 40% of observed charged particle doses with small deviations. The results also suggested that the integrated cosmic radiation dose(D) can be practically expressed in terms of flight time at the main flight altitude(T) plus 10 minutes, dose rate factor(R) and correction factors as follows:

$$D = [R(T + 10)/60] \times F_g \times V_t$$

where F_g is a correction factor due to geomagnetic latitude(F_g =1.1 - 0.95), and V_t is another correction factor due to solar activity(V_t =1.15 - 0.85). R is the dose rate as a function of altitude as presented in Table 1. The uncertainty of obtained results was estimated to be less than 5%.

Table 1. Cosmic radiation dose rates at various altitude.

Altitude(kilo feet)	19	22	25	28	31	33	35	37	39
Dose rate R(μ Sv/h)	0.38	0.57	0.83	1.12	1.42	1.62	1.83	2.03	2.23

ESTIMATED DOSE RATES

In case realistic integrated doses should be evaluated, attention must be paid to flight direction, too. That is because airflight time is considerably influenced by wind such as the westerly. Such was studied in east-west flights(Tokyo-Matsuyama), north-south flights (Tokyo-Hokkaido as well as Tokyo-Okinawa) along with other short distance flights. The mean doses and their ranges were 1 μ Sv(0.7-1.5 μ Sv), 1.5 μ Sv(1.0-1.8 μ Sv) , 2.5 μ Sv(2-3 μ Sv), and negligible(less than 0.5 μ Sv), respectively. Figure 1 compares the integrated doses between eastward(from Tokyo to Matsuyama: against the westerly) and westward(from Matsuyama to Tokyo: with the westerly) flights. It is clearly demonstrated that the integrated dose depend on both flight altitude and flight direction.

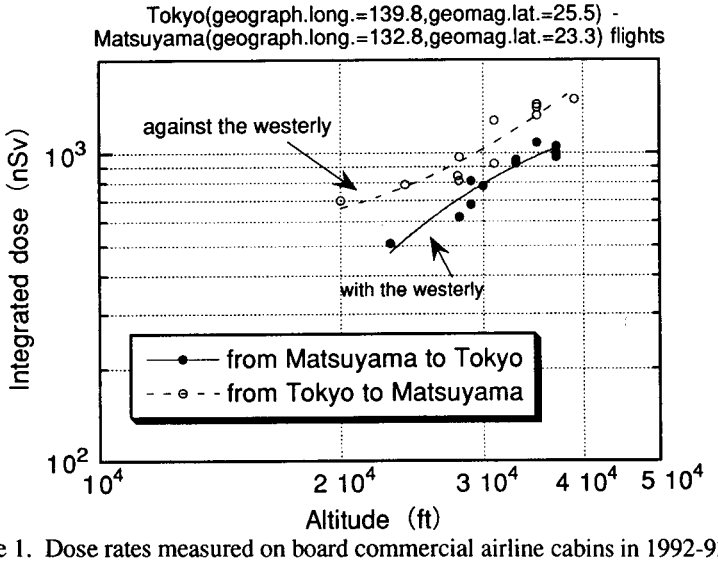

Figure 1. Dose rates measured on board commercial airline cabins in 1992-93.

As for domestic flights in Japan, it was obtained that a typical integrated dose in short flights (Tokyo-Osaka, Osaka-Kyusyu, etc.), middle flights(Tokyo-Hokkaido, Tokyo-Kyusyu, etc) and long flights(Tokyo-Okinawa, Hokkaido-Okinawa) were $\sim$0.5 μ Sv, $\sim$1.5 μ Sv and $\sim$ 3 μ Sv, respectively. International flights were also studied, and comparison between Japan-Europe routes, one via Anchorage and another via Singapore, was possible. Generally, the dose level estimated here are lower than other reported ones. Figure 2 compares altitude dependence of the dose equivalent rates in NCRP report and that of actually measured ones.

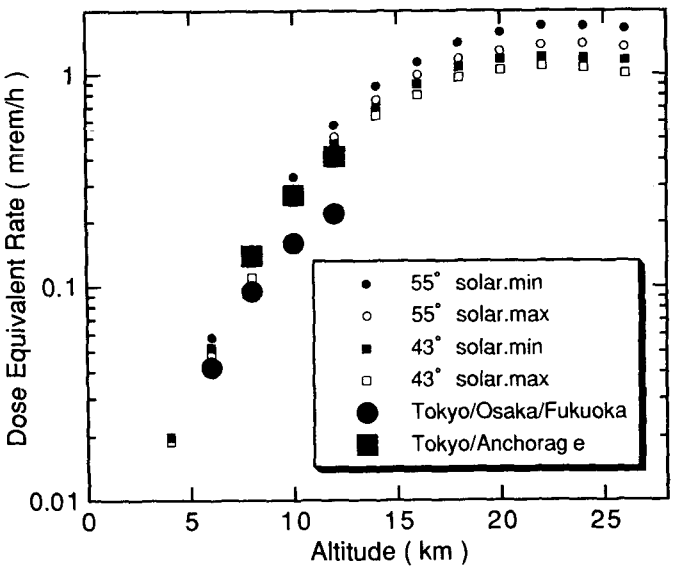

Figure 2. Dose equivalent rate in old unit at 5 cm depth in a 30 cm slab of tissue. NCRP values for Solar maximum and solar minimum are compared to 1000 km flights within Japan along with polar flight which passed Anchorage in Alaska.

CONCLUSION

To discuss the dose limit for aircrew and passengers, care must be paid not only to boarding hours but also to flight routes which reflect many other factors. As present data are based on numbers of actual commercial flights, it will contribute to find proper countermeasures to avoid over exposure.

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INVESTIGATION OF COSMIC RAY INTENSITY AT SEA LEVEL BETWEEN TOKYO BAY AND INDIAN OCEAN

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INTRODUCTION

In the Earth's atmosphere, the distribution of cosmic ray intensity changes as a function of both altitude and geomagnetic latitude. To estimate the human dose from cosmic rays, it is necessary to take into consideration these geographic factors. The purpose of this study is to investigate the actual distribution of cosmic ray intensity by *in situ* measurements on a global scale.

MEASUREMENTS

A wide area survey, which aimed at the estimation of cosmic ray intensity at sea level in Asia, was conducted during the KH93-3 Indian Ocean Research Cruise of R/V Hakuho-maru of the Ocean Research Institute, the University of Tokyo[1]. Cosmic rays were measured with a multi-channel spectrometer with a 3" ϕ spherical NaI(Tl) scintillation detector[2]. Geomagnetic field was also measured by a shipboard three component magnetometer (STCM)[3] simultaneously to examine the change of cosmic ray intensity. The measurements were carried out on the upper deck of the vessel from July 8, 1993 in Tokyo Bay to August 10, 1993 in Mauritius, southwestern part of Indian Ocean, by way of Singapore and Sunda Strait (Fig. 1).

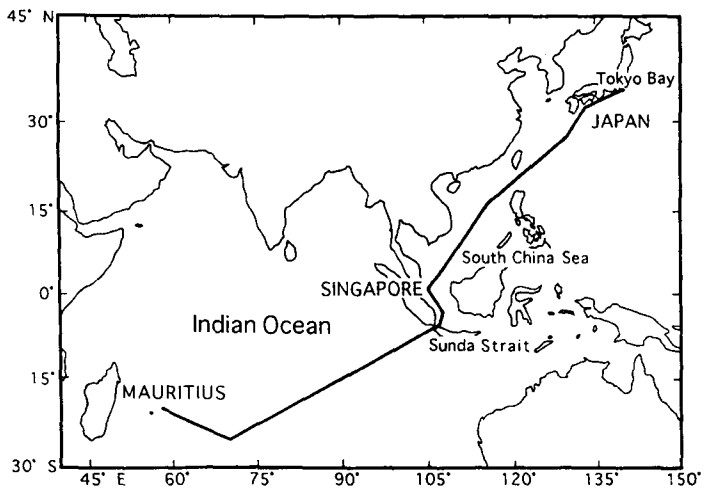

Figure 1. Index map of the survey area. Solid line shows the ship's track.

RESULTS

Analysis was made for the energy spectral region above 3 MeV by the peeling-off method, and the cosmic ray intensity was calculated as the absorbed dose rate in air from the ionizing component of cosmic rays. Then the atmospheric pressure effect was subtracted from the calculated intensity. Figure 2 shows a variation of the cosmic ray intensity with the geomagnetic latitude derived from the International Geomagnetic Reference Field (IGRF)[4]. Figure 3 also shows a relationship between the cosmic ray intensity and the horizontal component of geomagnetic field measured by the STCM.

The cosmic ray intensity decreased from about 30 to 24 nGy/h with the decrease of the geomagnetic latitude (Fig. 2), and also decreased with the increase of the horizontal component of geomagnetic field (Fig. 3). The minimum value of the cosmic ray intensity was observed at a geographic latitude of 8.5° N in South China Sea (Fig. 1) where the horizontal component of geomagnetic field was the largest, about 40,900 nT. This geographic latitude almost agrees with the geomagnetic equator derived from the IGRF. Based on Figure 2, the latitude effect in this study area, a rate of change of the cosmic ray intensity with the geomagnetic latitude, was estimated to be about 0.15 nGy/h/degree on the average.

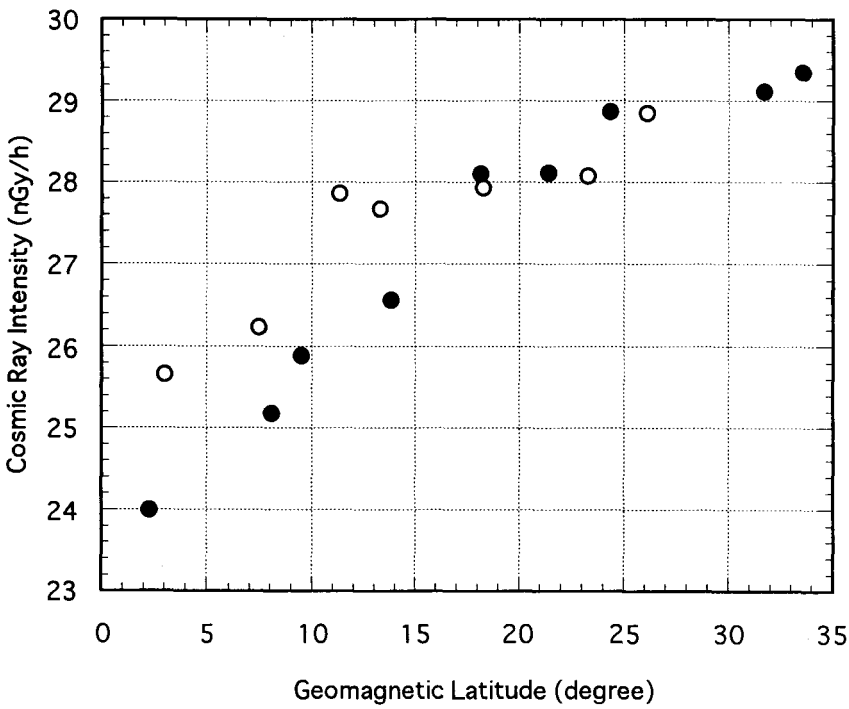

Figure 2. Variation of the cosmic ray intensity at sea level between Tokyo Bay and Indian Ocean with the geomagnetic latitude. Open and black circles show the data measured in the north and the south geomagnetic latitude, respectively.

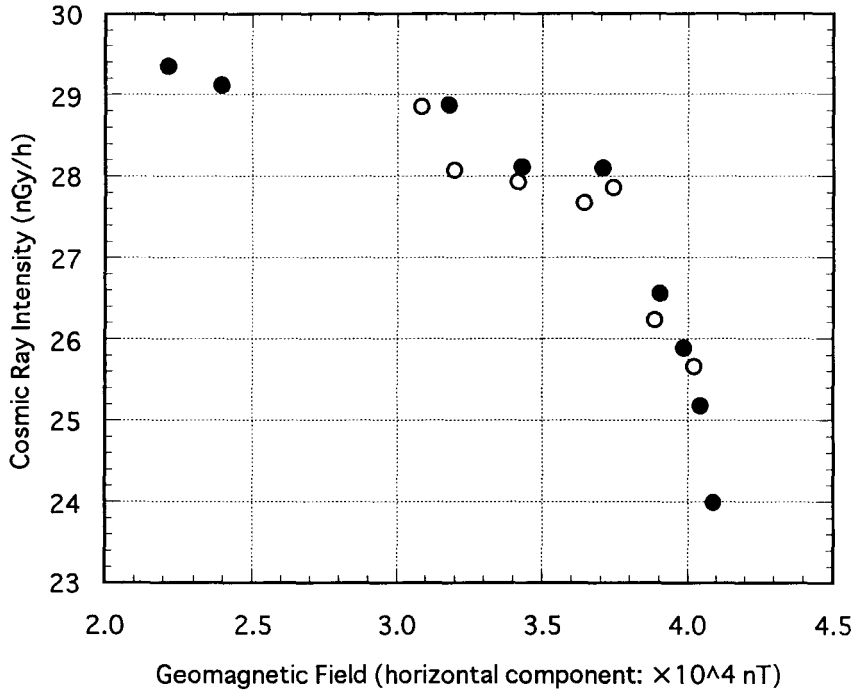

Figure 3. Relationship between the cosmic ray intensity and the horizontal component of geomagnetic field at sea level. Open and black circles show the data measured in the north and the south geomagnetic latitude, respectively.

SUMMARY AND CONCLUSIONS

This study provides the first direct measurement of the relationship between cosmic ray intensity and geomagnetic field at sea level by means of vessel. From the results (Figs. 2 and 3), the geomagnetic field is considered to be a good indicator of the cosmic ray intensity. Using the relationship, we can estimate the cosmic ray intensity at sea level on a global scale. In the same manner, the above observations are useful when we examine the human dose from cosmic rays and its effect especially in the populous subtropical and tropical regions around the studied area.

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MEASUREMENT OF COSMIC-RAY NEUTRON DOSES IN TAIWAN

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INTRODUCTION

In the natural radiation background, neutron dose is far much lower than gamma-ray dose generally. When one talks about external dose of natural background, only gamma-ray dose is considered or monitored and neutron dose is normally neglected. According to UNSCEAR 1993 Report (1), the average effective dose rate from cosmic-ray neutrons at sea level is estimated to be 3.6 nSv h^{-1} which is equivalent to 30 μSv per year. This value is comparable with the 50 μSv per year dose limit imposed to the site boundary of individual nuclear facility by the regulatory authority in Taiwan. For nuclear power plants, spent fuel interim storage facilities, and high energy accelerator facilities from which both gamma rays and neutrons emanate, the radiation dose at site boundary comes, therefore, not only from skyshine gamma rays but also from stray neutrons scattered from the facility buildings and from the air. The detection of cosmic-ray neutrons has become important and necessary in order to inspect the compliance of nuclear facilities with the dose limit imposed to the site boundary. The composition and intensity of the cosmic radiation change with time and location due to, among others, mainly the following factors (2) : solar-cycle modulation effect, solar flare effect, latitude effect, altitude effect, and temperature effect. Because this change with time and location is more prominent for the neutron component than for the ionized component, the need for long-term monitoring of cosmic-ray neutrons is clear. In this paper a measurement system consisting of large-size BF3 counters with polyethylene enclosure of different radii was set up. The response functions of the BF3 counters were calculated by using the one-dimensional transport code ANISN (3) performed in adjoin mode. A preliminary survey measurement of the cosmic-ray neutron intensity on the plain and on the mountains in Taiwan area was performed.

EXPERIMENTAL SET UP

In this study high sensitivity large-size cylindrical BF3 counters with polyethylene (PE) enclosure of different radii , namely , 5, 8.5, 10 cm and bare BF3 counters were used to measurement of cosmic-ray neutrons. The BF3 counter has a dimensions of 1.5" in diameter and 34" in active length and is filled with BF3 gas in pressure of 45-cm Hg. The enrichment of ^{10}B is 95%. Figure 1 shows the block diagram of the experimental set up. The 4-way fan out was used

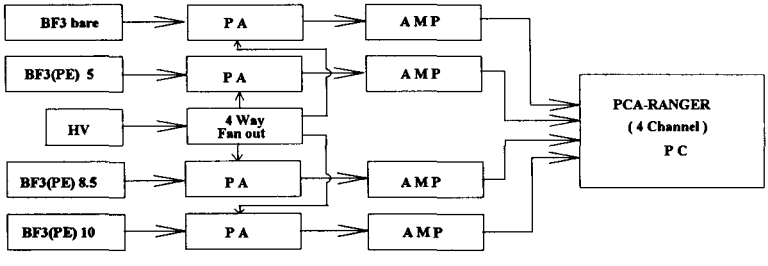

Figure 1. The block diagram of the experimental set up.

to distribute the high voltage to each of the four detectors. The amplifier signals from each detectors were fed into a PCA-Ranger card plugged in a personal computer (PC). The data were

acquired every 10 minutes and showed on the hard disk of the PC.

The response functions of the cylindrical BF3 counters with PE enclosures of different radii and bare BF3 counter were calculated by using ANISN code performed in adjoin mode with BUGLE cross section library (4) for energies below 17 MeV and HILO cross section library (5) for energies above 17 MeV. The detector calibration was made in a neutron calibration room with a normal 100 μg californium-252 source.

RESULTS AND DISCUSSION

In order to perform field measurement of cosmic-ray neutrons, a portable diesel generator for power supply and a tent for rain protection were carried with the neutron detection system. First of all, the measurement was made on the plain in National Tsing Hua University and in Synchrotron Radiation Center. Trips were then made to the Ali Mountain and the Hohuan Mountain in summer 1995. Measurements were made on the way at some representative locations with different altitudes. Table 1 shows the altitudes and the measured counting rates for the four BF3 counters at each measurement locations. The measurements at each locations look about one hour in order to make sure the stable operation of the counting system and the counting rates at each locations were the average values over the whole measurement period. Figure 2 shows the plot of measured counting rates of four BF3 counters as a function of altitude. It can be seen from Fig. 2 that the counting rate increases steadily with the altitude and the ratios of counting rates among the four BF3 counters keep more or less constant at each measurement locations, irrespective of the altitude and the BF3 counter with 5-cm polyethylene enclosure has the highest counting rates. Therefore, it seems reasonable to assume that the cosmic-ray neutron spectrum remains the same independent of the altitude. In our previous study (6) , the annual effective dose of cosmic-ray neutrons on the plain was estimated to be 14 μSv by using the BF3 counter with 8.5-cm polyethylene enclosure. When changes to radiation weighting factor for neutrons recommended by the ICRP in 1991 (7) were taken into account, it will lead to an increase of 50 %. of the effective dose rate of cosmic-ray neutrons. therefore, the effective dose rates of cosmic-ray neutrons on the plain was corrected to be 21 μSv /year. If the cosmic-ray neutron spectrum remains the same, as being a reasonable assumption, the effective dose rate will be proportional to the counting rate of the BF3 counter with 8.5-cm polyethylene enclosure. The effective dose rates at other measurement locations could then be estimated accordingly as include in Table 1.

Table 1 . Measured counting rates (counts / 10 min.) of BF3 counters and effective dose rates (μSv/ year) as a function of altitude.

Altitude (meter)	BF_3 bare	BF_3(PE) 8.5 cm	BF_3(PE) 5 cm	BF_3(PE) 10 cm	Effective dose rate
3275 (Mount Wulin)	120.2 (5.9)*	304.0 (12.9)	395.4 (10.7)	268.6 (9.2)	152
2500 (Mount Ali)	84.5 (5.2)	200.3 (9.0)	268.0 (21.5)	159.8 (7.9)	100
1800	58.0 (5.2)	124.3 (3.5)	157.0 (8.9)	99.0 (11.5)	62
1300	47.3 (8.2)	104.5 (14.2)	133.3 (2.2)	80.8 (9.0)	52
1148	40.7 (5.1)	87.8 (8.7)	123.7 (8.2)	75.3 (7.6)	44
500	29.8 (4.9)	61.3 (7.7)	71.5 (8.6)	45.8 (7.6)	33
0	25.3 (7.4)	42 (3.6)	58.5 (8.0)	35.3 (3.6)	21

* standard deviation

Figure 3 shows the plot of the effective dose rate of cosmic-ray neutrons as a function of altitude. The diamond marks show the data of this work. The smooth curve comes from analytical expressions developed in reference (8) and cited in UNSCEAR 1993 Report (1). It can be seen from Fig. 3 that the increase of cosmic-ray intensity with altitude indicated in our measured data agrees very well that cited in UNSCEAR 1993 Report although our measured effective dose rates are about 50% lower, which is clearly due to the lower latitude of the measurement locations (23.5°N). It is also interested to note from Table 1 and Fig. 3 that the effective dose rates of cosmic-ray neutrons at Mount Ali (2500 m above sea level) and at Mount Wulin (3275 m above sea level)were about five times and 7.2 times higher than that on the plain, respectively.

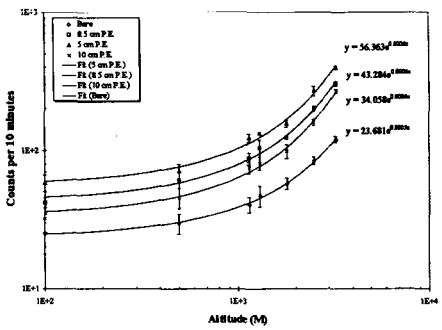

Fig.2 Measured counting rates of BF_3 counters as a function of altitude

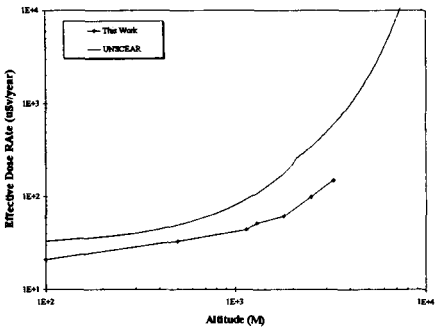

Fig.3 Effective dose rate of cosmic-ray neutrons as a function of altitude

CONCLUSIONS

In this paper a high sensitive neutron detection system has been successfully established. By using this detection system it is the first time the effective dose rates of cosmic-ray neutrons as a function of the altitude in Taiwan area has been measured. On the plain the effective dose rate was estimated to be about 21 μSv /year. On Mount Ali (2500 m above sea level) and Mount Wulin (3275 m above sea level) the effective dose rates were found to be about five times and 7.2 times higher than that on the plain. The measured variation of the effective dose rate as a function of altitude agreed very well with that cited in UNSCEAR 1993 Report. Since cosmic-ray neutron intensity changes with time and space, it is strongly suggested that the extensive survey and long-term monitoring should be taken as soon as possible.

ACKNOWLEDGMENT

This work was supported by the National Science Council, Republic of China, under the contract number NSC 84-2212-E-007-023 .

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NATURAL NEUTRON EQUIVALENT DOSE IN MIDDLE EUROPE REGION.

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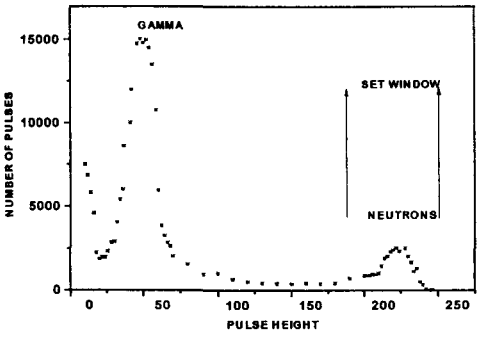
INTRODUCTION. Determination of the neutron fluence rate dependence on the elevation at small distances above the earth's surface is important both for estimation of the neutron radiation contribution to the whole population dose and from point of view of the influence of neutron radiation on detector back-ground as well. We carried out such the measurements at localities with various elevation from 113 m to 2632 m, placed between geodetic latitudes 48° -52° N and longitudes 16° - 20° E (Czech and Slovak Republic).

The spectrum of the ambient neutrons were determined using a calibrated Bonner spectrometer placed on the site with the highest elevation (Lopmnitsky Peak in The High Tatras, 2632 m). A ^{3}He-proportional counter with a sensitive volume of $1*10^4$ cm^3 was the second used detector [1]. The amplitude spectrum of the ambient neutrons registered by ^{3}He-detector is shown on Fig.1.The significant peak generated by thermal part of ambient neutrons (reaction He[n,p]T) was observed. We set the window of a single channel analyzer on this peak The back-ground counting rate (in the set window from interior surface alpha activity was N=(0.061±0.0002) s^{-1} The effective detection area (or sensitivity) C of the counter, determined according

Fig.1



$$N[s^{-1}] = C[m^2].\Phi\ [m^{-2}\ s^{-1}] \quad /1/$$

depends on the shape of the neutron spectra. For thermal neutrons it amounted to C_{th} =(0.015±0.0013) m^2. This value exceeds by approximately 600 times the sensitivity of a LiJ(Eu) crystal with dimensions of 4 * 8 mm.

The thermal part of natural neutrons spectrum Φ_{th} was detected directly by the proportional counter. In order to detect possible changes in the intermediate Φ_{in} and fast Φ_f components ambient neutrons at different sites we also used the ^{3}He-counter which was wrapped in layers of polyethylene (PE) moderators, 23 mm or 64 mm thick, lined (on the outside) by a thermal neutron absorber of Cd 0.8mm thick, for to shift the maximum of a response function into the intermediate or the fast region neutron energy.

Measurements on the other sites were carried out with only a ^{3}He-counter. We supposed that the shape of neutron spectrum, is elevation independent at law altitudes. This assumption is based on theoretical models of Lal [2], O'BRIEN [3] and MASARIK and REEDY [4] calculation.. Physical foundations for it is the existence of an equilibrium distribution of low energy nucleons (E < 500 MeV) through out the low altitudes region in the atmosphere.

RESULTS AND EVOLUTION OF MEASUREMENTS

The results of measurements are presented on Fig.2 (thermal part of natural neutrons) and Fig.3. (when were registered mainly intermediate or fast neutrons).The measured counting rates were corrected for the intrinsic background of the counter and they exhibit an exponential dependence on the elevation of the site **z**. Consequently the neutron fluence rate and dose equivalent rate can be approximated by the exponential function in the form ~ $\exp(\alpha * z)$. From our experimental data $\alpha = (0.85 \pm 0.05)\ 10^{-3}\ m^{-1}$.

For the neutron spectrum evaluation the SAND II. code has been applied. For the response functions of Bonner-Spheres the published data calculated by Hertel were used [5]. The neutron fluence rate were calculated from neutron spectra.

Fig.2

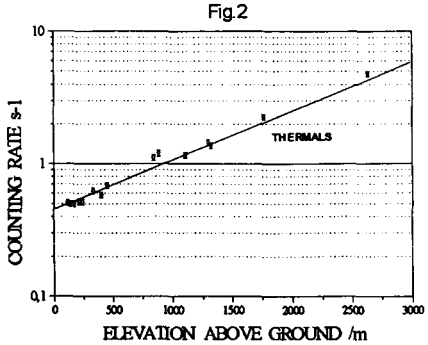
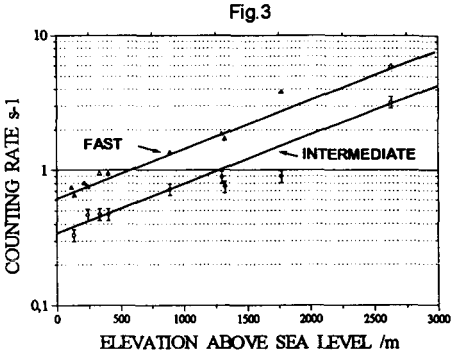


Fig.3



The dose equivalent rate were calculated from the neutron spectra. All data given in the publication have calculated before the ICRP 60 Recommendation has been published. In this way we got the value 40 nSv/h for elevation 2632 m . Partial and total fluence rate ϕ on this site determinated by Bonner spectrometer are on theTab.1 The extrapolation of the height dependence counting rate to zero altitude gives the values: $\Phi_{th}(0)=(30\pm6)m^{-2}*s^{-1}$; $\Phi_t(0) = (100\pm20)\ m^{-2}*s^{-1}$ .and H(0) = (4.1 ± 1.2) nSv/h.

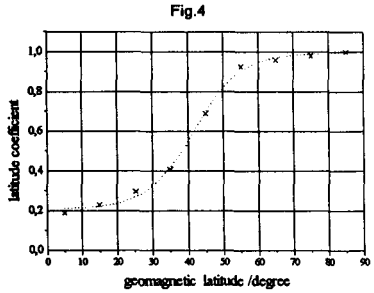
Energy group	$\phi[m^{-2}*s^{-1}]$
Φ_{th} ($E_n < 0.5$ eV)	280
Φ_{in} (0.5 eV ≥ E_n > 0.1 MeV)	330
Φ_f ($E_n \geq 0.1$ MeV)	340
Φ_t	950

DISCUSSION

Our measurement were carried out at varies localities with different relative concentrations of clay, sand and stone in the composition of ground. The statistical errors of our measurement were less then 5 %. As can be seen in Fig.2-3, the results confirm that the elevation dependence of neutron fluence rate and also dose equivalent can be with reasonable accuracy approximated with the functions ~ exp(α*z) , where α = 0.85 km^{-1}. Deformation of the neutron spectrum occurs with the increase of ground moisture content. Such deformation of the neutron spectrum lowers the annual dose equivalent. Because our measurements were done in the summer season during a long-lasting period without precipitation, the value of H(z) obtained should be considered as a maximum one.

In order to find the dependence of the mean value of the annual dose equivalent <H(z)> on relevant elevation, both <H(0)> and parameter α have to be decreased, because precipitation increases with increasing elevation.This fact is reflected in analytical expressions have been developed for the general relationship between annual dose and elevation for the neutron component of cosmic ray, which are given by BOUVILLE and LOWDER [6] and recommended by Forty-first session of UNSCEAR (1992) for estimate the distribution of collective effective dose with elevation. There are two expressions [6]: one for elevation z< 2 km, in the form H(z)=H(0)*exp(1.04*z), and second in the form H(z)=H(0)* [1.98*exp(0.7*z)] for elevation z > 2 km, where H(0)=20 μSv/a = 2.4 nSv/h. These equations may by applied to estimate doses from natural neutrons at habitable elevations around the world. Because there is strong dependence of neutron fluence rate on latitude, our experimental data valid for latitude ~50° N distinguish (Fig..4) from latitude-averaged value[6]. Comparing our value α = 0.85 km^{-1} with value 0.7 km^{-1} [6] we made conclusion, that seasonal variations should be lead to decreasing α maximum 20%. Unfortunately, the accuracy of experimental data is not satisfactory for making the unambiguously conclusion.

Fig.4



LCS was used also for estimation of latitude dependence of neutron fluence rate that is expressed through coefficient k_φ. Its values for sea level altitude are presented in Fig.4. Our measurements were carried out at near geomagnetic latitudes ~50°, that are characterized with k_φ =0.84 (values are normalized to unit for high latitudes) and we estimate latitude-

averaged value assign weights to Earth'area for different latitudes with step of 10°. We get the same value 2.4 nSv/h, that was used by BOUVILLE and LOWDER [6].

We can normalized the values in the front of the exponential factor (in accordance with the value for geomagnetic latitude φ=90°) as latitude-independent parts, that is multiply by factor k_φ . Then dependence of total, thermal neutron fluence rate and dose equivalent rate on elevation z and latitude φ can be parametrized

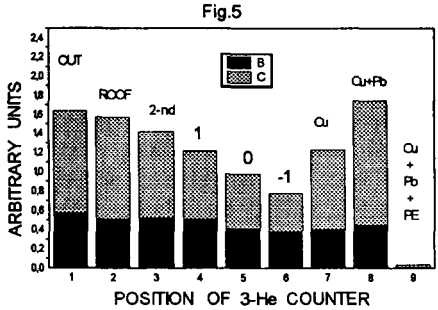
$$\phi_t (z,\varphi) = 119*k_\varphi*\exp(\alpha*z) \qquad /2/$$

$$\phi_{th} (z,\varphi) = 35* k_\varphi*\exp(\alpha*z) \qquad /3/$$

$$H (z,\varphi) = 4.8*k_\varphi*\exp(\alpha*z) \qquad /4/$$

Knowledge of elevation and latitude dependence thermal component of natural neutron fluence is important for determination of high-sensitive neutron detector backgrounds, because it is determined mainly by this energy group. Effective area of neutron detector C_{th} can be easily calculated or experimental determination using /1/, where ϕ_{th} is calculated from /3/ for a given locality. In the case of experimental determination, detector have to be located in the open space, far enough from massive constructions, buildings.etc. As an illustration Fig.5 shows the influence of the construction materials on the natural fluence rate (B - ϕ_{th} , C - ϕ_f + ϕ_{in}) on the locality outside the building of the Department of Nuclear Physics (position 1), on the roof (2) and the different floors (3,4). Into the basement there is a low background massive shielding , position (5) - the counter placed into the shielding, when it was made of copper only, next position - the shielding was made of copper and lead and last - the shielding was made of copper, lead and polyethylene (PE) with boron.

Fig.5



CONCLUSION

Results of our measurements show, that change in dose equivalent rate [nSv/h] from natural neutrons for the Middle Europe region can be described with reasonable accuracy exponential dependence H(z)=4.1* exp(0.85*z) valid for elevation z = 0 - 2.6 km, having an error about 30 %, where the errors of calibration (20 %), the statistical errors (5 %) and the error (20 %) caused the variation of moisture in ground in during year are included. The calculated by us the values k_φ quote to the good agreement with the latitude-averaged annual value <H(0) > , but they need experimental tests for localities with different latitudes.

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BE-7 CONCENTRATIONS IN GARDEN AND WILD VEGETABLES IN JAPAN

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INTRODUCTION

Beryllium-7 is a natural radionuclide produced by cosmic rays(1). Be-7 is found vividly in the atmosphere, so the concentrations in airborne particles were analyzed by many investigators(2-3). It is known that airborne particles with Be-7 in the atmosphere fall slowly to the ground and adhere to plant surface. However, Be-7 concentrations of foods were not measured too much. So we measured Be-7 concentrations of garden vegetables, wild vegetables, grasses, beef and milk, and calculated internal exposure dose from Be-7.

EXPERIMENTAL

Eleven species of garden vegetables (carrot, Japanese radish, cabbage, Chinese cabbage, spinach, cucumber, broccoli, etc.) and three species of grasses were collected from a farm in Towada city of Aomori Prefecture, and ten species of wild vegetables (*Anemone flaccida, Aralia cordata, Aralia elata, Cacalia hastata, Cardiocrinum cordatum var glehnii, Laportea macrostachya, Matteuccia Struthiopteris, Petasites japonicus, Pilea hamaoi, Pteridium aquilinum*) were gathered in a forest area of Mt. Hakkoda of Aomori. Beef and milk samples produced in Aomori were purchased at a store. As beryllium is scarcely scattered by heating, these samples were burned out to dry ash at 400 ℃ for 30hours using an electric furnace. The preprocessed samples were packed compactly into plastic vials, and counted for 24 hours using a Canberra 30% HPGe detector coupled a Canberra Series 35plus MCA. The surface areas of leaf were calculated by counting pixel numbers of picture using computer. Internal exposure doses from Be-7 in foods were obtained by S-value of MIRD Pamphlet(5) and the metabolic data of ICRP(6).

RESULTS AND DISCUSSION

Beryllium-7 concentrations of plant samples are shown in Table 1. Beryllium-7 concentrations of garden vegetables were from 0.2 to 25.3 Bq/kg, and concentrations of wild vegetables were from 0.8 to 23.5 Bq/kg. There is no difference in Be-7 concentrations between garden vegetables and wild vegetables. Leaf vegetables have almost higher concentration of Be-7. Though *Matteuccia Struthiopteris* and *Pteridium aquilinum* are ferns, their eatable stages are sprouts.

Figure 1 shows relationship between surface area and Be-7 concentration in some vegetable leaves gathered simultaneously at the same farm. Beryllium-7 concentrations of leaves correlate significantly with the surface area/weight ratios. What high concentration vegetables have proportionately broad leaves suggests that atmospheric Be-7 particles adhere to surface of leaves.

As grasses have the higher surface area/weight ratios, their Be-7 concentrations (especially timothy) were very high. Figure 2 shows relation between ablution

frequency and Be-7 concentrations of leaves when these grasses were washed by a synthetic detergent. Beryllium-7 concentrations of grasses fell markedly after first washing. However, these concentrations were not decreased after second, third and fourth washing respectively. This fact was found similarly in other vegetables. A binding portion of Be-7 should be on the leaf or in the leaf. Beryllium-7 concentrations of root vegetables and cucumber were very low level, and concentrations Be-7 in peeled vegetables were still lower. Consequently, it is supposed that the plants have little ability to absorb beryllium from roots.

There are probably two components of Be-7 that loosely adhere to the outside of vegetables and firmly cling to that.

Concentrations of Be-7 in beef and milk were all under limit of detection. As the concentrations of Be-7 in grasses were high, it was guessed that transfer coefficient of Be-7 to beef and milk from feed is very low. Therefore it is thought that contribution to internal exposure from Be-7 in beef and milk is negligible.

Table 1. Be-7 concentrations in vegetables.

Garden vegetables	
Broccoli	1.1
Cabbage	8.7
Carrot (leaf)	25.3
Carrot (root)	0.4
Chinese cabbage	11.2
Cucumber	0.2
Japanese radish (leaf)	10.4
Japanese radish (root)	0.2
Spinach	5.0
Wild vegetables	
Anemone flaccida	20.9
Aralia cordata	1.0
Aralia elata	4.1
Cacalia hastata	13.4
Cardiocrinum cordatum var glehnii	2.2
Laportea macrostachya	7.8
Matteuccia Struthiopteris	1.7
Petasites japonicus	23.5
Pilea hamaoi	9.2
Pteridium aquilinum	0.8

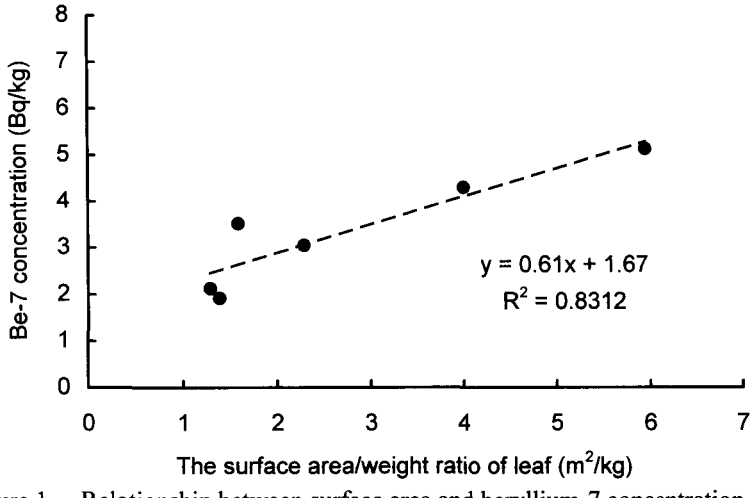

Figure 1. Relationship between surface area and beryllium-7 concentration in some vegetable leaves.

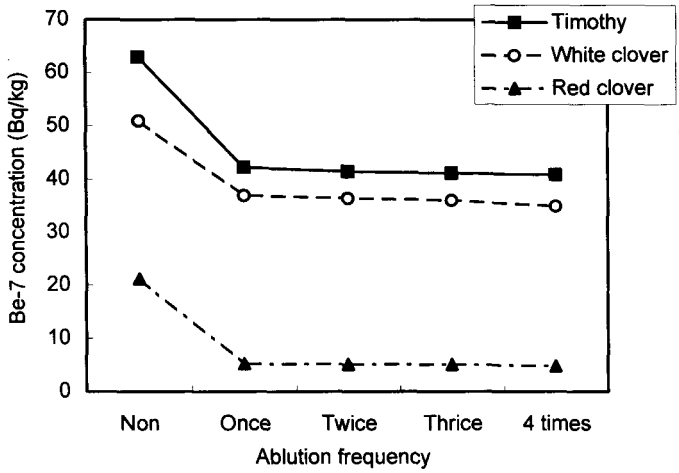

Figure 2. Deciduation of beryllium-7 from leaves by washing.

Beryllium-7 is taken in the body via the respiratory organ and gastrointestinal system, and contributes to internal exposure. Though there are very little reports on internal exposure from Be-7 in food, NCRP reported on the exposure dose from Be-7 in leaf vegetables as 8μrad/yr(80nGy/yr) and on the dose conversion factor to exposure as 2.7×10^{-9} mrad/pCi intake (0.73pGy/Bq).

We tried to evaluate the internal exposure from Be-7 in vegetables above-mentioned. Hypothetical Japanese annual intake amounts of leaf vegetables and wild vegetables are 35 kg/yr and 3 kg/yr respectively. The absorption coefficient of GI tract f_1 is 0.006, and absorbed Be-7 deposit in bone (6). The retention periods of stomach, small intestine, upper large intestine and lower large intestine were used for calculation as 1, 4, 13 and 24 hours respectively. Finally we obtained the dose conversion factor from the S value (5) as 7.8 pSv/Bq. We used the higher concentration values (leaves of carrot and *Petasites japonicus*) for calculation, then we obtained the internal exposure dose 7.5 nSv/yr.

Though the dose conversion factor in this study was ten times higher than NCRP report, the evaluated internal exposure dose was ten times lower than NCRP. If the dose conversion factor of NCRP, our concentration values and the annual intake amounts are used, calculated internal exposure dose is a hundred times lower than NCRP.

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Long-Term Variation of Atmospheric Beryllium-7 in Taiwan

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ABSTRACT

Taiwan Radiation Monitoring Center (TRMC) has measured the concentration of atmospheric ^{7}Be at the four stations on Taiwan for almost twenty years. Gamma emitting nuclides have been collected by air samplers and measured by gamma spectrometry. Cosmogenic ^{7}Be and other man-made radionuclides produced by Chinese nuclear weapon testings have been detected. Compared with ^{7}Be, man-made nuclides vanished quickly because of their short half life. The behavior of ^{7}Be in long-term concentration variation is different between northern and southern Taiwan because of local meteorological conditions. The results of Fourier analysis show that washout effect of rainfalls and transportation from stratosphere to troposphere are closely related to the variation of ^{7}Be concentration in near surface air. All the measured data of ^{7}Be show that the characteristic of yearly data coincides with the 11-year solar cycle.

INTRODUCTION

From the viewpoint of environmental preservation, it is very important to understand the short-term and long-term variation behaviors of radioactive or nonradioactive substances in the atmosphere. The understanding can provide valuable information of atmospheric changes from past to present, which may be used as baseline data for environmental monitoring. TRMC has monitored radionuclide concentrations in the atmosphere for the last twenty years. The measured data could help understanding the radionuclide distribution and determining the most important factors that affect atmospheric radionuclide transportation in the air. Fourier analysis can be used to check if the substance concentration varies periodically. In this paper we have analyzed the long-term data of radioactive substance concentration in the air by the Fourier method.

EXPERIMENT

Aerosol samples have been collected on the 0.8 μ m glass-fiber filters using constant flow-rate air samplers (Radeco HD-28) with pumping rate of 40 ℓ /min at four different sites in Taiwan. Filters were changed once a week. The sampling instrument on each site was setup in a weather house about 1 meter above ground. These filters were sent to laboratory and analyzed for gamma emitting nuclides by HPGe detectors.

In order to identify the reason for variation, a stainless water tray of 1 m^2 and gummed papers with 50×25 cm^2 were used to calculate deposition velocity in each station. A 1-m diameter pan for collecting rain water was set at station No.3 for calculating washout ratio.

RESULTS AND DISCUSSIONS

Results of measured gamma emitting nuclides show that cosmogenic ^{7}Be is the only nuclide found in the air of Taiwan. Some man-made nuclides such as ^{103}Ru, ^{106}Ru, ^{131}I, ^{134}Cs, ^{137}Cs, ^{140}Ba, ^{140}La, ^{141}Ce and ^{144}Ce produced by nuclear testings in China were detected. The consideration of variation behavior of these nuclides should be based on their production and removal processes. Man-made nuclides vanished within two months because of their short half life. Cosmogenic nuclides existing in the air are continuously generated.

The monthly average ^{7}Be concentrations measured in a 20-year period at four monitoring stations are shown in Figure 1. Spring peak and minima in fall and winter is a well known phenomenon.(1,2) However, in some areas of the world, two peaks may appear in spring and fall.(3,4) The northern part of Taiwan shows two peak during winter and spring while in the southern part two peaks appear in spring and fall. Both

demonstrates that these phenomena reveal two effects: one effect is with an obvious one-year cycle,and the other with a half-year cycle. In northern Taiwan, the one-year cycle is more notably than the half-year cycle.

The galaxy cosmic-ray dose is calculated by the CARI-2 computer program.(5) Input parameters include longitude and latitude of Taiwan, depth of atmosphere and heliocentric potential in each year.(6) The program calculates cosmic ray flux intensity and converts it into effective dose by conversion factors Based on ICRP No.60 table A-2. Correlation coefficients between ^{7}Be concentration and cosmic ray dose are listed in Table 1. Yearly average data have better, although weak, correlation with cosmic ray, but monthly data seem to have no correlation. This result indicates that the 11-year solar cycle of cosmic ray plays an important role of long-term variation of ^{7}Be concentration because of the change in production rate. However, for seasonal variation cosmic ray makes no obvious influence.

Transportation from stratosphere to troposphere has been shown to occur in spring and fall because of the direction change of the jet flow in stratosphere. This is the reason that half-year cycle variation exists. Deposition and washout by rainfall can be collected by water trays, gummed papers and a pan. Dry deposition is related with aerosol size that varies negligibly in natural environment. This can be proved by the result obtained from ^{7}Be on gummed paper. We could divide the lid height by residual time of ^{7}Be to estimate dry deposition velocity, which is 0.4 cm/s in Taiwan. That is quite close to the calculated value 0.45 cm/s obtained from monitored result of gummed paper. However, total deposition velocity is different as shown in Table 2. The 10-year's average washout ratio is 390. It means that after raining ^{7}Be in rain water is 390 times higher than in air. Washout effect by rainfall is the only reason for different deposition velocities and seasonal variations.

CONCLUSION

From the results of analysis mentioned above, we conclude that:

1. Concentration of ^{7}Be in air has minor difference between northern and southern Taiwan, with respective value of about 4 mBq/m^3 in the north and 3 mBq/m^3 in the south.
2. Seasonal variation of ^{7}Be concentration behavior is different between northern and southern Taiwan. One year cycle in Northern Taiwan is apparent.
3. Seasonal variation is due to different deposition velocities caused by washout effect of rainfalls.
4. Half-year cycle may be due to air transportation from stratosphere to troposphere.
5. Variations of yearly average data are almost consistent with the 11 years' solar cycle.

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Table 1. Average ^{7}Be concentrations and their correlation coefficients with cosmic ray.

Station Number	Longitude (degree)	20 years Average (mBq/m^3)	Yearly data Correlation Coefficient	Monthly data Correlation Coefficient
1	25.30 N	3.89±2.04	0.43	0.25
2	25.20 N	3.85±1.95	0.81	0.40
3	22.67 N	2.74±1.59	0.37	0.08
4	21.95 N	2.78±1.55	0.20	0.24

Table 2. Total deposition velocity and average total flux of ^{7}Be.

Station Number	Deposition velocity (cm/s)	Total flux by water tray method ($Bq/m^2 \cdot M$)	Corresponding residual time (day)
1	0.95	103.8	17.1
2	0.87	80.0	18.6
3	0.69	32.8	23.5
4	0.66	31.3	24.6

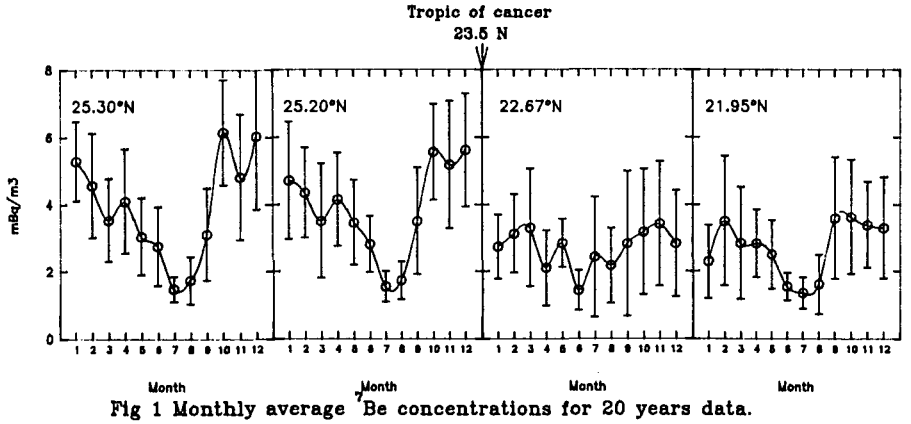

Fig 1 Monthly average ^{7}Be concentrations for 20 years data.

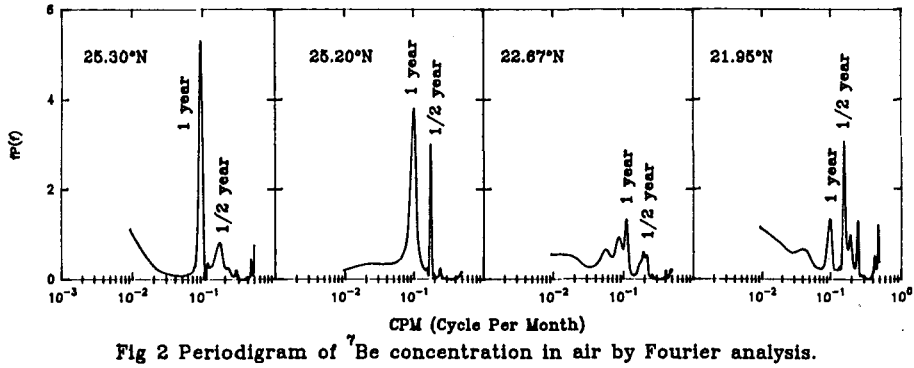

Fig 2 Periodigram of ^{7}Be concentration in air by Fourier analysis.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE A New method for monitoring of mixed radiation in natural environment

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MAJOR SCIENTIFIC TOPIC NUMBER 1 ..... (see page 7)

ABSTRACT (See instructions overleaf)

Natural radiation environment is characterised by existence of terrestial gamma component and cosmic mixed radiation component. The cosmic component is in fact the mixture of different types of particles whose contribution to the total absorbed dose depend on the latitude and altitude. At the sea level and at the altitudes of the passenger aircrafts, fotons, muons and neutrons have to be considered.
A new method for determination of ambient dose equivalent in natral radiation environment with the sensitivity down to single nSv is based on the use of high pressure 3He proportional counter in a polythene moderator. The very high sensitivity, especially for monitoring of neutron component , is achieved by applying of an active spectrometry system which register the pulse height apectra from 40 mm diameter spherical 3He proportional counter. The inherent background of this counter was determined in Low Level Laboratory at Asse salt mine. It is shown that spectral analysis of the signals from 3He detector give not only high sensitivity with regard to ambient dose equivalent but also improves the quality of the measurements. A special instrumentation for low-level neutron monitoring is described in which a quality control method has been implemented.

Cross Sections for Production of ^{7}Be, ^{22}Na, and ^{24}Na in Thin Targets of Carbon, Magnesium, Aluminum, Silicon, and Silicon Dioxide by Bombardment with Monoenergetic Protons at 30 to 67.5 MeV

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ABSTRACT

Measurements of cross sections were made for the proton-induced production of ^{7}Be, ^{22}Na, and ^{24}Na from carbon, magnesium, aluminum, silicon, and silicon dioxide. The cyclotron of the Crocker Nuclear Laboratory at the University of California at Davis was used to bombard small stacks of thin, high purity target foils with protons at discrete energies between 30 and 67.5 MeV. Following bombardments, pulse-height spectroscopy was used to count easily-detectable radionuclides in the target foils. The cross sections measured agreed with prior studies where thin targets were used. Applications for the results of this work may include: dose-component calculations for radiation buldup due to cosmic-ray bombardment in the hulls of air- and spacecraft; shielding and dosimetry calculations to support the application of proton accelerators in medical radiation therapy; or interpretation of the history of meteorites and the processes that shape planetary surfaces.

ACKNOWLEDGEMENTS

This project was directed and funded by Dr. Peter Englert of San Jose State University and sponsored by Dr. Carlos Castaneda of the University of California at Davis Crocker Nuclear Laboratory. Mr. Anthony Beverding and Miss Kyeong Kim worked with me during collection of the data.

INTRODUCTION

The probability for the occurrence of a nuclear reaction between an accelerated particle and a stationary target nucleus is expressed as a quantity called the reaction cross section. A cross section is the effective target area presented by the stationary nucleus from the perspective of the incident particle. The purpose of this study was to measure the cross sections for production of readily detectable radionuclides in high purity foils of common materials. Energetic protons are present at very high altitudes, in space, and are used in special types of medical radiation therapy. Accurate cross sections for proton-induced reactions in materials found on planetary surfaces and in medical radiation therapy settings, or used to construct air- and spacecraft, may be valuable as humanity's knowledge and reach expand.

EXPERIMENTAL AND MATHEMATICAL METHODS

Stacks of five, thin, high-purity (at least 99.999%) targets were bombarded with protons accelerated in a vacuum by the Crocker Nuclear Laboratory (CNL) cyclotron at the University of California at Davis on nine days during 1993. Target stacks were composed of three, 15-mm-diameter disks of carbon, magnesium (both 0.125-mm thick), silicon, or silicon dioxide (both 0.500-mm thick) with 0.055-mm aluminum foils upbeam and downbeam. Proton energies within the stacks were calculated using the modified Bethe-Bloch formula as described by Attix (1). Proton current was measured using a Faraday cup at the end of the beam line. After bombardment, short-lived ^{24}Na activity in the aluminum, magnesium, silicon, and silicon dioxide targets was promptly measured at CNL and San Jose State University, where longer counts for ^{22}Na were also made. Carbon and silicon dioxide targets were also counted for ^{7}Be. Production cross sections were calculated using the relation described by Fink, Sisterson, Vogt, *et al.* (2). The percent uncertainty associated with each measured cross section was calculated by the root-sum-of-squares method.

RESULTS

The results of the measurements described are shown below. Table 1 contains cross sections with associated percent uncertainties for proton-induced production of ^{7}Be from carbon and ^{22}Na-^{24}Na from magnesium and aluminum. Table 2 contains cross sections with uncertainties for proton-induced production of ^{7}Be, ^{22}Na, and ^{24}Na from silicon and silicon dioxide.

Table 1. Measured Cross Sections for ^{7}Be, ^{22}Na, and ^{24}Na Production in C, Mg, and Al vs. Proton Energy.

Measured Cross-sections (mb) ± Percent Uncertainty (95% C.L.)					
Target Material:	Carbon	Magnesium		Aluminum	
Radionuclide Produced:	^{7}Be	^{22}Na	^{24}Na	^{22}Na	^{24}Na
$E_{protons}$ (MeV)					
27.04				n/d*	0.01 ± 11.4%
28.14					0.10 ± 10.2%
29.77			5.57 ± 3.7%		
30.00				0.71 ± 68.7%	0.01 ± 18.3%
35.00					0.92 ± 7.9%
39.62					2.15 ± 8.8%
39.96				41.69 ± 4.6%	1.40 ± 4.3%
40.01				42.16 ± 4.2%	1.34 ± 5.1%
41.97	23.38 ± 6.2%				
42.02		82.85 ± 3.2%	8.67 ± 3.8%		
42.33	27.13 ± 6.7%		8.35 ± 3.8%		
42.50				46.09 ± 4.7%	1.94 ± 3.9%
42.50				46.49 ± 5.0%	1.97 ± 4.0%
45.00					4.43 ± 8.8%
45.11				50.08 ± 5.9%	4.23 ± 7.8%
48.25				43.57 ± 4.6%	6.30 ± 7.8%
48.91				37.12 ± 7.2%	5.70 ± 8.0%
49.03				37.88 ± 5.4%	6.24 ± 4.0%
49.31		73.52 ± 4.8%	7.57 ± 4.0%		
49.54	25.23 ± 6.5%				
49.58		74.48 ± 4.3%	6.85 ± 7.9%		
49.85	23.95 ± 6.6%	77.42 ± 5.7%	7.59 ± 4.0%		
50.00				34.81 ± 8.6%	6.45 ± 4.2%
50.00				37.42 ± 7.2%	6.57 ± 3.9%
50.00					6.89 ± 7.8%
52.50				36.88 ± 7.2%	7.81 ± 7.7%
58.12				29.58 ± 5.5%	10.48 ± 3.8%
58.19				29.11 ± 6.6%	10.53 ± 4.2%
59.34	23.00 ± 5.6%				
59.41		67.52 ± 3.7%	7.43 ± 4.0%		
59.61	22.62 ± 6.4%				
59.64		68.59 ± 3.8%	7.15 ± 4.0%		
59.87	21.50 ± 6.3%		7.17 ± 4.0%		
60.00				26.01 ± 6.0%	10.86 ± 4.1%
60.00				28.41 ± 5.8%	10.92 ± 4.1%
63.75				22.39 ± 9.7%	11.73 ± 7.8%
64.09				26.59 ± 9.4%	12.25 ± 7.8%
66.66					10.95 ± 4.2%
66.75					11.14 ± 4.1%
66.90	19.89 ± 6.8%				
66.96		61.92 ± 4.6%	6.61 ± 4.2%		
67.14	20.24 ± 6.5%				
67.17		61.97 ± 4.0%	6.57 ± 4.2%		
67.39	19.09 ± 6.9%		6.76 ± 4.2%		
67.50				26.65 ± 7.9%	11.08 ± 4.3%
67.50					11.29 ± 4.2%
67.50					12.01 ± 7.7%
67.50					12.07 ± 7.8%

*No peak detectable above background.

Table 2. Measured Cross-sections for ^{7}Be, ^{22}Na, and ^{24}Na Production in Si and SiO_2 vs. Proton Energy.

Measured Cross-sections (mb) ± Percent Uncertainty (95% C.L.)						
Target Material:	Silicon			Silicon Dioxide		
Radionuclide Produced:	^{7}Be	^{22}Na	^{24}Na	^{7}Be	^{22}Na	^{24}Na
$E_{protons}$ (MeV)						
29.77	n/d*	0.38 ± 35.9%	n/d			
30.49						n/d
32.70						0.01 ± 13.7%
34.80				1.87 + 8.9%	1.26 ± 5.5%	0.001 ± 9.7%
41.70				2.90 ± 12.4%	1.35 ± 12.4%	
41.42						0.48 ± 8.5%
41.61	0.62 ± 18.0%	1.19 ± 15.3%	0.29 ± 3.6%			
43.16						0.58 ± 8.5%
44.84				5.28 ± 6.8%	1.40 ± 14.2%	0.69 ± 8.5%
46.74					4.57 ± 46.2%	0.60 ± 7.7%
48.32					7.54 ± 36.0%	0.64 ± 7.7%
49.65	0.22 ± 26.1%	3.14 ± 3.9%	0.68 ± 7.7%			
49.85				7.70 ± 7.3%	8.82 ± 6.1%	0.75 ± 7.7%
51.02	0.48 ± 22.7%	12.04 ± 3.8%	0.74 ± 7.7%			
52.36	0.50 ± 21.0%	14.12 ± 3.5%	0.82 ± 7.7%			
59.07	1.51 ± 44.3%		1.24 ± 3.4%			
59.17	0.64 ± 32.4%	21.94 ± 2.8%	1.21 ± 3.3%			
64.98				8.91 ± 9.1%	20.53 ± 8.1%	
66.19				8.72 ± 7.1%	19.50 ± 5.0%	2.06 ± 7.6%
65.20	0.88 ± 23.1%	24.40 ± 3.8%				
66.30	0.70 ± 33.8%	24.25 ± 3.6%	0.83 ±6.5%			
67.39	0.66 ± 35.8%	24.35 ± 3.8%				

*No peak detectable above background.

DISCUSSION

Cross sections measured for the $^{27}Al(p,3pn)^{24}Na$ reaction (Table 1) agreed up to 50 MeV proton energy with those reported in a recent compilation (2). Above 50 MeV, the cross sections for production of ^{24}Na from aluminum described in this work agreed with those reported by researchers who used a similar thin-foil bombardment geometry at the Harvard Cyclotron Laboratory (3). Cross sections for proton-induced ^{22}Na production in aluminum (Table 1) also agreed with those reported in Reference 2. Cross sections for ^{7}Be production in carbon (Table 1) were 5 to 10 mb higher than those reported in a prior study (4). The cross sections measured for the $^{25}Mg(p,2p)^{24}Na$ and $^{26}Mg(p,2pn)^{24}Na$ reactions (Table 1) did not clearly define the proton energy at which the production maximum occurs. Cross sections for the ^{22}Na production from magnesium showed a gradual decline from 83 to 62 mb between 42 and 67 MeV proton energy. The cross sections measured for ^{22}Na and ^{24}Na production from silicon (Table 2) showed maxima near 60 MeV proton energy. The results in Table 2 defined the thresholds for production of ^{7}Be, ^{22}Na, and ^{24}Na from silicon dioxide, but did not clearly define the maximum cross sections.

CONCLUSION

High purity, thin target foils of carbon, magnesium, aluminum, silicon, and silicon dioxide were bombarded with protons at discrete energies from 30 to 67.5 MeV. The ^{7}Be, ^{22}Na, and ^{24}Na produced were measured by pulse-height spectroscopy. Experimental cross sections for production of these radionuclides were calculated using the results of the spectroscopic analyses and the target and bombardment parameters. Where existant, agreement with previously-reported values was observed except where Coulombic interactions in the target stacks may have caused proton energy losses.

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LUNG CANCER RISK FROM RADON
MODIFIERS OF EXPOSURE - EFFECT RELATIONSHIP

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Research supported by the Internal Grant Agency of the Ministry of Health - Reg.No 18591-3

INTRODUCTION

Studies of underground miners of uranium and other substances are at present the principal source of information on the long term effects of exposure to radon and its progeny. One of the largest such studies is that of uranium miners in West Bohemia (Jáchymov). This study, sometimes referred to as the S cohort, was set up in 1970 by the late Josef Ševc (2). The purpose of this paper is to show, how the exposure - effect relationship is modified by time since exposure, age at exposure, and exposure rate.

METHODS

The study population of the S cohort involve uranium miners, that started underground work at the Jáchymov and Horní Slavkov mines in the period 1948-59, and had worked for at least four years. A total of 4320 men satisfied these criteria.

During the decade up to 1990, follow-up of the cohort mainly relied on the national population registry. In order to improve the follow-up, a series of additional checks were conducted: in the files of the Czech and Slovak Pensions Offices, by local enquiries, and by direct correspondence (4). These additional efforts resulted in an increase of more than 10% in the numbers of men known to have died or emigrated.

An exceptional feature of the S study is the large number of measurements of radon concentrations made in each mine-shaft (mean number per year and shaft was 223 in the period 1949-60). Each man's annual exposures to radon progeny in terms of working levels were estimated combining measurement data with the men's employment details. Recent data revisions revealed that for some of the men (about 10%), exposures at other Czech mines had not previously been taken into account (3). In the most recent revision of exposures, appropriate adjustments were made for individual job categories (miners, other manual professions, supervisors, and exploratory workers) using information on exposure levels in different work places (stope, drift, cross-cut, and raise).

Generally, person-years at risk were calculated for each man, starting at four years after entering underground employment and ending at the earliest of date of death, emigration, 85th birthday, loss to follow-up, or 1.1.1991. The person-years were cross-classified by attained age, cumulated exposure in windows formed by time-since-exposure, exposure rate, and age-at-exposure. Poisson regression models for excess relative risk (ERR) were

fitted by maximum likelihood method. It was assumed that the number O of deaths observed in each cell had mean in the form

$$O = c\,E\,(1 + ERR(w,x)),$$

where E is the number expected from national rates, ERR is excess relative risk, w and x denote exposure history and modifying variable, respectively, and c is an intercept term that allows the mortality rate for 'unexposed' cohort to differ from that in the general population.

RESULTS AND DISCUSSION

The increased mortality (O/E=1.58) in the cohort is largely affected by mortality from lung cancer. Nevertheless the mortality from violence and accidents is also increased, namely in the first part of follow-up, whereas the mortality from other causes increases after 20 years since first exposure (Tab.1).

Tab.1: Time and cause specific mortality

Time since 1st exp	Lung cancer	Violent deaths	Other causes	Respiratory diseases	Circulatory diseases	Cancers excl lung
4- 9	5.22 *	1.80 *	0.79	0.48	0.86	0.83
10-19	8.62 *	1.75 *	1.10	1.23	1.22 *	1.08
20-29	5.14 *	1.30 *	1.27 *	1.38 *	1.27 *	1.13
30-	3.12 *	1.37 *	1.22 *	1.35 *	1.14 *	1.30 *

(the asterisk denotes significant increase at 5% significance)

A similar time pattern is observed in some groups of diseases generally associated with tobacco consumption. When the cohort was identified in 1970, information on the smoking habits in the cohorts could not be recorded. As the increase in mortality from cancers other than lung cancer is not likely to be associated with radon exposure (6), the data indicate that smoking in the S cohort might be different from the general population. This conclusion is supported by a survey in other uranium mines (study N) estimating about 75% smokers in the miners, somewhat more than 66% corresponding the male average in the country (5) .

Most of the miners' studies demonstrated the linear relationship between relative risk and cumulative exposure (5). In a simple model in which the excess relative risk increases linearly with total 5 year lagged cumulative exposure, two parameters are estimated: the coefficient of ERR/WLM and intercept. The overall ERR/WLM estimate 0.013 for the most recent data is higher than the comparable estimates previously reported for the study: 0.0034 (5) and 0.0064 (4). The corresponding intercept is 2.43 (CI:1.73-3.41).

Factors known to influence the excess relative risk per unit exposure were analyzed. The strong influence of time since exposure (TSE) was found. In addition, the effect of exposure rate was investigated. The cumulative lagged exposures in the three TSE windows were split into two windows according to their concentrations in term of WL (0-3.9, 4+). It was found that the ERR/WLM estimate corresponding to higher exposure rates was significantly lower. The effect of age, that was found in almost all miner studies (5), was investigated in the same way as the other exposure modifiers, i.e. by the method of windows. The analyses demonstrated a significant influence of the age-at-exposure factor when both time-since-exposure and age-at-exposure factors were included in the model. The estimates in the model with all the exposure modifiers are given in Tab.2 .

Tab.2: Effect of time since exposure, exposure rate, and age

	Estimate	95% CI	Chi-sq	DF
Intercept	1.46	1.00 - 2.14		
ERR/WLM	0.124	0.077 - 0.199		
TSE:			69.53	2
5 - 14	1			
15 - 24	0.26	0.11 - 0.41		
25 - 34	0.07	0.02 - 0.13		
Exp.rate:			8.69	1
0 - 3.9	1			
4 -	0.47	0.21 - 0.72		
Age-at-exp			52.07	2
- 29	1			
30 - 39	0.60	0.36 - 0.85		
40-	0.28	0.18 - 0.38		

The confidence interval for the intercept in the last model is narrower in comparison to other models. Further investigations of the intercept showed that when age factors were present in the model as continuous variables, the age-dependent intercept remained nearly constant (close to 1.5) suggesting that in the absence of exposure to radon, the estimated mortality from lung cancer in the cohort would be about 1.5 times higher than in the general population. If this estimated baseline lung cancer mortality in the study were true, the ERR/WLM coefficient related to simple cumulative exposure would be about 0.015 (95%CI: 0.0135-0.0165).

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THE INFLUENCE OF RADIATION AND NON-RADIATION FACTORS FOR LUNG CANCER RISK IN WORKERS OF ATOMIC PLANT MAYAK

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INTRODUCTION

All possible risk factors must be taken into account for the evaluation of radia risk of human cancer because of the polyetiology of most types of tumors. evaluation of such a "confounding" factor as the smoking, that is a strong carcino agent, is very important for the hygienic reglamentation of irradiation based on lung cancer risk. These circumstances are omitted usually in the epidemiolo investigations of occupational cohorts to be compared, because an exact estimatio smoking factor is very difficult in a large cohort. On the other hand indus hygienists are of the opinion that the persons, working under bad conditions, s more than those in the general population.

Only prospective (cohort) investigations are known for ^{239}Pu incorporation and cancer, where the age are taken into account besides the level of irradiation ef (1,2,3). The use of the "case - control" method broadens the investigation possibi and can give new results.

SUBJECTS AND METHODS

The investigation of lung cancer by "case - control" method was carried for personal of the first Russian atomic plant Mayak. The main factor of the professi effect was the radiation one: the external gamma-irradiation and, in most cases, contact with the airborne ^{239}Pu. External gamma-irradiation doses were accounte dosimeter control service of the enterprise with individual film badges. The amou incorporated ^{239}Pu was estimated by its spontaneous excretion with urine (biophy laboratory of the branch № 1 of Biophysical Institute, the laboratory leader d V.F.Khokhryakov).

The main group consists of all lung cancer cases from 1966 till 1991 among personal of basic shops of the atomic plant, verified by morphological investiga (162 persons, among them 148 men and 14 women). Pair control (persons, that not fallen ill with lung cancer) was matched among the personal of the same plant persons, 296 men, 42 women). Matching was made by sex, year of birth (±5 years), of the working begin at the enterprise (±2 years), profession, working place. The on the smoking was received after direct interviews by unified method, hold by me experts.

STATISTICAL ANALYSIS. The main and the control groups, were compare unvariable and multivariable analysis. The procedure of multiple logistic regres with stepwise selection of independent variates, based on the maximal likeli method, was used for multivariate analysis, was calculated odds ratio (OR). More attributive risk (AR) was calculated. The BMDP package was used for statis estimation of data.

RESULTS

Eleven potential risk factors (professional, condition of life, medico-biological) evaluated using a logistic regression model, 5 insignificant factors, were excl

(preceding professional factors, age of smoking's start, frequent pneum pulmoneum tuberculosis), the rest was arranged (by odds ratio) in decreasing o smoking > plutonium pneumosclerosis > plutonium incorporation in body > chr obstructive pulmonary diseases > decrease of body mass > external gamma-irradia OR for them were accordingly 6.6, 4.6, 3.1, 1.8, 1.8. The portion of the occupati cancers among the workers of the atomic plant, evaluated on the base of attrib risk, is 26%.

Dose-response relation was investigated for the three most important factors of cancer risk (smoking, plutonium incorporation and external gamma-irradiation). gradation were used to describe plutonium incorporation. "Zero" or the lo gradation serves as the basic level to be compared increasing factor gradations "case-control" analysis. A trend to decreased of morbidity risk was noted for t gradations following "zero" 0.149-0.59 kBq, 0.6-2.29 kBq and 2.3-8.99 kB plutonium incorporation, OR-ad was 0.56, 0.59 ($P<0.1$), 0.83; trend of the descen branch of the curve was the significant value ($P<0.05$). The risk to fall ill with cancer became clearly defined at 9-35.6 kBq (OR-ad 2.48) and increased sharpl 35,7-140,6 kBq (OR-ad 59,3) (Fig. 1). Dose-response for absorbed lung dose similar character.

Four gradations were used to describe external gamma-irradiation in the ge cancer group. A certain dose dependence was traced using the crude odds estimates (OR-cr): the highest increase of OR-cr 2.2 was at the gradation >4.0 However, no statistically significant OR were obtained after adjustment. The abs of a clear-cut dose dependence of lung cancer development on the external ga irradiation testifies to a lesser effect for our sample cohort comparing with A-b survivors whose risk of lung cancer occurrence at the dose of over 2 Gy is twice a We can be suggested that, this difference is due to low dose rate in our case, that chronic irradiation.

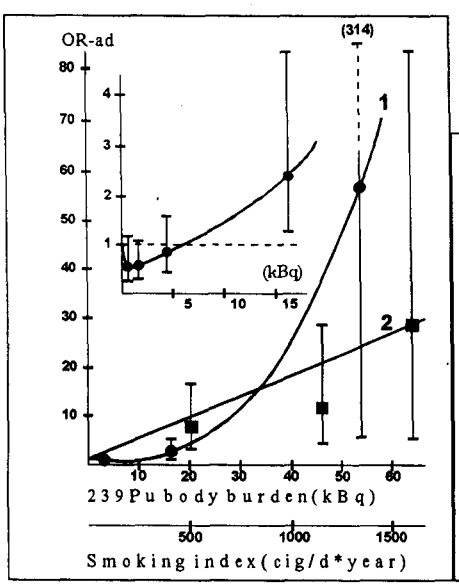

Figure 1. Lung cancer risk, depending on ^{239}Pu body burden and smoking (1 - ^{239}Pu, 2 - smoking, bar - 95% CI)

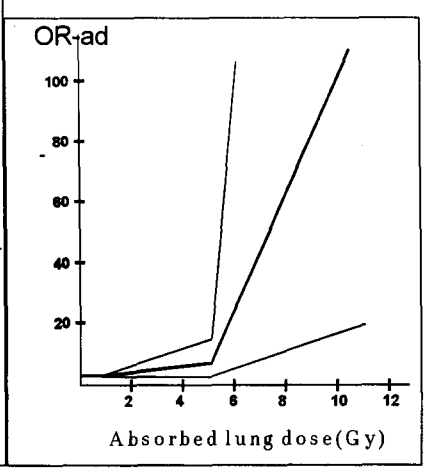

Figure 2. Lung cancer risk (with 95% CI) for separate linear segments of dose-responce curve

Fig. 1 show the dose dependence on smoking for the general carcinoma group progressing increase was observed at all gradations. At the last gradation it was 20. dependence approximated a linear one. Having in view the linear character of response dependence and based on the data presented we can evaluate the risk f smoking. Smoking one pack of papiroses daily for 5,10 and 20 years led to 2-,4-, an fold increases in the risk of lung cancer respectively, as compared non-smokers. increase or decrease in the number of papiroses smoked per day varies the risk respecti Our risk estimates are in good agreement with the values obtained from prospe investigations in which smoking was investigated as an independent factor /BEIR 1980/.

As distinct from a simple relationship for smoking, the dose-response relation plutonium incorporation is more intricate: it has a non-linear threshold character. decrease in lung cancer incidence ratio at low plutonium incorporation is describe American authors at three nuclear enterprises: Los-Alamos Laboratory, in Hanford Rocky-Flats (1). The mechanism of the possible effect may be: the first one is the activa of immune defence, the second one is the intensification of synthesis of DNA-repara ferments. Data on the non-linearity of the effect and on the presence of threshold at radiation levels are accumulating as well. The investigations on indoor radon con point to that fact (4).

Taking into account the non-linear character of the relation between lung cancer and plutonium influence we have used the square and linear-square models to descri Equations of following form were obtained:

$y=(-0.21\pm0,08)x_1+(0.024\pm0.002)x_1^2$ (I), $y=(0.02\pm0,0005)x_1^2$ (II)

$y=(-1.36\pm0,59)x_2+(1.11\pm0.07)x_2^2$ (III), $y=(0.95\pm0,021)x_2^2$ (IV)

were y=ORad, x_1 - ^{239}Pu body burden (kBq), x_2 - absorbed lung dose (Gy)

It should be noted, that the linear-square model contains a the negative linear term. evident, that this negative linear term can represent the defence mechanism of low [2] doses.

The estimate of the threshold was carried on this equations base. The threshold fo ^{239}Pu body burden correspond 3.8 kBq, for absorbed lung dose - 0.85 Gy. The regres coefficients from equations II and IV corresponds to the excess relative risk owing kBq^2 or 1 Gy^2. Non-linear dose-response curve was divided into three linear inter Excess risk was obtained for them by logit-regression (with the adjusting of smo external gamma-irradiation and mass index): I 0-0.85 Gy, subthreshold effect; II 0.8 Gy - slow increase, excess relative risk 1.7 Gy^{-1}; III 5.21-17 Gy - fast increase, excess rel risk 21.3 Gy^{-1} (Fig. 2).

Interaction of these three factors was estimated. Multiplicativity of body burden [2] and external irradiation is pointed out only high levels of both factors. The interactio plutonium incorporation and smoking, external gamma-irradiation and smoking different at different smoking levels. Additivity were observed in low inte multiplicativity and antagonism in middle and high interval of smoking scale. T effects were considered the base of the two-mutation model of radiation carcinogenesis

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CONSTITUTION OF A GROUP OF CZECH AND FRENCH URANIUM MINERS IN ORDER TO ESTIMATE LUNG CANCER RISK LINKED TO LOW CHRONIC EXPOSURE TO RADON DECAY PRODUCTS.

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SUMMARY

West Bohemian and French uranium miners are characterized by a long duration of exposure to radon and its decay products, in comparison to most of the other groups of miners, studied in the recent international joint analysis by the National Cancer Institute (NCI) in USA. This analysis has confirmed the linearly increasing risk of lung cancer by cumulative radon exposure, describing the different factors that may influence this dose-response relationship. One of the main factors presently discussed is the influence of the exposure-rate effect: in other words, has the same cumulated exposure spread over 10 years the same risk of lung cancer as if it is cumulated in 2 years ? The implication of an inverse exposure-rate effect for low chronic exposures as well as some methodological approaches will be discussed and tested by using the data of the Czech and French cohorts. These two cohorts present annual exposures varying by a factor of 5 to 10, French exposure rates being close or even less than 0.1 Working Level during the last 20 years. The project is integrated in a larger European project on uranium miners, co-ordinated by IPSN.

INTRODUCTION

Cohort studies of uranium miners in the Czech Republic and in France have demonstrated a clear increase of lung cancer risk linked to occupational radon exposure. Both studies contributed substantially to an international joint analysis co-ordinated by the National Cancer Institute (NCI) in Bethesda (USA). The results of this analysis, based on more than 2,700 lung cancer cases observed on 65,000 miners, demonstrated a linearly increasing risk of lung cancer deaths with cumulated exposure to radon decay products for all the cohorts. Factors able to influence this dose-response relationship have been discussed recently in a paper which summarizes the results of this joint analysis (1). An inverse dose-rate effect has been observed in 10 out of the 11 cohorts (2). This inverse dose-rate effect has not yet been demonstrated for low cumulated exposures (< 50 WLM*). Its implication in the extrapolation of the risk coefficient obtained in these miners studies to the risk estimated for domestic radon could be of main importance, because it would lead to higher risk estimation for low chronic daily exposures protracted over the whole duration of life than for the same total exposure cumulated in only a few years. The purpose of our project is to focus on the different factors involved in the study of a low chronic exposure: precision of individual dosimetry, comparability of the cohorts in dosimetric survey, duration of exposure, confounding factors.

EUROPEAN PROJECT ON URANIUM MINERS STUDIES

A European project has been accepted for the next 4 years by the Commission of the European Union (DG XII), in order to study more precisely the different components able to influence the dose-response relationship obtained by the miners studies.

In the large joint analysis, realized by the NCI, not only uranium miners were included but also other miners (iron, tin.....), the criteria of inclusion being the existence of measured information about radon exposure for at least part of the working period. In the European project, only uranium miners will be included, in order to eliminate components which may be present in other types of mines and could influence the dose-response relationship between lung cancer and radon exposure.

The following uranium miners groups will contribute to this European project:

1. Two cohorts of miners from the Czech Republic, defined by the period of entry in the mines and by the region;

2. Two cohorts of uranium miners in France, characterized by the period of entry and, in parallel, by the precision of exposure estimation;

3. A very large group of uranium miners from Germany, having experienced a more or less precise survey of exposure depending on the period of work.

The results concerning the Czech and French uranium miners cohorts have already been partially published and these cohorts are presently undergoing an extension of follow-up with a new estimation of the risk coefficient next year. The German large group needs more time to be followed up in a precise way and the first step decided in the European project is a case-control approach with the aim of identifying other components, like tobacco, able to influence the risk of lung cancer amongst uranium miners.

Consequently the European project is supporting two parallel approaches: the first is estimating the dose-response relationship and its modifying factors, the second is a case-control approach, testing first the feasibility of the collection of different factors directly or indirectly linked to the mining environment. The different steps of the cohort approach are described hereafter.

EXPOSURE ASSESSMENT

The first step of the Czech-French collaboration will be a critical examination of the individual exposure estimation for the different periods of work. This exposure assessment will be discussed not only for radon and its decay products, but also for gamma and long lived uranium dust exposure. A job matrix will be elaborated, mainly to test consistency of the individual exposure for the periods during which direct measurements were rare.

Both the Czech and French cohorts are characterized by a long duration of exposure, giving the opportunity for testing the risk for different durations of exposure and, giving in parallel some guaranty for describing the whole history of an individual mining experience. Indeed, in the joint analysis of NCI, a previous experience of some other mining activity was a modifying factor of the dose-response relationship. Including in this joint analysis mining populations with a very short duration of underground mining (for example less than 2 years) as well as those having passed all their working activity in the same type of mine may also have introduced some bias, these populations being probably different with respect to many other factors.

Distribution of the registered exposures shows a time-dependent variation, linked mainly to the period at which radioprotective measurements were introduced in the mining environment. This period of decline of the annual exposure rate is the same in the Czech and French cohorts, close to the year 1956. Since that time, the individual registration of the exposure is complete and has been realized in time, on a regular basis. Consequently exposure assessment has to focus on different periods of exposure, but mainly on the ten first years, when individual exposure was estimated indirectly. In France, since 1956, the same approach of dosimetric survey has been applied to all the miners and has been registered in time.

For estimating the risk linked to chronic low annual exposure, limiting our cohort analysis to those miners who had begun to work underground since 1956 is probably the best unbiased approach, provided the power of our study is high enough to estimate this risk linked to low annual exposure. Increasing the number of miners by a joint analysis of the Czech and French underground miners makes this analysis more realistic. The opportunity of including or excluding those having worked before 1956 will be tested in this common approach. For those miners entered since 1956, a registration of the gamma exposure and ore dust was done in time on an individual basis in France. Approaching a risk linked to low exposure needs a high precision in the individual exposure and an accurate estimation of confounding factors, mainly of those concomitant to the radon exposure. The analysis will be realized by joining the three following cohorts:

- French miners having worked underground for at least 2 years and having begun underground work before 1972. The French cohort may be extended to a larger group in the future, the follow-up of those having entered since 1972 being not yet complete;

- Czech S cohort, including west Bohemian uranium miners (Jachymov, Horni Slavkov, Chodov), first empoyed between 1948 and 1959;
- Czech N cohort, including uranium miners in central Bohemia (Pribram), first employed between 1968 and 1974.

Considering the present endpoint of follow-up in December 1990, the French cohort is contributing with 1,785 miners for 50,000 person-years, each of the two Czech cohorts respectively for 100,000 person-years, with 4,230 miners in the S cohort and 5,625 miners in the N cohort.

The following table indicates the distribution of mean annual exposure in WLM*, from 1946 to 1990.

Table 1: Mean annual radon exposure ($WLM.y^{-1}$) in the cohorts

Period	French cohort	Czech S cohort	Czech N cohort
1946 - 1950	25.5	50.7	
1951 - 1955	30.5	21.9	
1956 - 1960	2.9	11.8	
1961 - 1965	2.9	9.6	
1966 - 1970	3.5	6.4	1.0
1971 - 1975	2.8	1.8	1.1
1976 - 1980	1.8	0.8	0.8
1981 - 1985	1.0	0.8	0.6
1986 - 1990	0.4	1.2	0.6

** WLM (Working Level Month) is a unit of exposure multiplying a concentration of radon decay products by the duration of exposure. A yearly exposure to 11 WLM corresponds roughly to a monthly exposure to 1 WL, the monthly exposure being defined as 170 working hours. 1 WL is equivalent to any combination of radon decay products in 1 liter of air, that results in the emission of 130,000 MeV of energy of alpha particles.*

METHODOLOGY OF STATISTICAL ANALYSIS

The methodology of statistical analysis is close to the one used for the joint analysis by NCI, but with two modifying points: the time dependent modelling of the dose-response relationship includes also an approach with an external reference group and a different concept of the influence of the rate effect. The usual approach to analyse the exposure rate effect is based on a variable defined as the proportion of cumulative exposure and corresponding cumulative duration of the exposure. However, due to ventilation improvement in the mines, the actual exposure rates were different not only within single studies but also during separate miners' employment periods. Consequently, to take into account the fast decline in time,mainly (between 1955 and 1956), we propose a new approach based on exposure rate windows. The individual cumulative 5 year lagged exposures at each moment of the follow-up are split into four windows according to radon concentrations, the first group being as low as achievable. In other words, if the number of cancers is high enough for the estimation of the risk linked to low doses, then the distribution of annual exposures, as described in table 1, makes it possible to estimate the lung cancer risk for radon concentrations as low as 0.1 to 0.3 WL. These concentrations are lower than those achievable in animal experiments and comparable to annual exposures of 1.1 to 3.3 WLM which correspond to annual radon concentrations indoor comprised between 200 and 700 $Bq.m^{-3}$.

CONCLUSION

We consider that the follow-up of low exposed miners will be, in the next years, the best way for testing the linear dose-response relationship and its possible modification by the exposure-rate effect in the field of low chronic alpha radiation exposure, provided that a precise information on individual exposure to radon and to other confounding factors is available.

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INDOOR EXPOSURE TO RADON- COHORT STUDY

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The research is being performed with the support of the Internal Grant Agency of the Ministry of Health (grant IGA-18611-3) and the Grant Agency of the Czech Republic (GAČR 313/ 94/ 1736).

INTRODUCTION

In the centre of a region of granitoids, carrying geological term the Central Bohemian Pluton, is a smaller area with considerable geological disturbances comprising 80 villages with 2501 houses. The houses situated on the lines of such breaks exhibit very high levels of radon-222 and products of its transformation. In the adjacent houses outside the lines of disturbances the levels are manifold lower, being still higher than the average value in the country (40 Bq/m^3 of the equivalent equilibrium concentration of radon -EEC). Other conditions being also favourable for assessing possible effects of cumulated radon exposure, e.g.. prevailing agricultural character of the area with only small migration of inhabitants, a prospective (non-parallel) cohort epidemiological study was started in 1991 by the late Josef Ševc.

METHODS

The study population includes inhabitants of the area, who had resided there for at least three years and at least one of these between 1.1.1960 and 31.12.1990. Persons with mining experience were not included in the cohort.

Data on vital status of each cohort member, residences in the past, smoking habits, occupational and other exposures have been collected through interviews of residents or relatives at home during which also data on constructional changes of buildings have been gained. Information on vital status and causes of death was collected from and/or verified at the local authorities, in the national population registry and in the health documentation of the corresponding district hospital or oncological facilities and their pathologico- anatomical departments (with regard to histological characteristics of the tumour).

Cohort members were entered in person-years accumulation either at 3 years after their birth for those born after 1960 or at 1961 for those born before 1960 and removed at the earliest date of death, emigration, or termination of follow-up. Expected numbers for mortality analyses were taken from the country statistics.

One year lasting measurements of radon daughter products by integral dosimeters (Kodak film LR 115) were performed in almost all the dwellings of the specified area (mostly 2 detectors in a house).

Individually estimated exposures were calculated by combining residence information (number of years spent in the study area) with exposure measurements in the resident dwellings). At this stage of the study the concentrations measured in a particular year (1991 or 1992) were taken as representative for all the time an individual lived in the dwelling and no specific methods were used for the assessment of previous exposures, as will be done further (measurement of Pb^{210} in surfaces, evaluation of possible influence from construction changes in the building, chromosomal analysis etc.). Fot the time spent outside the study area the country mean values were applied. Missing measurements within the study area were replaced by the corresponding site means. Individual exposures in the last 5 years were ignored (lagging) for the analysis of relation of cancer incidence to the cumulated exposure.

RESULTS AND DISCUSSION

Table 1. displays the distribution of 3727 measurements performed in the greater part of area houses , especially in permanent dwellings (1869 out of 2501), the measurements in the other part (mostly consisting of houses used for recreational purposes) are being actually revised.

Table 1: Distribution of the equivalent equilibrium concentration of radon (EEC) in 1869 houses.

EEC Rn $Bq.m^{-3}$	N(3727)	%
20-99	170	9
100-199	514	28
200-499	895	48
500- 999	246	13
1000-	44	2

The overall mean was 327 $Bq.m^{-3}$ and the maximum value found was 11 000 $Bq.m^{-3}$. The existence of two groups of dwellings with different radon levels both substantially higher than the indicated average country value was confirmed.
Maximum individual cumulated exposure found was 386 $Yr.kBq.m^{-3}$, the mean value among the cohort members was 13 $Yr.kBq.m^{-3}$.
A total of 11865 inhabitants satisfied the above mentioned criteria, however, 485 people had to be excluded for the given analysis because of incomplete data. By the end of 1994 in the cohort of 11379 persons, 2769 death were registered and their cause of death identified. The distribution of the death causes by calendar period is given in table 2.

Tab 2: Cause and calendar period specific mortality.

		lung ca		non lung ca		violence		other		unknown	all causes	
Period	PY	O	O/E	O	O/E	O	O/E	O	O/E	O	O	O/E
1961-65	27 468	12	0.83	52	0.90	8	0.36	197	0.94	6	275	0.90
1966-70	31 507	16	0.81	62	0.85	24	0.80	282	1.01	10	394	0.98
1971-75	32 720	27	1.25	72	0.91	20	0.64	269	0.84	17	405	0.90
1976-80	34 537	25	1.08	68	0.81	21	0.64	312	0.87	11	437	0.88
1981-85	36 450	34	1.45*	68	0.80	23	0.71	358	0.95	13	496	0.96
1986-90	38 476	20	0.83	73	0.83	26	0.79	329	0.94	9	457	0.92
1991-94	31 901	11	0.56	43	0.61	17	0.60	184	0.67	50	305	0.78
total	233 060	145	0.99	438	0.81	139	0.66	1 931	0.89	116	2 769	0.91

Table 3. relates the ratio of observed to expected fatal lung cancer cases to individual cumulative exposure by 1990.

Table 3: Lung cancer mortality by cumulative exposure categories by 1990.

Exposure category (Yr. kBq. m^{-3})	Person- years	O	O/ E
0- 4	79 556	7	0.78
5- 9	51 294	22	0.81
10- 19	50 591	69	1.26*
20-29	15 279	25	1.02
30-	6 480	11	0.99
total	203 201	134	1.06

The coefficient of the excess relative risk per unit exposure (Yr.kBq. m^{-3}) calculated from these data is 0.019 with 95 % confidence interval 0.006- 0.032, which using conversion factors recommended in ICRP publication No. 65 (1) on assumption of 7000 hours per year of exposure and equilibrium factor 0. 4 corresponds to 0.01 and 0.004- 0.02 per WLM. This value falls within the range of risk coefficients found in eleven epidemiological studies on miners (2) and is near to the most recent finding in the Czech uranium miners (3).

These results must be considered as preliminary only, the collection of the data on mortality and improvement of exposure estimates, resulting e.g. from methods, indicating the possible value of previous exposure, still proceeds and possible significance of important factors, which can contribute to the lung cancer mortality, such as smoking, were not yet evaluated. Although the majority of deaths in the cohort has already been registered, further cases might be identified a.o in those respondents who moved out from the area.

Data on individual cumulative exposure and lung cancer incidence will then be analysed by appropriate statistical methods, taking account of the factors which can influence the lung cancer incidence such as age, gender, smoking habits, exposure to another noxae in occupation etc. For the analysis, regional and national data on age and gender specific lung cancer mortality as well as an internal control group approach will be used. An evaluation of the collected information by the nested-in case control approach, which would enable better comparison with the data from other epidemiological indoor radon studies, prevailingly of the case- control type, will be applied too.

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HISTOLOGICAL TYPES OF LUNG CANCER IN URANIUM MINERS

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INTRODUCTION

Long term exposure to radon and its progeny is one of the most important health problems. Epidemiological studies have demonstrated that exposure of miners to radon in a mine atmosphere can cause lung cancer. It has been recognized that lung cancer risk in uranium miners is associated with increased incidence of certain histological types, especially epidermoid and small cell type (1, 2). Previous results showed that the basic dependence of the relative risk for the two main histological types is linear with cumulative exposure (3). However, there is a suggestion that time and age modifiers of the dependence may be different for the two types. The aim of the study was firstly to verify the assumed differences in incidence of histological types of lung cancer for the studied cohort and general population and secondly to characterize the relation of histological types specific incidence to different exposure patterns.

The study is based on data of the oldest Czechoslovak cohort, which belongs among the largest ones with the longest follow-up. The cohort includes 4320 former uranium miners who started working in uranium mines in West Bohemia in the period 1948-59, the work lasted more than 4 years (4, 5). The study method is long-term prospective follow-up. The cohort was divided in two groups according to exposure rate pattern in order analyze the impact of exposure conditions on main histological types incidence. The exposure rate never exceeded the level of 5 WL since the third year of exposure in the group 1 while the exposure rate exceeded at least once the level of 5 WL in group 2. The first two years of exposure had not been taken into consideration in order to confirm the hypothesis about the inhibitory effect of high exposure rate levels. Consequently, it could influence lung cancer risk level caused by previous exposures.

The basic method of the study is an analysis of observed frequencies of lung cancer and their histological type specification in relation to expected standardized frequencies (age and time period). In accordance to WHO classification and from previous data analysis (6), lung cancer cases were divided into four groups:

1 - epidermoid
2 - small cell
3 - adenocarcinoma
4 - other histological types

RESULTS

A total of 705 lung cancer cases were recorded by 31 December 1990. The ratio of observed and expected numbers was 5.11. Morphological diagnoses were available in 458 cases. This represents histological type specification in 65% of lung cancer cases in the cohort as shown in Table 1.

Tab. 1: Distribution by histological types

Histological type	Code	Cases	%
no information	0	201	28.5
epidermoid	1	173	24.5
small cell	2	185	26.2
adenocarcinoma	3	31	4.4
other types	4	29	4.1
unspecified	5	40	5.7
no material	9	46	6.5
Total	-	705	100.0

Tab. 2: Age and histological type specific mortality

Age	Small cell		Epidermoid	
	O	O/E	O	O/E
- 44	24	35.02	7	10.64
45 - 54	71	11.97	39	7.20
55 - 64	65	4.27	81	5.39
65 - 84	25	2.65	46	3.79
Total	185	5.91	173	5.20

Histological types could not been specified in 41% of cases. In 40 cases the morphological findings are available (code 5), but it is impossible to specify any reliably histological type from these. In 46 cases, no material is available to specify the histological type (code 9),althought the diagnosis of lung cancer is correct.

Table 2 summarizes observed (O) and expected (E) death frequencies in dependence on attained age for both main histological types. The decrease of relative risk with age is apparent in both histological types. A more detailed analysis shows that decrease of relative risk in epidermoid type is less rapid than in small cell type. From this fact follows that risk of lung cancer death will be significantly lower in small cell carcinoma than in epidermoid one in higher age of former miners. This is the contrary of situation in younger age groups of miners.

Table 3 demonstrates dependence of relative risk on cumulated exposure (lagged by 5 years) for the two types. The relative risks show similar values in corresponding exposure categories.

Tab. 3: Exposure and histological type specific relative risk

Exposure WLM	Small cell		Epidermoid	
	O	O/E	O	O/E
0 - 99	18	2.72	17	2.42
100 - 199	69	5.03	62	4.26
200 - 299	48	8.36	42	6.98
300 - 399	21	7.72	21	7.07
400 -	29	11.68	31	11.52

RISK MODELS

Epidemiological studies have demonstrated that time since exposure and attained age considerably influence excess relative risk of lung cancer (7). In this study, the modifying effect of three factors on exposure - risk relationship was examined (Table 4):

(1) time since exposure (TSE),
(2) age at exposure (AM),
(3) time pattern (TP).

Tab. 4: Relative risk models for histological types

ERR/WLM	Epidermoid	Small cell
	0.066	0.329
TSE		
5 - 14	1	1
15 - 21	0.926	0.110
25 - 34	0.150	0.014
35 -	-0.016	0.008
AM	-0.0338	-0.0878
TP	0.708	0.534

The estimated values of model parameters for the two histological types show important differences (Tab.4). The ERR/WLM was 6.6% in epidermoid and 32.9% in small cell type (5 times higher). The effect of time pattern was found significant in small cell type and not in epidermoid type, which confirm hypothesis about parallel inhibitory effect of alpha radiation on small cell type of lung cancer. Significant negative values of the AM parameter confirm the assumed decrease of lung cancer risk with age at exposure. This trend is more pronounced in small cell type.

SUMMARY

The recent results of the study confirme the linear dependence of exposure and relative risk, decrease of alpha radiation effect with time since exposure and decrease of effect in older age categories.

More detailed analysis of exposure - effect relationship in main histological types of lung cancer confirmes hypothesis formulated by J. Ševc about parallel inhibitory effect of alpha radiation in higher exposure rates which followed after period of initiation of malignant process of bronchial epithelium cells. This phenomenon is observed in small cell type of lung cancer. These conclusions are based on application of proportional risk model.

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DECEASES BY RESPIRATORY CANCER AMONG URANIUM MINERS

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ABSTRACT

The deceases by respiratory cancer(bronchopulmonary and laryngeal) arised among miners from two uranium mines for the period 1960-1993 were studied.The control groups were formed by the male population deceased by the same cause in the native localities of the miners.

It resulted an enhanced occurrence risk of lung cancer among miners in comparison to the unexpected controls.

INTRODUCTION

Bronchopulmonary cancer is a specific occupational desease for uranium miners,the cause being the protracted exposure to high levels of alpha active radon daughters; depending on the magnitude of the pulmonary doses the desease occuring after 10-15-20 years from the beginning of exposure.(1-3)

For the knowledge of sicken risk by cancer of respiratory tract(bronchopulmonary and laryngeal) among uranium miners it was conducted an epidemiological study for two uranium mines from Romania for a 34 years period(1960-1993).

METHODOLOGY

The studied period(1960-1993) was established on the basis that a longer study period offers a higher certainty to find respiratory cancer among uranium miners.Were chosen out the localities that have provided the working force for the two mines,after that,from official records, were extracted the deaths by respiratory cancer among the adult male population.

Then fallowed the identification of retired miners from personnel records.So,it could be established the occupational route, the level of cumulative exposure to radon(WLM) and the smoking habit for each miner deceased due to respiratory cancer.

The controls were made up by unexposed adult male population from the native localities of the miners.

The statistical computations had in view the incidence of deaths/10.000 person-year, the expectancy, the observed/expected cases ratio and the relative risk coefficient.

The interpretation of data was made depending on age, years of underground service, cumulative exposure to radon and smoking habit for the miners, but on age for controls.

RESULTS AND DISCUSSION

The study group for Mine I was of 28,796 persons from 22 localities and for Mine II was of 8,164 persons from 2 localities.

In the period 1960-1993, in the areas of Mine I and Mine II were observed 103 deaths by respiratory cancer among miners and 480 among controls, respectively 34 among miners and 130 among controls.

As regarding the anatomical position in the respiratory tract, as for miner as for controls, obviously prevailed the bronchopulmonary cancer (over 80%).

The incidence of deaths by respiratory cancer is significantly increased among miners in comparison to controls ($p<0,001$) : almost 6 times higher for Mine I and almost 3 time higher for Mine II (Figure 1)

Incidence/10,000 person-year

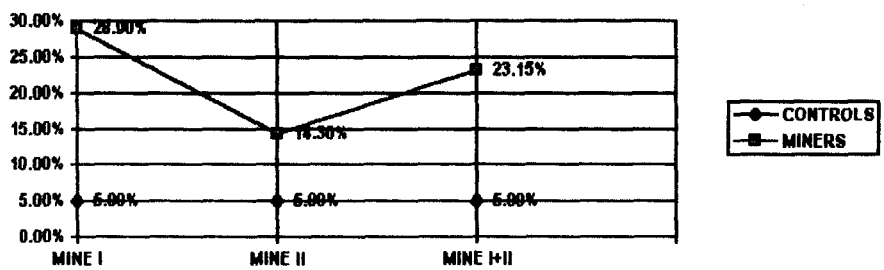

Figure 1. The incidence of deaths by respiratory cancer for the miners from Mine I and Mine II in comparison to the controls(1960-1993)

The risk of being taking ill by respiratory cancer, as expressed by the observed/expected cases ratio, as well as by the relative risk coefficient, it is significantly higher for miners in comparison to controls (Mine I:5,78>1.00 and Mine II : 2.80>1.00).

The average age at death for the miners from Mine I and Mine II(57.5 and 53.8years,respectively) is significantly lower($p<0,001$) in comparison to the controls(61.7 years and 61.3 years,respectively).(Figure 2)

Figure 2. Mean age at the date of death by respiratory cancer for the miners from Mine I and Mine II in comparison to controls(1960-1993)

The average years of underground service for the deceased miners from the two mines are 15 years for Mine I and 12 years for Mine II, predominanting the service in uranium mines: 11.3 years for Mine I and 11.1 for mine II.

As for the cumulative exposure to radon, deceases by respiratory cancer show enhanced ratios among miners having a cumulative exposure over 120WLM(Figure 3).The average exposure of the 103 deceased miners form Mine I was 401.1WLM and that for the 34 deceased miners from Mine II was 119WLM.

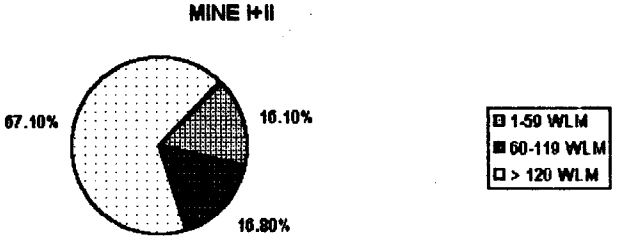

Figure 3.Distribution of deaths for the miners from Mine I and II versus cumulated exposure to radon(1960-1993)

The information regarding smoking habits of Mine I miners allowed to correlate the frequency of deaths and smoking habits, being proved the sinergetic effect of exposure to radon and smoking. The risk coefficient due to smoking is more obvious if the deaths are discussed in comparison to the smoking index(4)(no. of daily smoked cigarettes multiplied by no. of years of smoking):the proportion of respiratory cancer cases is 62.4% for smokers, with an index higher than 500, in comparison to only 4.3% for those having an index under 150.

The latency period calculated for miners deceased by respiratory cancer averaged to 26.7 and 21.0 years for Mine I and Mine II, respectively. The significantly increased incidence of deaths among miners, the obsviously lower age at death in comparison to controls, taking into account, also, the level of occupational exposure prove the occupationl character of respiratory cancer among uranium miners.

CONCLUSIONS

The epidemiological study of deaths by respiratory cancer among uranium miners proved that the increased incidence of deaths, much higher in comparison to controls(5.7 times for Mine I and 2.8 times for Mine II), is due to cumulative exposure to radon and its daughters, and it confirmed the occupational

character of deceases. Also, it can not be neglected the sinergetic effect of smoking, over 90% from the deceased miners being smokers.

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DOSE AND RISK ASSESSMENT FROM RADON IN THE URAL

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INTRODUCTION

The radiation situation in the South and Middle Urals is quite complicated due to two main reasons:

- High radon exposure levels derived from the geological structure of Ural region (relatively high concentrations of U and Th in lithosphere, ground waters with high Rn concentration, increased permeability areas etc.).
- Consequences of radiation accident at "Mayak" nuclear plant in 1957. As a result of this accident a considerable territory of Chelyabinsk and Sverdlovsk regions was polluted by mixture of nuclear fission products $^{90}Sr+^{90}Y$ - 5.4%, $^{95}Zr+^{95}Nb$ - 24.9%, $^{144}Ce+^{144}Pr$- 66%, $^{106}Ru+^{106}Rh$ - 3.7%, ^{137}Cs - 0.036% (the so called East Ural Radioactive Trace - EURT).

The main purpose of this paper is to compare the radiation risks due to equivalent doses in different organs and effective doses resulting from radiation pollution and radon and thoron daughters exposure.

METHODS OF MEASURING Rn AND Tn CONCENTRATIONS

The survey was made by the following two methods: aspiration technique of measuring of equivalent equilibrium concentration of radon and thoron and determination of season mean concentrations of Rn by applying the nuclear track nitrocellulose detectors. The cellulose nitrate films were used as sensitive materials. The cellulose nitrate detector was placed in special plastic chamber, provided with silicon rubber filter preventing the penetration of ^{220}Rn and radon daughters into chamber space. The measuring chambers with detectors were exposed for 1-3 months. Detectors were etched in 6N NaOH solution after exposition and track density was determined by spark counter.

The equilibrium equivalent concentration EEC of radon and thoron was determined by Markov modified method [1].of measuring filter α-activity after deposition of radioactive aerosols on it. For measuring of equivalent equilibrium concentration of radon and thoron we used both serial alpha-radiometers and devises developed at our Institute together with the Urals State Technical University. The results obtained by both methods correlate well.

DOSE ASSESSMENT

The intake of radionuclides for EURT zone due to inhalation and food chains was estimated by the official Methodical Recommendations [2]. Equivalent doses for different organs and effective doses were calculated using ICRP Publication 56 data [3].

The data on average annual gaseous radon activity (C_{Rn}), equilibrium equivalent concentration of thoron (EEC_{Tn}) and annual exposure effective doses (E_{Rn}, E_{Tn}) in various districts for village types of buildings are shown in table 1. Doses due to radon and thoron daughters exposure were calculated using ICRP Publications 50 and 65 [4,5].

During our investigation we obtained that the very high concentrations of thoron daughters are characteristic for the Urals region. Average world value of annual effective exposure dose due to thoron is listed in Report of UNSCEAR, 1993 [6] and is equal to 0.067 mSv/a. So, we can see that the thoron exposure in the Urals is an order of magnitude higher than the average world values.

Table 1

Average annual gaseous radon activity, equilibrium equivalent concentration of thoron and annual exposure effective doses in Sverdlovsk region

District	C_{Rn}, Bq/m^3	E_{Rn}, mSv/a	EEC_{Tn} Bq/m^3	E_{Tn}, mSv/a
Bogdanovich	117.1	2.1	3.7	1.3
Kamensk	110.7	2.0	2.6	0.91
Rezh	231.6	4.1	3.8	1.3

RISKS ASSESSMENT

During our work we used some assumptions and risk projection models :

1. Absolute risk model for risk assessment of leukemia [7].
2. Constant relative risk model for assessment of total cancer risk excluding leukemia and some internal organs [7].
3. Modified relative risk model (Jakobi model) for risk assessment of lung cancer due to radon and thoron daughters exposure [5].
4. For all risk projection models we assumed linear no-threshold dose response.

Risk comparing was made for rural districts of Sverdlovsk region: Kamensk and Bogdanovich which were polluted during the radiation accident on "Mayak" nuclear plant in 1957. For considered in this paper territories the average surface contamination by ^{90}Sr was ~3 Ci/km^2. The Rezh district was chosen because it's the district with maximum radon exposure rate in Sverdlovsk region. Statistical data on survival probability, mean life expectancy (L_m=59.7, L_f=71.4 years) and baseline cancer risks typical for Sverdlovsk region were used for risks assessment. Relatively low regional survival probability and mean life expectancy decreases radiation cancer risks in comparison with thatin the developed countries.

The results of radiation risk assessment and the attributable excess cancer death frequency F_r are listed in table 2. For individual organs the equivalent doses are shown, for total cancer risk assessment - the effective doses without contribution of red bone marrow. For radon and thoron exposure the average annual effective doses are shown.

As one can see from the table 2 the lifetime radiation risks due to radon exposure in the village houses in Sverdlovsk region are comparable to or exceed the radiation risks due to serious radiation accident.

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Table 2

Comparison of radiation risks due to radon exposure and consequences of accident on "Mayak" nuclear plant in 1957 (surface contamination by ^{90}Sr 3 Ci/km^2)

Organ	Age of exposure	Dose (mSv)	Baseline cancer risk		Radiation risk		Attributable excess frequency F_r (10^{-5} a^{-1})	
			Males	Females	Males	Females	Males	Females
All cancers excluding leukemia	0-9	69,5	0,140	0,121	0,0059	0,0089	9,84	12,52
	10-19	47,9	0,140	0,121	0,0025	0,0038	4,25	5,29
	20-29	41,2	0,140	0,121	0,0020	0,0028	3,34	3,96
	30-39	41,2	0,135	0,119	0,0009	0,0012	1,50	1,69
Leukemia	0-9	125,0	0,0022	0,0029	0,00091	0,00075	1,52	1,05
	10-19	164,7			0,00059	0,00033	0,99	0,46
	20-29	80,9			0,00056	0,00037	0,93	0,52
	30-39	80,9			0,00071	0,00027	1,19	0,38
Stomach	0-9	15,0	0,024	0,020	0,00019	0,00031	0,32	0,43
	10-19	10,9			0,000085	0,00014	0,14	0,19
	20-29	9,8			0,000067	0,00011	0,11	0,15
	30-39	9,8			0,000028	0,000047	0,047	0,066
Small intestine	0-9	44,1	0,00036	0,00029	0,0000084	0,000013	0,014	0,018
	10-19	25,8			0,0000030	0,0000047	0,005	0,007
	20-29	20,5			0,0000021	0,0000032	0,004	0,005
	30-39	20,5			0,0000009	0,0000014	0,001	0,002
Upper large intestine	0-9	221,9	0,0072	0,0123	0,00085	0,0028	1,42	3,93
	10-19	132,3			0,00031	0,00103	0,52	1,45
	20-29	97,3			0,00020	0,00066	0,34	0,93
	30-39	97,3			0,00008	0,00029	0,14	0,40
Lower large intestine	0-9	635,3	0,0052	0,0068	0,0018	0,0044	2,93	6,22
	10-19	385,3			0,0007	0,0017	1,09	2,32
	20-29	283,7			0,0004	0,0011	0,70	1,50
	30-39	283,7			0,0002	0,0005	0,30	0,65
Liver	0-9	14,8	0,011	0,012	0,000087	0,00019	0,15	0,27
	10-19	13,5			0,000049	0,00011	0,082	0,15
	20-29	13,6			0,000043	0,000094	0,073	0,13
	30-39	13,6			0,000018	0,000041	0,031	0,057
Lungs	0-9	47,0	0,052	0,0097	0,00130	0,00047	2,17	0,66
	10-19	40,1			0,00068	0,00025	1,14	0,35
	20-29	32,9			0,00049	0,00018	0,82	0,25
	30-39	32,9			0,00021	0,00008	0,34	0,11
Lifetime radon exposure (EURT zone)		3.1 mSv/a	0,052	0,0097	0.013	0.0077	21.2	10.7
Lifetime radon exposure (Rezh district)		5.4 mSv/a	0,052	0,0097	0,022	0,013	36.2	18.4

ESTIMATION OF INFLUENCE OF RADIOACTIVE FACTOR ON CANCER RISK AMONG INDUSTRIAL CITY DWELLERS

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The estimation of the role of chronic radioactive exposure (ChRE) in development of malignant neoplasms under the conditions of constatnt exposure of population along with radiation to other carcinogenic factors is a complex epidemiological problem. Such situiation is typical for city Kamensk-Uralsky (Middle Urals, Russia), where some enterprises intensively polluting the environment with carcinogenic substances are located (i.e. the aluminium-smelting plant). In 1957 the city was subjected to the influence of radioactive fallout as a result of an accident at PA "Mayak" and forming the East Urals Radioactive Trace (EURT). The most contaminated was the north-western part of the city, where the contamination density was about 4 Ci/Sq.km by Sr-90, at the rest of the territory it was about 1 Ci per Sq.km. Cancer incidence in Kamensk-Uralsky during 1960-1994 has increased 1,8 fold, but it can hardly be connected only with the influence of radiation.

The city has a population of 210 000, which is served by oncological centre. The materials of this centre made it possible to form sufficient groups of cases for our study and to obtain all the necessary information about each of them. The base cohort for the information of control groups was the part of the population that had never had any neoplasms diagnosed.

Many cancer risk factors are well known. That's why the influence of one of them (in this particular study - radioactive exposure) may be estimated only in the result of multifactorial analysis. For this analysis we used the pattern recognition methods, that have proved their effectiveness in many other medico-ecological studies(1,2,etc.).

Some investigations were held aiming at estimation the role of radioactive factor in development of cancers of three most widespread sites: lung,stomach and breast. Information about all known cancer risk factors for both "case" and "control" groups was collected. The information included data concerning age, sex, nationality, smoking habit, alcohol abuse, diet, residing in one of the 4 areas differing in intensity of non-radiational ecological contamination, residing close to a major industrial atmospheric polluter, occupation, duration of occupational exposure to carcinogens, characteristic of housing conditions (building materials, plastic flooring in the house, gas stove), chronic diseases, family cancer history, predisposition phenotype evaluated by the peculiarities of dermatoglyphic pattern (palm and finger prints) on the basis of previously elaborated criteria, length and strength of radioactive exposure.

Diversity and lack of reliable information on radiological situation in EURT zone in Sverdlovsk region after the accident gave no opportunity to obtain the reliable retrospective data concerning the individual radiation doses for the dwellers of this zone. ChRE is estimated by two indirect parameters: (a)the fact of residing in the north-western (most contaminated) part of the city during the years of maximal radioactivity of the territory (1957-62) and (b) total length of residing in the EURT zone.

In our breast cancer studies the information of risk factors was completed by the data about gynecological status of women. In lung cancer studies the analysis embraced in total 18 factors. Basing on this results it was shown that this information

is sufficient enough to reliable discrimination between the "case" and "control" patterns. Along with this the quantitative estimation of each factor influence was given (Table 1). The results of similar analysis of complex of 22 risk factors for breast cancer and 20 risk factors for gastric cancer are shown in Tables 2 and 3.

At analysis of studied cancer risk factors we noted first of all the expected character of their interrelation with the disease. For example , risk of cancer development was higher in smokers, in workers of aluminium smelting plant, it has increased with age ,etc. This particular coincidence of obtained results with the already proved influence of some biological and environmental factors on malignant neoplasms development made it possible to estimate adequately the role of ChRE, that proved to be quite sufficient. Moreover, it was noted, that risk of development of cancer of all three studied sites increases with both duration of residing at the EURT territory and, particularly, with residing in the most radioactively contaminated zone during the years of maximal intensity of radioactive contamination (1957-62).Thus,applying the special methods of analytical epidemiology and multifactorial analysis gave us an opportunity to receive the proofs of the chronic radioactive exposure influence on cancer incidence among the dwellers of EURT zone.

Table I

COMPARATIVE DEGREE OF INFLUENCE OF THE MOST SIGNIFICANT LUNG CANCER RISK FACTORS (for population of Kamensk-Uralsky)

Rank	Name of the feature	Comparative informativity (relative units)
1	Sex	1.000
2	Smoking	0.749
3	Occupation	0.360
4	Building material	0.319
5	Residing in the zone of maximal radioactive contamination in 1957-1962	0.247
6	Age	0.245
7	Alcohole abuse	0.207
8	Genetic predisposition	0.152
9	Chronic obstructive lung disease	0.152
10	Residing close to a major industrial atmospheric polluter	0.136
11	Plastic flooring in the house	0.098
12	Total length of residence in EURT zone	0.087

Table 2.

COMPARATIVE DEGREE OF INFLUENCE OF THE MOST SIGNIFICANT BREAST CANCER RISK FACTORS (for population of Kamensk-Uralsky)

Rank	Name of the feature	Comparative informativity (relative units)
1	Menopause	1.000
2	Alcohol abuse	0.844
3	Residing in the of zone of maximal radioactive contamination in 1957-62	0.676
4	Building material	0.581
5	Residing in the zones of various degrees of non-radioactive technogenous pollution	0.516
6	Total length of living inthe EURT zone	0.491
7	Genetic predisposition	0.339
8	Gas-stove	0.339
9	Occupation	0.248
10	Age	0.245
11	Family cancer history	0.216
12	Plastic flooring in house	0.175

Table 3

COMPARATIVE DEGREE OF INFLUENCE OF THE MOST SIGNIFICANT GASTRIC CANCER RISK FACTORS (for population of Kamensk-Uralsky)

Rank	Name of the feature	Comparative informativity (relative units)
1	Sex	1.000
2	Occupation	0.498
3	Gastric diseases (gastritis,ulcer)	0.460
4	Smoking	0.451
5	Genetic predisposition	0.364
6	Alcohol abuse	0.326
7	Age	0.249
8	Residing close to a major industrial atmospheric polluter	0.231
9	Residing in the zones with various degrees of non-radioactive technogenic pollution	0.211
10	Gas-stove	0.195
11	Total length of living in the EURT zone	0.185
12	Residing in the of zone of maximal radioactive contamination in 1957-1962	0.139

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DOCUMENTATION OF OCCUPATIONAL ACCIDENTS AND DISEASES BY IONISING RADIATION IN GERMANY

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The investigation of accidents and diseases, caused by ionising radiation, is important not only from the more technical point of „accident prevention" but even more from medical and biological aspects in investigating human response to exposure of ionising radiation.
As industrial use of radiation sources grew rapidly from the 70th on, and so did the workforce handling the sources, it is evident, that the amount of occupational exposed persons who will now come to an age where (spontaneous) cancer diseases become manifest is also growing. There will be a certain number of people who will declare this disease being caused by the occupational exposure. Investigation and decision in such cases have to be performed in short time and will be enhanced by a good experience from former cases.
In Germany the workman's compensation institutions (Industrial Injury Insurance Institutes [IIII] - Berufsgenossenschaften) are responsible for all questions of medical care, rehabilitation and compensation of occupational accidents and diseases. They decided to provide a registry for all cases which came to investigation. Since 1980 the institute of radiation protection which was founded in the late 70th by two of this IIII (Berufsgenossenschaft der Feinmechanik und Elektrotechnik and Berufsgenossenschaft der Chemischen Industrie) is performing the data-collection. The quality of the data depends of course on the details which are reported to the Institute in each case and on the criteria, when at all a case should be reported to the registry. Unfortunately it was not possible to edit general reporting criteria. Each Industrial Injury Insurance Institute in Germany (there are 36 of them) decides by itself about the reporting criteria Of course this leads to a loss of information as is shown later.
For statistical purposes the numbers of occupational accidents and diseases, categorised to the causing influence, which have been announced for compensating at the IIII, are reported to the ministry of health and are edited in the annual „Accident Prevention report" of the federal government.
A more detailed view to this numbers can be derived from the annual report of the IIII´s association.
Figure 1 gives a annual distribution for the last ten years of all cases where overexposure was announced as reason of injury or disease and also shows how many cases left, where the investigation gave reason to certify this relation and where the injury or disease was acknowledged as occupational.

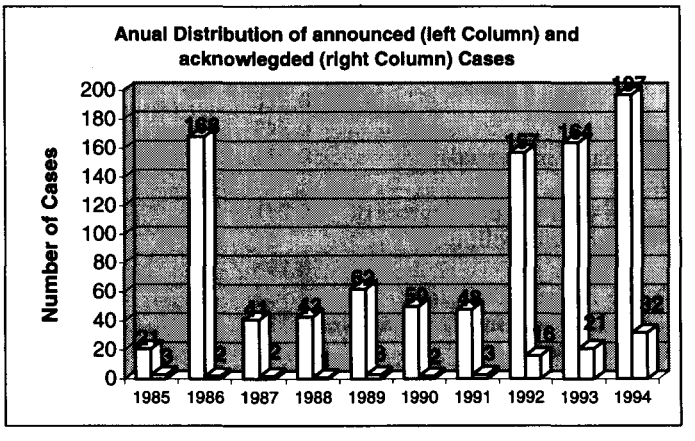

Fig.1 Annual distribution (1985-1994) of injuries and diseases which have been announced as occupational disease due to ionising radiation and which have been acknowledged as those.
(Without the numbers of the IIII of the mining industry - see text)

It becomes obvious, that the number of diseases which proofed to bee radiation induced is much smaller than the announced number. The peak at 1986 results from German workers in the former SU near Tschernobyl whose registration was done in precaution and without having been really exposed. Although the numbers of the IIII of the mining industry are taken off, the increase from 1992 on results from miners of the uranium mining in the former DDR. According to special regulations in Germany a certain number of those compensations have to be done by other IIII.

As can be seen from Fig.1. a total amount of 951 cases of announced diseases (and injuries) have been reported in Germany from 1985 to 1994. Not included in this number are the cases announced to the IIII of the mining industry. Further not included are the incidents which had happened in the former DDR.
Using the same criteria, we will find only documents of 361 cases in our registry. Only 37% of all cases are reported by the IIII to our institute.
An enhancement was brought in by updating the registry with the cases from the former DDR. The much more centralised system of occupational health and safety including radiation protection of the workers promoted the centralised collecting of the documents.
Including this cases and extending time to 1995 makes it possible to look to total of 517 cases.
As the health situation of the east German uranium miners is subject of special investigation programs, all of this cases will be omitted in the further analysis of this paper.
If doing so, 371 documents are remaining. 102 of them are connected to Tschernobyl as described previous. So there are left 269 files in the registry which indicate the situation in industry and health service. A view to the annual distribution of the year of announcement is done in fig. 2.

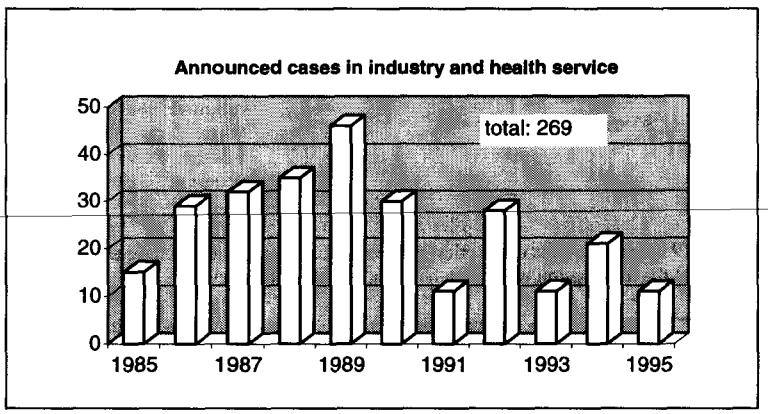

Fig.2 Annual distribution (1985-1995) of injuries and diseases which have been announced as occupational caused by ionising radiation. All cases connected to mining and the registered cases in connection with Tschernobyl (see text) have been omitted.

After a continuos increase until 1989, which corresponds to the increasing number of radiation workers, the numbers seem to decrease until today much faster than the number of radiation workers decreases (today about 340,000 persons). We presume, that this decrease is caused simply by rearranging the administration structures of the IIII after the German unification in October 1990 which led to a certain neglecting of reporting. It will be one of the next investigations of the registry to check this phenomenon.
The next step of analysis is to get a survey of the branches where the cases are coming from. This is indicated in fig. 3.

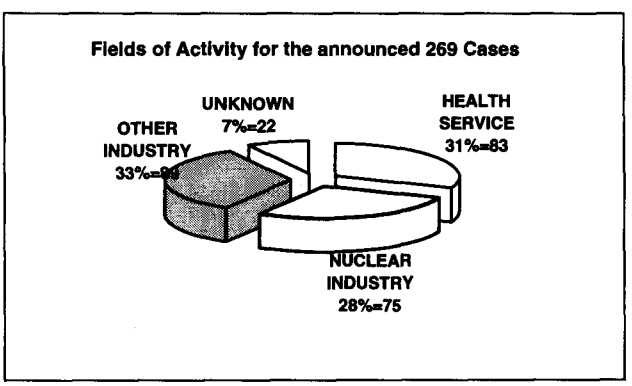

Fig. 3 Distribution of the left 269 announced cases to the reporting branches

According to missing reporting criteria, only in 165 of the 269 files we find a clear indication of the state of the case. That means, that 104 cases have been reported at a certain state of investigation, but the reported information was never updated. So we are not able to classify them, we even do not know, whether they are still in work, whether they are acknowledged or refused.

Finally we can work through the 165 remaining cases. Figure 4 shows that in 110 of this a decision was made: 19 have been acknowledged and 91 have been refused. In 55 cases work is still in progress.

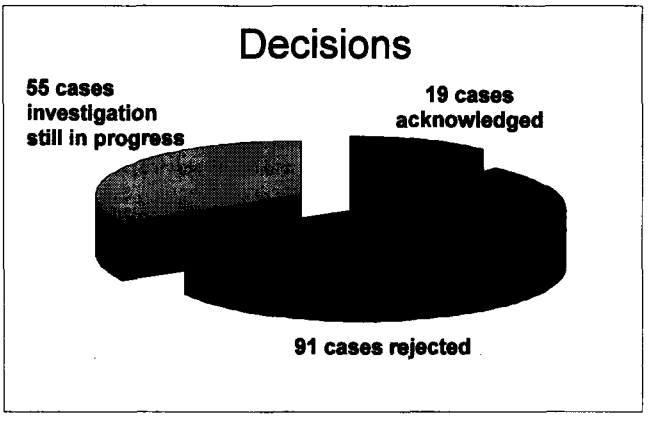

Fig.4 Only in 165 of the reported files is indicated which decision was made or whether the work is still in progress.

Interest is focused now to those 19 cases where the connection between radiation exposure and disease was accepted. The following tables list the information which can be taken out of the registry.

# of cases	Exposure
3	acute
15	chronicle
1	unknown

As can be noticed from this table, the by far most accepted diseases are caused by chronic exposure. 13 of these belong to the health service and concerns to physicians or medical personnel who used to apply X-rays - often in the 50th and 60th without adequate radiation protection measures. Two of the acute cases come from the non-destructive material testing and one is registered in health service again.

Information about diagnoses is sometimes poor and is given in the next table. Unfortunately, the dose information in the registry is quite insufficient. Even in cases of acute radiation injuries sometimes effective dose is reported but organ dose or skin dose is omitted. Sometimes a dose value is given but is obviously wrong. In other cases the reported unit of a dose-value does not fit the rest of the information.

# of cases	group of diagnoses/reported doses
2 acute	X-ray-burn; ulcers
4 chronic	cancer of homeopathic system/515 mSv/-/1500mSv/1000mSv all eff.
1 chronic	cancer of thyroid/1019mSv eff.
1 chronic	cancer of the skin/150 Gy org. (thumb)
9 chronic	unknown
1 acute	unknown

Although a lot of information can be derived from the registry, it is shown in this paper, that a reanalyses of the original data documents is necessary for enhancing the quality of the medical and dosimetric data.

Epidemiological Studies Using the National Dose Registry of Canada

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Introduction

The National Dose Registry(NDR) is a centralised record keeping system for occupational doses of all monitored workers in Canada(1,2). Its foundation is Canada's National Dosimetry Service which was started by Health Canada in 1951. Since that time the Registry has expanded to receive and include records from nuclear power stations which perform their own dosimetry, uranium mines, and other dosimetry processors. It currently includes records for 520,000 workers some of which date back to the early 1950's. The workers are classified into 80 different occupational categories(e.g. radiologist, reactor mechanical maintenance)and the employing organizations into 14 different categories(e.g. hospital, reactor). The Registry resides on an HP3000 Model iX computer. It contains approximately 3.4 million annual summary dose records and an estimated 16 million discrete transaction or dosimeter records.

In addition to its use in epidemiological studies the Registry is also used to: i)assist the regulatory authorities in their control of occupational radiation exposures; ii)monitor long term dose trends and; iii) provide dose histories for use in radiation exposure planning and for use in compensation claims. The NDR is involved in a number of epidemiological studies on the long term effects of radiation. These will be described separately below.

Paternal Radiation Exposure and Childhood Leukaemia

This case control study(4) was undertaken in collaboration with the Ontario Cancer Treatment and Research Foundation who were the initiators and lead agency in the study. The study was prompted by findings which suggested a link between paternal radiation exposure and childhood leukaemia(3). Funding was provided by the Atomic Energy Control Board of Canada and Health Canada.

Cases were children ages 0-14 , living within the proximity of a nuclear facility in the province of Ontario who died from leukaemia or were diagnosed between 1950 and 1988. The facilities covered in the study are listed below.

Type of Facility	Location	Dates of Operation
Research and development	Chalk River	1944 - Present
Uranium Mining/Milling	Elliot Lake	1954 - Present
Uranium Refining	Port Hope	1934 - Present
Nuclear Generating Station	Rolphton	1962 - 1971
Nuclear Generating Station	Bruce	1967 - Present
Nuclear Generating Station	Pickering	1971 - Present

Out of a total of 3,366 childhood leukaemia cases in the province 112 were from within the proximity of one of the nuclear facilities. 8 controls per case were selected, matching on date of birth(within 3 months of case) and the region where the mother lived at time of birth.

Parents' names were taken from the children's birth certificates and a computer file of parents names was created. The names in this file were then linked to the records of the National Dose Registry in order to obtain any dose record information. Linkage was by computer and involved weighted linkage based on agreement of identifiers.Out of the 1002 fathers of the children(cases and controls), 95 linked to dose records in the Registry(84 controls and 11 cases) .

Dose records included dose from external gamma and neutron doses as well as internal dose from tritium and Radon daughters.Gamma doseś for Uranium miners were available back to 1981. Prior to that date dose estimates were used based on occupancy factors and mine characteristics.

The overall odds ratio was 1.07 with the 95 % confidence interval (95%CI)from 0.22 to 2.29. The findings thus do not support the hypothesis that paternal radiation exposure prior to conception is related to childhood leukaemia. Further details are shown in the table below.

Category	# Cases	# Controls	Odds Ratio(95%CI)
Never Exposed	101	806	1.00
Reactor Worker	4	48	0.64(0.22-1.86)
Uranium Miner	5	26	7.27(0.59-88.7)
Other/Unknown	2	10	1.59(0.34-7.44)
All Linked	112	890	1.07(0.50-2.29)

Cohort Mortality Study(1987)

This large scale cohort study(5) was initiated by Health Canada in the mid 1980's. Funding was by Health Canada, the Atomic Energy Control Board, and Statistics Canada. The cohort consisted of 206,426 workers(105,346 males, 101,168 females) in Canada who were monitored for external radiation between 1951 and 1983, and who had at least a surname, first given name, sex, and year of birth in their records. (This was necessary to ensure the successful linkage of the records). The total collective dose for the cohort was 12.9 person-Sv. The median attained age(as of 31, December 1987) was 41 years for males and 36 for females indicating that this is a relatively young cohort.

This cohort was linked to the mortality data base maintained by Statistics Canada using probabilistic record linkage techniques(6). Records in both files were grouped into phonetically equivalent "pockets". The identifiers of records in corresponding pockets of each file were then compared and assigned a weight based on the agreement between the identifiers. Records above a certain threshold weight were then deemed to be linked and belong to the same individual. This threshold weight was set at the point where the false positive link rate was equal to the false negative rate. The false links were identified by reviewing a sample of linked records and comparing with subsidiary information such as occupation and place of last employment.

Through this process a total of 5,430 deaths were identified. Cause of death was taken from the mortality data base and coded to correspond to the 9th revision of the ICD codes(7). Standardised Mortality Ratios(SMR's) relative to the Canadian population were calculated. Trend tests for mortality were undertaken in which the dose was lagged by 2 years for leukaemia and 10 years for solid tumours. A relative risk regression model was used to determine the excess relative risk(ERR).

A selection of the results is summarized in the table below. The overall SMR was 0.59 for males and 0.62 for females indicating a strong "healthy worker effect". While for males and females there were no specific sites which showed a significant trend($p<0.05$) or excess relative risk, the p-value for all cancers(males) was significant(0.041) as was the ERR(3.0). However this appears to be a weak association which disappears at the 95% confidence level.

Summary of Results - Males

Cause of Death	# Obs. Deaths	SMR	Dose Trend p value	ERR(% per 10mSv) (90% Conf. Int.)
Prostate	71	0.88	0.708	2.6(-5.0,10.1)
Multiple myeloma	17	0.73	0.802	6.5(-17.0,29.9)
Leukaemia	56	0.77	0.855	0.1(-4.7,5.0)
Leukaemia(Ex CLL)	46	0.83	0.733	0.4(-4.9,5.7)
Myeloid Leukaemia	27	0.88	0.379	1.9(-6.2,10.1)
All Cancers	1136	0.68	0.041	3.0(1.1,4.9)

Summary of Results(Cont'd) - Females

Cause of Death	# Obs. Deaths	SMR	Dose Trend p value	ERR(% per 10mSv) (90% Conf. Int.)
Multiple myeloma	5	0.69	0.875	0.2(-70.6,71.0)
Leukaemia	26	0.85	0.835	-0.2(-19.4,19.1)
Leukaemia(Ex CLL)	23	0.87	0.837	-0.2(-20.3,20.0)
Myeloid Leukaemia	11	0.71	0.858	-0.2(-20.5,20.2)
All Cancers	495	0.71	0.756	1.5(-3.3,6.3)

Cohort Cancer Incidence Study(1987)

A cancer incidence study using the cohort(1987) from the mortality study is currently in progress. This study is being funded by Health Canada. The cohort has been linked, using the same techniques outlined above, to a Cancer Incidence Reporting System maintained by Statistics Canada. A file of linked records for an estimated 5,252 workers from the cohort reported as having cancer has been created and is currently being analyzed. First analysis is expected to be completed in 1996.

Cohort Mortality Study(1994)

An updated NDR cohort has been created in preparation for updating the cohort(1987) study. The study is being funded by Health Canada. This study will include workers monitored between 1984 and 1994 and will also include uranium miners who were not included in the original study. In addition, a number of individuals who were left out of the original study because of lack of adequate identifiers have been fully identified and included in the updated cohort. The new cohort is estimated to contain 380,877 workers with a collective dose of 18,100 person-Sv.

Contribution to the IARC study on Nuclear Industry Workers.

Canada is also participating in the International Agency for Research On Cancer study of nuclear workers(8). A subcohort of the 1994 mortality study will be created for separate analysis and as Canada's contribution to the international study. It is estimated that the cohort of nuclear industry workers will consist of 58,000 workers.

Conclusions

As is shown above it is possible to use a centralised data base of occupational radiation exposures for epidemiological studies as well as for regulatory control and other purposes. While the requirements may be different for different purposes these can be accommodated by the careful design of the dose registries.(9)

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GERMAN CONTRIBUTION TO THE INTERNATIONAL EPIDEMIOLOGICAL STUDY TO THE RISK OF WORKERS IN THE NUCLEAR INDUSTRY

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INTRODUCTION

The International Agency for Research on Cancer has initiated and coordinated epidemiological cohort studies among employees in nuclear industry in several countries with the aim to carry out a common analysis. The magnitude of the number of persons who are included in this study and the derived number of person-years is assessed in such a range that it should be possible to get information to the risk of carcinogenic effects after a chronic low-dose radiation exposure. The results of that analysis will be compared with the risk factors derived from persons who are exposed by high-dose radiation: e.g. the A-bomb survivors of Hiroshima and Nagasaki.

MATERIAL AND METHODS

A group of experts has worked out guidelines for the collection and analysis of the data (1). The study will only include persons who work in the nuclear industry. The German study population is restricted to workers of the nuclear power plants and larger contractors who are working as occupational radiation exposed persons of the category A as defined in the German radiation protection regulatons. The basis for the German data is a data collection of the Industrial Injuries Insurance Institute for the Electrical Industry including Precision Mechanics.

For reason of prevention of industrial accidents and occupational diseases the Industrial Injuries Insurance Institutes (IIII) are committed to ensuring a safe and healthy working environment. Beside laws and statutory ordinances of the government the IIIIs have issued Accident Prevention Regulations (Unfallverhütungsvorschriften). One of these regulations is the „Preventive Occupational Medical Care" (VBG 100) (2). That VBG 100 describes the required occupational medical examinations: a first examination before employment starts, examinations during employment in regular intervals and regular examinations after the end of the employment. The latter examinations are also required for occupational radiation exposed persons of the category A.

For the organization of these medical examinations since 1989 the companies have to report all employees to the IIII. Personal and occupational data are reported. Among the latter annual dose is additionally necessary for appropriate risk assessment. This can be explained by evaluating the probability of causation (PC). The PC of a radiation-caused late effect like cancer is different for different dose distribution. That effect is shown in the following example.

Suppose, leukaemia is diagnosed for a person of age 60. The following table shows the calculated PC (3) for exposure at different ages.

Dose [mSv]	Age at exposure [years]	Latency Period [years]	Probability of Causation [%]
50	40	20	2.7
50	45	15	4.0
50	50	10	6.3
50	55	5	9.2
5 x 10/year	41 - 45	19 - 15	4.7

Table 1. Calculated PC for an exposure to different ages

It can be seen that the age at exposure and the latency period have an important influence on the risk estimates. Both quantities are derived from the annual dose distribution.

PRELIMINARY RESULTS

Until today about 6,800 persons who are or had been working in the nuclear industry have been reported from the companies. 774 of them have already stopped working as occupational radiation exposed person. The lifetime-dose distribution of these 774 persons is shown in Table 2.

Dose Range [mSv]	0	>0 - <=50	>50 - <=100	>100 - <=150	>150 - <=200	>200 - <=250	>250 - <=300	>300 - <=350	>350 - <=400	>400 - 575
Number of Workers	138	489	56	26	22	8	10	5	7	13

Table 2. Distribution of the lifetime-dose of the 774 persons who have stopped working as occupational radiation exposed person of category A.

It will be possible to consider the effect of risk dependency of dose distribution over time because the annual dose distribution belongs to the reported data. Figure 1 shows examples of the dose distributions for 4 persons.

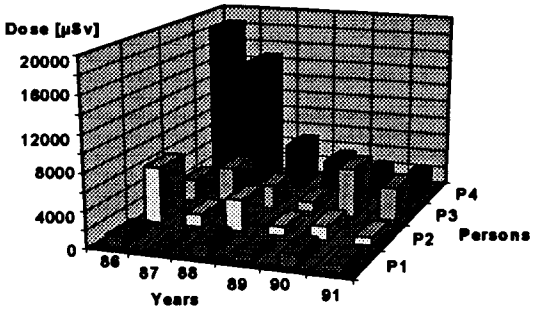

Figure 1. Example for different annual dose distributions among 4 workers

It can be seen that similar lifetime-dose can be a result of very different dose distributions.

An important factor for a precise risk calculation is not only the number of persons included in the study but also the number of person-years. That number depends on the lenght of the follow-up period for the workers in each company. The earliest possible starting point of the observation period is the date of commissioning of the installations. It will be important to obtain this data as complete as possible therefore it might be necessary to fix starting points for the companies different from the date of commissioning.

In a first step the statistical power of the German data alone was calculated based on the assumptions that 96,000 person-years were accumulated until the year 1995 and that the average age of the workers at time of entrance into the study was 30 years. Then the German part of the cohort study has enough power to exclude a 3fold or higher relativ risk (with 95% probability) for lung cancer (power about 100%) and leukaemia (power about 97%) by following up the cohort until the year 2000 (4).

A first joint analysis of the international study is planned in 1996/97.

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RECONSTRUCTING EXPOSURES TO LOW DOSES OF IONISING RADIATIONS OF THE FRENCH ATOMIC ENERGY COMMISSION (CEA) WORKERS

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INTRODUCTION

An international study, initiated by the International Agency for Research on Cancer, is being carried out concerning the effects of protracted low doses of ionising radiation. The following organisms: Commissariat à l'Energie Atomique (CEA), COGEMA and their subsidiaries are involved in this study, which concerns a retrospective cohort of workers. The main objectives are to assess health effects through a mortality analysis and to compare results with currently admitted dose-effect relationship.
The doses registered for the monitored workers of the cohort refer to the notion of « whole-body dose equivalent », which results from the necessity to comply with radiation protection standards and not from epidemiological purposes. It is thus important to reconstruct the historical practices of dosimetry and radiation protection for two purposes: i) to be able to define rules for assessing cumulated doses in a consistent way over time and facilities included, and ii) to assess the biases resulting in the differences noted down.
This paper discusses the role of two main factors: the technical evolution of dosimetry and variation in dose thresholds used in registration.

TECHNICAL EVOLUTION OF DOSIMETERS

Apart the fact that the introduction of a new type of dosimeter appears from table 1 as for a given year, the dates indicated should not be considered as an absolute reference, since the introduction of a new type of dosimeter is usually progressive. Thus it is often difficult to assert with certainty the metrological origin of a given dose even knowing year and place of registration.
Dosimeters used for compliance with radiation protection standards have always been films, either dental films in the earlier years or dosimetric films, even when thermoluminescent dosimeters became available (in the early 1980s'), the argument being to privilege traceable information.
The quick saturation of dental films implied a reading every week or every fortnight. The other disadvantage was the overestimation of low energy X rays: for instance, for a given dose of 50 keV the blackening resulting from X rays could be 30 times the one resulting from radium (with which the dosimeter calibration was made). To avoid such an overestimation, systematic investigation was made every time the dose was beyond a threshold value and correction was made if necessary. This practice allowed to take into account the highest overestimation component. However, it did not allow to correct for overestimation below this threshold which become non-negligible when cumulated. The only way to correct this bias retrospectively is to use information on the workers' activity during that period. For such a cohort (40,000 workers expected), it is impossible to detail for each individual the exposures of a working life. Another method is to assess exposure indirectly through a matrix associating the probable exposure for each job or, failing that, at the laboratory or the workplace level. It is intended to make such a matrix for CEA workers but the quality of the resulting information will largely depend on the remembrance of older or retired workers for the period considered here: the early 1950s'. This matrix would also take into account other carcinogenic exposures if possible and particularly internal contamination.
In the early 1960s', new film dosimeters appeared. Two types of dosimeters were distributed depending on the workers' activity: the first one was for people working with X generators which produced at that time low energy X rays and the second was for reactor workers exposed to high energy γ rays and thermal neutrons. Several types of films have been in use in CEA facilities (Kodak type 1, 2, 3 films and Du Pont films). These films yielded comparable results.
The dosimeter still in use now integrates filters for all relevant types of external radiation. It appeared around 1967 in Fontenay-aux-Roses and 1965 in Marcoule (with negligible difference).

Table 1: Dosimeters used for routine personal monitoring of external radiation in CEA
(films processed in Fontenay aux-Roses: all facilities, except Marcoule, Pierrelatte and La Hague)

Years	Dosimeter	Filters	Comment
1950-1956	Dental film	Open	Quick saturation and fading
1957-1959	Dental film	Open Pb, Sn	Softening of darkening due to low energy X rays
1960-1966	Kodak type 1 film	Dependent on worker's activity:	
		Low energy X rays exposure: Open 0.2 mm Cu 0.6 mm Cu	X generators → low energy X rays (medical and biological applications, cristallography)
		High γ exposure Open Sn (γ) Cd (thermical neutrons)	Reactors → high energy rays (γ and thermal neutrons)
1967-	Kodak type 1 film	One-body Dosimeter Open 300 mg/cm² plastic 1.5 mm Al 0.2 mm Cu 0.6 mm Cu 0.4 mm Pb + 1 mm Sn	 β, low energy X and γ rays 30 to 100 keV X and γ rays high energy photons and thermal neutrons

DETECTION AND RECORDING THRESHOLDS

The *detection threshold* is the first dose value detectable by a given dosimeter. It depends on the technology used for monitoring and usually decreased over years (see table 2). This value may be different from the *recording threshold* which is defined as the minimal positive value actually recorded.
In CEA group, the recording threshold has often been taken higher than or equal to the detection threshold considering that the recorded dose should take uncertainties into account to err on the side of safety. In North American and British facilities included in a joint analysis performed by the International Agency for Research on Cancer (IARC), the recording threshold was smaller than or equal to the detection threshold in most cases (Cardis, 1995). The idea was to associate to doses below detection threshold an average of actual realistic values, thus smaller than the detection threshold.

Table 2: Monitoring and recording practices for γ exposures in the civilian CEA (except Marcoule and La Hague)

Years	Number of dosimeters issued annually per worker	Detection threshold (mSv)	Recording threshold (mSv)	Convention for doses below level of detection (mSv)	Illustrative example (Annual value in mSv) [2]
1951-1952	52	(1)	0.10	<0.10	0
1953-1954	26	(1)	0.50	<0.50	0
1955-1956	26	(1)	0.50	0.50	13
1957	26	(1)	0.25	0.25	6.5
1958	26	(1)	0.25	/	0
1959	26	0.50	0.50	/	0
1960-1962	12	0.20	0.35	/	0
1963-1994	12	0.20	0.35	0	0
1994 (end of)	12	0.10	0.35	0	4.2
1995	12	0.10	0.20	0	2.4

[1] Unavailable

[2] Worker exposed each fortnight to 0.08 mSv

Values adopted for doses below or at the detection threshold may consistently affect cumulated doses especially when the major part of the population is exposed around the detection threshold, as in most of the cohorts of nuclear workers (Kendall et al., 1984). An arbitrary but illustrative example is described in table 2. A worker exposed to 0.08 mSv every fortnight during his working life would be annually recorded as having cumulated 13 mSv in 1955 and 1956 (26X0.50), 6.5 mSv in 1957, 0 between 1958 and 1993 and 2.4 mSv in 1995, for an actual annual value of 2 mSv. As noted by Kendall et al. (1992), several corrections should be made to avoid the overestimation of the earlier years: threshold doses can be set to zero or the minimum recordable dose (i.e. threshold multiplied by the number of issues in a year) can be subtracted from all annual dose sums. Remeasurements of old films may also be relevant if the recording threshold proved to have been taken much higher than the detection threshold (Smith and Inskip, 1985).
Differences between facilities have to be taken into account as well. The same worker's annual dose would have been recorded 2.4 mSv (12X0.20) in La Hague between 1968 and 1995 (recording threshold being 0.15 mSv and recording increment being 0.05 mSv). With such a protracted exposure for 10 years, this yealds from 0 to 24 mSv for the total cumulated dose.
Those variations due to recording practices should have to be taken into account when defining the classes of doses in the cohort analysis and specially the lowest exposed group. For an illustration [0-10 mSv] is the most common first dose class considered in recent nuclear worker cohort studies (Cardis et al. 1995, Kendall et al., 1992).

CONCLUSION

Technology in dosimetry for external individual exposure evolved from dental films without any filters to one-body dosimeter integrating up to 6 different filters. Main bias for the first period (till approximately 1957) is resulting from the overestimation due to low energy X rays. A method to eliminate or reduce this bias is to use a job exposure matrix. « CEA-EXPO » retrospective job-exposure matrix currently under way should help to do so. Results will largely depend on the remembrance of old workers and on technical information found in the archives.
It has been shown that variations of detection and recording thresholds over time and facilities may have non-negligible consequences on cumulated doses. When all dosimetric data are computerised, different rules will be applied for estimating the cumulated doses. Sensitivity studies will be made when analysing the final cohort to see the influences on conventions adopted and chose the best approximations.

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Acknowlegments:

Authors are grateful for the help of the members of the working group « exposures and validation of dosimetry »: Mrs Dumeau, Mr Dagorn, Mr Espagnan, Mr Le Bar, Dr Le Goff, Mr Magri, Mr de Padua, Mr Pescayre, Dr Tourte and particularly to Mr F. Bermann, member of the dosimetry sub-comittee for the IARC study.

THE THYROID: MEDICAL SURVEILLANCE OF EXPOSED WORKERS

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ABSTRACT

In this study the authors: 1) outline a rational approach to diagnosis and management of thyroid diseases; 2) evaluating the prevalence of thyroid diseases on a group of 207 radiation workers observed over a six years period, identify abnormalities which require a careful evaluation and a more restrictive judgement of fitness.

INTRODUCTION

In the system of dose limitation, closely correlated to the hypothesis of a linear relation between dose and occurrence probability of stochastic damage, medical surveillance undoubtely acquires obligations with regard to tumour prevention (secondary prevention). ICRP 90/77 places the thyroid in the group of organs of most oncological competence. The American Cancer Society, in its recommendation on the early diagnosis of cancer in asymptomatic people, extends the diagnostic indication to numerous sites (thyroid included), which also coincide fairly closely with the sites having a high risk coefficient. Based on the above considerations, thyroidology should not be neglected. The only factor that has been shown unequivocally to cause thyroid cancer is radiation exposure.

THYROID CANCER AND THYROID BENIGN ABNORMALITIES AT INCREASED RISK OF CANCER

The thyroid is an uncommon site of cancer accounting for 0,6% of cancers among men and women respectively. In 95% of cases, thyroid cancer presents as a nodule in the thyroid. In contrast to rare thyroid cancer, thyroid nodules are extremely common particularly among women. The prevalence of thyroid nodules is about 5% of the adult population with a female: male ratio of 4:1. Thus most thyroid nodules are benign and the problem is to identify those that are likely to be malignant. Many studies have considered the possibility that thyroid cancer is preceded by other thyroid abnormalities, including endemic goiter, sporadic goiter, benign thyroid nodules, lymphocytic thyroiditis, and Graves' diseases. The findings remain inconclusive; nevertheless, this question is of great interest from the point of view of radiation protection, since these thyroid diseases are common and it is important to know whether such patients should be considered at increased risk of developing thyroid cancer.

CASE CONTRIBUTION

In the period 1889-1995 a total of 207 exposed workers 87 females and 120 males, ranging in age from 19 to 65 years, were tested with clinical examination, FT3, FT4, TT3, TT4, TSH and ultrasound. Selected cases, in the presence of doubtful thyroiditis, were tested with thyroglobulin autoantibodies and thyroid microsomal autoantibodies. Workers known, by palpation or ultrasound, as carriers of alterations of the glandular morphology were tested with scintigraphy, 131 I thyroid uptake and eventualy T3 suppression test in case of doubtful differential diagnosis between hot and cold nodules. Nodules less than 5 mm were not considered. Only in the presence of cold, solid and palpable nodules a fine-needle aspiration biopsy was performed to diagnose malignant nodules, benign nodules or follicular neoplasms. A total of 69 cases of thyroid diseases were found, with a prevalence of about 33% (females 36%, males 29,5%), (Table 1).

Table 1: Prevalence of thyroid diseases in 207 exposed workers. 1989-1995.

NONTOXIC GOITER:	DIFFUSE: 5(2.4%)	
	NODULAR: 23(11,1%)	MONO: 6(2.4%)
		MULTI: 17(8.2%)
DIFFUSE TOXIC GOITER: 2(0.97%)		
HASHIMOTO'S THYROIDITIS: 3(1.5%)		
		MALIGNANT 2(0,97%)
	PALPABLE:	FOLLICULAR ADENOMA 1(0.5%)
		COLLOID NODULES 7 (3,4%)
SOLITARY NODULES: 30(14.5%)		
	UNPALPABLE 20(9.7%)	
MULTIPLE NODULES: 6(2.9%) UNPALPABLE		

DIAGNOSTIC DISCUSSION

With the exemption of hot nodules that as a rule are not malignant and have not been suspected of malignancy, we found 5 groups of thyroid diseases that, representing special problems, require some considerations.

1 - **MULTINODULAR GOITER:** Two patients evaluated for single nodules were found to have additional small thyroid nodules. It is commonly stated that demostration of multiple nodule on ultrasound in a patient with a single palpable nodule makes malignancy unlikely. No carefully controlled study supports this contention, however, and the malignant potential of any nodule should be based on other criteria. In most cases, this should include a FNA biopsy.

2 - **CISTIC NODULES:** Also cystic nodules that as a rule are not cancer, have not been considered suspicious of malignancy, but since cancer is occasionally found in the wall of the cyst, it has been thought necessary the echographic surveillance of cystic lesions every periodical visit.

3 - **HASHIMOTO'S THYROIDITIS (HT):** A high prevalence of HT was confirmed. Since a patient with HT may develop, even if rarely, lymphoma of the thyroid gland, a protocol of caution for the follow up is recommended. Ultrasound should be carry out semiannually.

4 - **FOLLICULAR ADENOMA:** Considering the limits of FNA biopsy in the presurgical differential diagnosis between follicular adenoma and follicular carcinoma, we have referred to the surgeon the case of follicular adenoma. In this 1 case there was an increased chance of malignacy (large lesion over 2 cm firm nodule, young patient, values of nuclear parameters - area and perimeter - 30% over the normal.

5 - **HISTORY OF RADIATION THERAPY:** In 1 case of benign solitary nodule there was a history of radiation therapy to the neck area. Althoug any given nodule has essentially the same chances of being malignant regardless of whether there has been previous upper body radiation, caution is in order.

CONCLUSION

The case report, in particular the two papillary carcinomas, show how the diagnostic protocol suggested above could become useful also from the point of view of the secondary cancer prevention.

The diagnostic protocol by us conciliates with due economy but with the right balance, the real requirements of medical surveillance, reducing as far as possible undesiderable effects such as damage from excessive protection and patient/physician delay, which is extremely dangerous in the early diagnosis of tumours.

MEDICAL JUDGEMENT OF FITNESS FOR RADIATION WORK

As far as the medical judgement of fitness for radiation work is concerned, one has to keep in mind some main criteria, i.e., the clinical findings; the specific risk evaluation; the social, economical and psychological parameters. Thus, in his evaluation, the physician should always consider the risk/benefits balance.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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ABSTRACT (See instructions overleaf)

Two successives incidents with I-131 occurred in the Radioisotope Production Facility at IPEN-CNEN/SP, contaminating areas and workers. However, no evidence cause was found, being the case still under investigation. In this paper is analysed the probable causes of how it could has been happened and the consequences, which were scanned by radiological protection measurements. Individuals internally contaminated were monitored by "in vitro" and "in vivo" bioassay. The results were used to estimate the thyroid's exposure risk reflecting the collective dose

THE IMPORTANCE OF RADIATION AND NON-RADIATION-FACTORS FOR THE STOMACH CANCER INCIDENCE IN WORKERS OF THE ATOMIC PLANT MAYAK

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INTRODUCTION

Stomach cancer is one of the most widespreaded oncological diseases among the men. It is known about the increase of stomach cancer incidences and mortality among the A-bomb survivers (1) as well as among the professionals subjected to the chronic exposure to external gamma-irradiation and internal exposure due to body burden 239 Pu of high doses (2). Inasmuch as this disease is polyetiological, the determination of non-radiation factors contribution is important for more accurate estimation of the effect of ionizing radiation on its advent and for its more successful prophylaxis. With this in mind a retrospective investigation of the personal of the atomic plant MAYAK was conducted, who start their work in 1948 to 1960.

MATERIALS and METHODS

503 workers of the atomic plant MAYAK were examined that were subjected to external chronic gamma-irradiation with the total dose of 0 to 8,46 Gy and body burden ^{239}Pu 0 to 39,6 kBq. The data on individual external gamma-irradiation doses were presented by enterprise monitoring survice and data on body burden ^{239}Pu by biophysic laboratory of Branch No 1 of Biophysic Institute (the chief of the laboratory is Dr. V.F.Kchochryakov). From 1955 to 1992 stomach cancer was diagnosed by 157 persons (141 men and 16 women) morphologically confirmed (autopsy and biopsy during surgical operation). 346 persons (313 men and 33 women) without cancer served as pair control, matched on sex, age (±5 years), beginning of the work at the plant (±2 years) and profession. Besides radiation factors the effects of medical, biological and social factors were studed influencing the development of this disease (table 1). The data about the smoking and alcohol taking were received from individual interviews. The contribution of studed factors to stomach cancer incidences was estimated by odds ratio (Or_{cr}-odds ratio crude and Or_{ad} - odds ratio adjasted), calculated on base of the model of multiple logistic regression. Attributable risk (AR) was calculated for statistically significant factors, that allows to determine what part of the investigated cancer incidences is subjected to the each factor. Statistic programms SAS and BMDP were used.

RESULTS AND DISCASSION

The mean values of total and maximum doses of chronic external gamma-irradiation in the main and in the control group were significantly not different, but the part of persons with the total external gamma-dose more than 3 Gy was significantly higher in the main group (table 1). No important differences were noted between the main and the control group in ^{239}Pu content in the organism and in the frequency of previous professional contacts with chemically dangerous substances.

A significant increase of current smoker part and smoking index value was noted in the main group but in the control group the part of exsmokers was increased. The age of start smoking was the same in both groups, there were no differences in alcohol taking.

The differences in constitution between the main and control groups were characterized by mass index decrease of persons in the main group some years before tumor diagnose. The frequency of chronic hypocidity and anacidity gastritis, colitis, gallbladder diseases was significantly higher in the main group. The percent of stomach and duodenal ulcers was the same in the both groups, but there were differences in ulcer localization. The mean age of the persons fallen ill with cancer was 51.2±0.8 years and the period from the begin of the contact with ionizing radiation up to tumor diagnose was 22.1±1 years.

Table 1

COMPARISION OF STOMACH CANCER CASES AND NON-MALIGNANT CONTROL

	Risk Factors		Mean or percentage ± error		"P" value
			case	control	
			n = 157	n = 346	
	Matched variables				
1	Sex (% men)	%	89,8 ± 2,4	90,5 ± 1,6	> 0,05
2	Years of birth	yr	1924,0 ± 0,8	1925,5 ± 0,5	> 0,05
3	Age of work's start	yr	29,2 ± 0,6	28,1 ± 0,4	> 0,05
4	Part of worker's profession	%	79,6 ± 3,2	76,3 ± 2,3	> 0,05
	Professional factors				
5	Total external γ-irradiation	Gy	126,9 ± 13,6	104,3 ± 5,9	> 0,05
6	Total external γ-irradiation > 3 Gy	%	12,7 ± 2,7	5,8 ± 1,3	< 0,01
7	Maxim.. years rate γ-irradiation	Gy	48,4 ± 6,7	35,7 ± 2,6	< 0,1
8	^{239}Pu body burden	kBq	1,32 ± 0,32	0,76 ± 0,11	< 0,1
9	Previous professional contacts with chemically dangerous substances	%	14,0 ± 2.8	8,4 ± 1,5	< 0,1
	Social factors				
10	Smoking: current smoker	%	62,4 ± 3,9	44,8 ± 2,7	< 0,001
	exsmoker	%	13,4 ± 2,7	27,2 ± 2,4	< 0,001
	never smoker	%	24,2 ± 3,4	28,0 ± 2,4	> 0,05
11	Age of smoking's start	yr	17,5 ± 0,6	17,6 ± 0,8	> 0,05
12	Smoking's index	cig/d	700,0 ± 37,8	559,0 ± 22,1	< 0,01
13	Alcohol taking: regularly	%	22,3 ± 3,3	17,6 ± 2,0	> 0,05
	occasionally	%	57,3 ± 3,9	57,5 ± 2,7	> 0,05
	rarely	%	20,4 ± 3,2	24,9 ± 2,3	> 0,05
	Medical and biological factors				
14	Body height	cm	169 ± 0,6	169 ± 0,4	> 0,05
15	Body weight	kg	70,4 ± 0,8	72,5 ± 0,5	< 0,05
16	Quetlete's index	kg/m^2	24,6 ± 0,3	25,5 ± 0,2	< 0,05
17	Chronicle hypoacidity gastritis	%	18,5 ± 3,1	8,4 ± 1,5	< 0,01
18	Chronicle anacidity gastritis	%	26,8 ± 3,5	6,4 ± 1.3	< 0,001
19	Chronicle hyperacidity gastritis	%	1,9 ± 1,1	3,8 ± 1,0	> 0,05
20	Stomach and duodenal ulcer:	%	11,5 ± 2,5	11,0 ± 1.7	> 0,05
	stomach ulcer	%	7,6 ± 2,1	0,3 ± 0,3	< 0,001
	duodenal ulcer	%	3,8 ± 1,5	10,7 ± 1,7	< 0,01
21	Stomach polyps	%	2,5 ± 1,3	0,3 ± 0,3	<0,1
22	Chronicle colitis	%	12,7 ± 2,7	6,6 ± 1,3	< 0,05
23	Gallbladder diseases	%	24,8 ± 3,4	14,5 ± 1,9	< 0,01

The influence of 13 potential risk factors on stomach cancer incidence were studied by multiple analysis based on logistic regression model. 5 significant factors were resolved, OR_{ad} was calculated for each of them (table 2): for the external gamma-irradiation with the total dose more than 3 Gy OR_{ad} = 2,1; for anacidity gastritis OR_{ad} = 8,0; for hypoacidity gastritis OR_{ad} = 5,2; for current smokers relative exsmokers and never smokers OR_{ad} = 2,7 (table 2). By the calculation of attributable risk it was shown that among all studied cancer incidence only 5% were cased by external gamma-irradiation with the dose more than 3 Gy; the main

part of cancer incidences - 29 % is connected with chronicle stomach disease and 24% of them are caused by current smoking.

Besides the study of independent influence of different factors on cancer incidences the evaluation of their interaction is made. In this case it was noted that the interaction between external gamma-irradiation with the total dose more than 3 Gy and gastritis is

Table 2.

DEPENDENCE of STOMACH CANCER of VALUE DIFFERENT RISK FACTORS

Factors	Range	Frequency (n)		OR_{cr}	OR_{ad}		AR and
		case	control		Point est	95% CI	95% CI
Total external γ-irradiation (Gy)	0-3 > 3	137 20	326 20	2,4*	2,1	1,01-4,4	5,2% (0,7%-9,4%)
Hypoacidity gastritis	no yes	128 29	317 29	2,5**	5,2	2,45-11,0	7,8% (2,3%-13,0%)
Anacidity gastritis	no yes	115 42	324 22	5,4**	8,0	4,4-14,6	16,1% (9,7%-22,0%)
Stomach ulcer	no yes	145 12	345 1	28,6*	41,2	5,2-328	5,1% (2,1%-8,2%)
Stomach polyps	no yes	153 4	345 1	9,0			
Noncurrent smoker Current smoker		59 98	191 155	2,0**	2,7	1,7-4,3	24,4% (10,3%-36,3%)
Alcohol taking	irregularly regularly	122 35	285 61	1,3			

* - $P < 0.05$, ** - $P < 0,01$

submultiplicative and the interaction between gamma-irradiation and smoking, alcohol taking is multiplicative.

In such a way as a result of the conducted study it was determined, that the chronic external gamma-irradiation causes the statistically significant increase of stomach cancer incidences by the total dose more than 3 Gy. It is known, that the significant increase of cancer risk by the dose more than 200 rad was noted by the A-bomb survivors in the group with the age of irradiation moment that corresponded to the age of MAYAK workers at the moment of the first contacts with ionizing radiation /1/.

Among the examined persons more higher probability of cancer incidence was noted for current smokers relative to never smokers and exsmokers and the value of smoking index hadn't any great importance, that is the main importance had just the fact of current smoking. The data from references on this problem are different but the many authors doesn't find correlation between the stomach cancer and smoking. In our case cheap sorts of papirosen with high content of toxic substances and without filter were taken chiefly and just that fact could promote cancer. It was not found direct correlation between stomach cancer and alcohol taking. This is in accordance with literature data but it must be noted, that the examined persons could give incorrect information from social reasons about the quantity of alcohol taking, specially lowing it.

The most high probability of stomach cancer was noted among the workers with chronicle stomach diseases. In this connection we are studying now the problem on the possible influence of ionizing radiation on the incidence of them and , if such an influence will be confirmed, the stimation of the contribution of ionizing radiation to the stomach cancer incidences among the personnel could be changed.

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IRPA9
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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

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More than 6,000 residents and primary school or kindergarten students in Taiwan have been identified to have received continuously excessive low dose γ-ionizing radiation (cumulative 0.5 to 100 rems) from various environmental sources, including radioactive apartments and radioactive classrooms containing Co-60 contaminated steel rebars or window frames, since 1983 to 1995 (Lancet, 342:1544). Several correlated studies have been undertaken in order to understand the health effects of such exposure in human, including exposure dose assessment by the time-activity-based environmental monitoring (a joint program with the National Institute of Occupational Safety and Health, U.S.A.), a systemic and comprehensive medical surveillance program, the growth and development evaluation in children borne and grown up in the apartments, and the cytogenetic monitoring in their peripheral lymphocytes. The risk of various health effects will be demonstrated and compared with those of current risk coefficients employed by the International Commission on Radiological Protection(ICRP). This report will be an update of

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THE ESTIMATE REPRODUCTIVE HEALTH STATUS OF POPULATIONS EXPOSURED IN LOW DOSES IN RESULT OF CHERNOBYL DISASTER.

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One of the indices characterising the health of the population is the reproductive period including such estimations as birth rate, the health status of pregnant women, the frequency of untoward outcomes of pregnancy, the health status of newborns. In accordance with these evidence one may judge about the health status not only of the present but also of the future generations.

One of the major manifestations of impairment of the reproductive health is stochastic effects in the offspring of the exposed parents. The stochastic effects having their origin in gonads can be reproduced and revealed in the form of inherited impairments in the offspring of an exposed person. Such effects may become apparent in many generations. The dominant mutations dispeay mainly in the first and second generations after exposure while the recessive mutations add genetic following generations. In addition to the effect in parents' gonads, the radiation may the affect the embryo and fetus. Within the period from three weeks after the conception to the outcome of the pregnancy the stochastic effects seem to be able to appear.

The epidemiological investigation (1980 - 1992) reproductive health status of the population exposed due to the Chernobyl accident was performed in two districts of Bielorussia (Gomel and Mogilev) and in two districts of Russia (Briyansk and Riyazan) having different levels of territory contamination with ^{137} Cs and ^{90} Sr as well as different levels of dose loading of summary (external and internal) radiation.

In order to estimate the reproductive health, the data of the questionnaires of women living in the mentioned districts, the long-term data of regional medical statistics as well as the epidemiological data of medical help to pregnant women and newborns according to the chosen criteria were used.

In Gomel and Mogilev districts of Belorussia and in Briyansk and Riyazan districts of Russia the doses of external and internal radiation determined by the counting method were from 0,4 sZv to 15,2 sZv by 1992, and the doses for thyroid gland - from 15 sZv to 450 sZv.

The study of the long-term dynamics of frequencies of alive births and medical abortions in Gomel, Mogilev and Briyansk districts revealed the decrease of the frequency of alive births during post-accident period followed by the growth of the frequency of medical abortions. This process was typical both for the exposed population and for the control one.

The analysis of the long-term dynamics of the frequency of the spontaneous abortions in the radioactive contaminated districts beginning from 1987 to 1992 showed their growth from 4,1% in 1987 to 8,1% in 1992.

The average annual summary frequencies of untoward outcomes of pregnancy during the pre-accident period (from 1980 to 1985) were 9,6 ± 0,9% as to the post-accident period (from 1986 to 1992) they were 13,4 ± 1,0%.

In the control population the frequencies of untoward outcomes of pregnancy were 4,8 ± 1,2% and 4,3 ± 0,4%, respectively.

Thus, the above mentioned index is positively higher for the population of the contaminated districts during the post-accident period compared to the same index for women of the control population.

So, in the contaminated districts, on the one side, the decrease of birth rate and the increase of the frequencies of medical abortions are observed that may be considered the consequence of the stress situation and regarded as conscious regulation of the family structure under the conditions of radiophobia; on the other side, the growth of the frequencies of untoward outcomes of pregnancies and increase of morbidity among newborns that may be regarded as the increase of the pressure of the mutagen factor are revealed.

The frequencies of untoward outcomes of pregnancies (UPO) depending on the doses received by the population are determined. Their average annual meanings are as follows:

9,4 ± 0,6% (1 - 3 sZv); 8,2 ± 0,5% (3 - 5 sZv);

8,9 ± 0,9% (5 - 10 sZv); 13,2 ± 1,2% (above 10 sZv).

The frequency of toward outcomes of pregnancy is positively higher only in the total dose above sZv and on the territories with the density of the ^{137} Cs contaminated soil with 5 - 15 Ci/km^2 or more.

The analysis of the epidemiological data in Riyazan and Briyansk districts (the level of ^{137} Cs contaminated from 1 Ci/km^2 to 5 Ci/km^2 and from 5 Ci/km^2 to 15 Ci/km^2 , respectively) revealed the tendency of the decrease of the reproductive health of the population common tot the contaminated and controlled territories outlined since 1980. The impairment of the reproductive health is in the decrease of the birth rate in 1,13 - 1,2 times, in the growth of incidence of spontaneous abortions from 5% to 10%, in the increase of the morbidity of the pregnant women, in the growth of the frequency of the prematurely born children, in the increase of the frequency of the sick newborns.

The most distinct tendency of decreasing the reproductive health both in the contaminated and in the control districts was observed since 1989 that is likely to be connected with worsening the general economic situation in the country. In the contaminated areas of the Riyazan district the most district negative dynamic of the indices analysed than during the pre-accident period was observed. The intensive indices (the coefficient of negative dynamic) of the general morbidity of the pregnant women increased from 1,2 to 1,41 (18%), of the general frequency of the UPO from 1,51 to 1,93 (28%), of the general frequency of morbidity of newborns from 0,98 to 1,2 (22%), of the prematurely born children from 0,85 to 1,17 (38%).

In the contaminated areas of the Briyansk districts with density of ^{137} Cs contaminated soil the change of the positive dynamic in to the negative one during the post-accident period in comparison with the pre-accident period was observed in two indices: the frequency of congenital anomalies (the coefficient changed from 0,66 to 1,41, that is 214%) and the frequency of stillbirths (the coefficient changed from 0,66 to 1,39, that is 210%).

In general the results achieved show that in the population living on the territories contaminated with radionuclids the most distinct negative dynamic of reproductive health than in the control areas is observed dourly the post-accidental period. The highest intensively of the negative dynamic and for certain the lower (than in the control) absolute indices of the reproductive health are revealed in the population living on the territory with ^{137} Cs contamination from 5 Ci/km^2 to 15 Ci/km^2 . Under the equal conditions of ^{137} Cs contamination of soil the most distinct decrease of indices of the reproductive health was revealed in the areas with the most original level of the reproductive health that witness about the important role of the original level of the reproductive health in forming the effects of the exposure of the population. In the structure of the chosen indices of the reproductive health the loading place according to the intensively of the negative dynamic takes the indices characterising UPO and the status of the newborns, that is the indices reflecting the somatic stochastic effects of radiation influence.

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MAJOR SCIENTIFIC TOPIC NUMBER 2.8-2.3 (See page 7)

ABSTRACT (See instructions overleaf)

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Immediately after the Chernobyl accident the Registry of exposed persons had been established in the former USSR. At present time the National Chernobyl Registry of Russia set up at the Medical Radiological Research Centre of RAMS, Obninsk, is working. Medical and dosimetric information on 435 thousand of individuals exposed to radiation including 160 thousand of liquidators is kept at the Registry.

Analysis of demographic and dosimetric information of the Registry related to liquidators allowed us to estimate the attributive risk of mortality caused by malignant tumors as 2.8%. This estimation has been confirmed by available actual data of the Registry for the period from 1986 to 1995. At the same time we have established that there is the statistically significant relationship between the dose of exposure and incidence rate of malignant tumors for liquidators.

By results of epidemiological analysis of thyroid cancer cases performed in Bryansk Oblast by the "case-control" technology it was possible to estimate the relative risk of this type of diseases as 7.15 (1.52; 33.8) at the dose of 1 Gy.

The results obtained allow us to say that theoretically expected estimates and actual data correlate well.

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The results obtained allow us to say that theoretically expected estimates and actual data correlate well.

INCIDENCE OF CONGENITAL ABNORMALITIES AMONG THE NEWBORNS FROM BUCHAREST AFTER THE NUCLEAR ACCIDENT FROM CHERNOBYL

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INTRODUCTION

The nuclear accident at Chernobyl has resulted in an important environmental contamination.

In order to assess the health effects of the accident on the newborn children an incidence study was conducted during the period October 1986 - December 1987.

MATERIALS AND METHODS

Data on natality, perinatal deaths, congenital abnormalities and deaths during the first 6 days after birth were collected. These data were compared to those obtained from the statistical centre of the Ministry of Health regarding a similar period before the accident. In addition, a questionnaire regarding the developpement of pregnancy was administered to the mothers.

RESULTS AND DISCUSSIONS

The data were analysed separately for those exposed during the foetal period and for those exposed during embryonic period. The incidence of congenital abnormalities was 2% (130 cases out of 6484 newborns) for those exposed during the foetal period and born beetween 1 October 1986 - 31 December 1986, and 2% (127 cases out of 6220 newborns) for those exposed during the embryonic period and born beetween 1 January 1987 - 31 Mars 1987 (table 1).

The incidence of congenital abnormalities was 2.1% during 1 January 1987 - 31 December 1987 against 2.6% during 1 January 1985 - 31 December 1985 (the period before the accident).

The proportion of stillbirths registered during the study period were: 1.3% for those born during the interval 1 October 1986 - 31 December 1986 and 0.9% for the interval 1 January 1987 - 31 December 1987; the proportion of stillbirths for the period prior to the accident (1 January 1985 - 31 December 1985) was 0.7%.

The mortality during the first 6 days of life was 1.4% for the period 1 October 1986 - 31 December 1986 and 1.1% for the period 1 January 1987 - 31 December 1987, as compared to 0.5% for the period 1 January 1985 - 31 December 1985.

Also, the incidence of the various types of malformations during the study period is not significantly different from the incidence during the reference period.

No significant differences were shown beetween the incidence of the studies outcomes for the period before the accident as compared to the interval following the accident, except the mortality during the first 6 days of life which was significantly increased (test "U").

The increased number of deaths during the first 6 days of life is probably due to the other factors than exposure to radiation: life conditions, work conditions, difficult pregnancy (table 2).

The exposure for the period 1 May 1986 - 31 December 1987 was estimated to 1.2 mSv to representing the committed effective dose equivalent for an adult from Bucharest.

The value was computed by the group of physicists working in our institute.

We lack the data on the dose delivered to embryo and foetus.

CONCLUSIONS

1. No significant differences were shown beetween the incidence of the studies outcomes for the period before the accident as compared to the interval following the accident.

2. The increased number of deaths during the first six days of life is probably due to the other factors than exposure to radiation.

Table 1. Number of cases found for the study periods (intervals).

	1October1986 - -31December1986	1January1987 - -31Mars1987	1January1987 - -31December1987	1January1985- 31December1985
Newborns alive	8673	6200	26.596	36.255
Newborns alive with malformations	156	127	640	942
Newborns dead (stillbirths)	85	88	227	262
Newborns dead with malformations	25	20	75	-
Number of deaths 0-6 days of age	90	82	296	193
Number of deaths malformed children 0-6 days of age	34	29	109	58

Table 2. The frequency of several teratogenic and inherited factors among mothers who gave birth to malformed children.

Occupational exposure		Diagnostic iradiation	Ill-pregnancy	Family history malfor-mations	Nutri-tional defici-encies	Antibiotic intake during pregnancy	Alcohol intake during pregnancy	Smoker
Toxic	Radioactive							
5.5%	0.4%	0.3%	8.7%	2.3%	15%	5.9%	1.3%	3.3%

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CHILDHOOD LEUKAEMIA IN ROMANIA AND THE CHERNOBYL ACCIDENT

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INTRODUCTION

During the days of April 29 and 30, and May 1, the direction of the contaminated cloud from the Chernobyl Nuclear Power Plant has changed, resulting in a contamination of the atmosphere in our country. Several reports have been presented on the probable long term consequences upon the populations living outside the area of close vicinity to the Plant[1,2,3] . These reports suggest that the possible excess of cancer due to the exposure to ionising radiation would be impossible to distinguish from the background incidence. Following the WHO recommendations, we focused our study on short-term consequences of the Chernobyl accident on childhood leukaemia[2] . The present study was designed to show whether the frequency of leukaemia has increased during the time period following the nuclear accident[4] . We studied the frequency of childhood leukaemia, its geographical distribution in Romania, and the possible changes of this distribution after the Chernobyl accident.

MATERIALS AND METHODS

The study population consists of children 0 to 14 years old, by 5-years age-groups. Since the cancer registry could not provide proper data on the incidence of leukaemia, we decided to include mortality in the study; the death certificates seldom specify the type of leukaemia, therefore we considered deaths from all types of leukaemia. The study included all children from the 41 districts of Romania. The study periods two equal time periods: before the accident (1981-1985), and after the accident (1987-1991). Data on deaths from childhood leukaemia were collected from the Central Commission for statistics (Ministry of Internal Affairs); data on the population 0-14 years of age, for 5 years age-groups, for each of the 41 districts and for each year of the study periods, were drawn from the Centre of Medical Statistics (Ministry of Health). Estimates of radiation doses due to the nuclear accident were taken from the UNSCEAR 1988[5] . Cumulative mortality was computed for each district over the two time periods (1981-1985 and 1987-1991), using as denominators the population for the midyear of each interval.

RESULTS

For the period before the accident (1981-1985), the distribution of the cumulative mortality from leukaemia is shown in Figure 1. The mortality for the entire country was 13.54/100,000 for all age-groups (14.92 for the 0-4 years old age group, 15.68 for the 5-9, and 10.13 for 10-14). After the Chernobyl accident, the geographical distribution of cumulative mortality from childhood leukaemia has somewhat changed (Figure 2). The rate for the entire country was 13.24/100,000 (13.72 for 0-4 years old, 16.64 for 5-9 years old, and 9.83 for 10-14 years old). Four districts showed a greater increase of the mortality rate (Figure 7). The age distribution of the mortality in these districts during the two time periods, is shown in Figures 3, 4, 5, and 6.

DISCUSSION

For the entire study period, the mortality from childhood leukaemia was a little higher in the southern and south-eastern part of Romania. Before the accident, the distribution was quite uniform, and changed somewhat after 1986. The southern and south-eastern parts of the country, where mortality rates are higher, overlap with the area with medium dose of radiation received during the period following the nuclear accident. They are also the areas where agriculture has been most intensive, and where chemical industries (oil processing plants, fertilisers) are most frequent. Four districts have shown an increase of the frequency of deaths from childhood leukaemia, of which only one is located in the area with high radiation doses due to the Chernobyl accident, one is located in the low dose area, and two in the medium dose area.

CONCLUSIONS

Further information are needed, both on the radiation doses due to the nuclear accident, and on the other factors that might induce an increase in the frequency of leukaemia, before valid conclusions can be drawn on the possible influence of the Chernobyl accident on the frequency of childhood leukaemia.

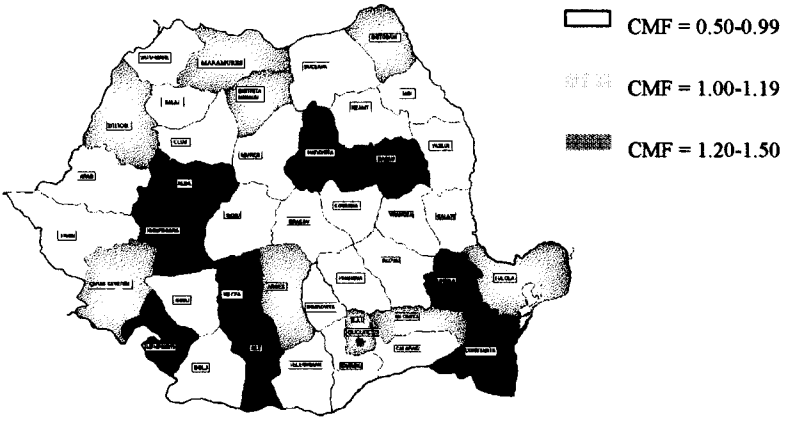

Fig. 1 Cumulative mortality from childhood leukaemia

1981 - 1985

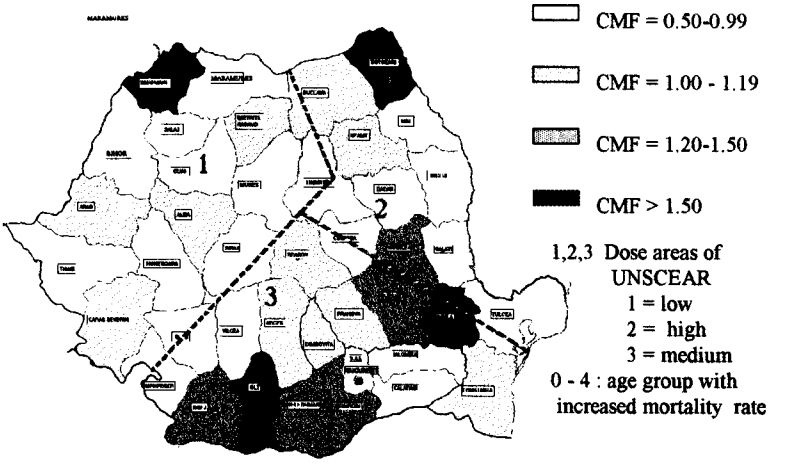

Fig. 2 Cumulative mortality from childhood leukaemia
1987 - 1991

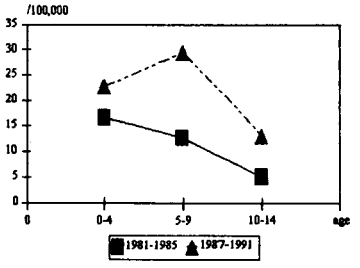

Figure 3. Cumulative mortality from leukaemia-district Satu-Mare

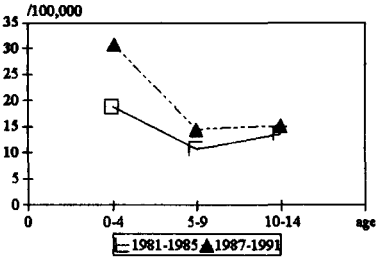

Figure 4. Cumulative mortality from leukaemia-district Botosani

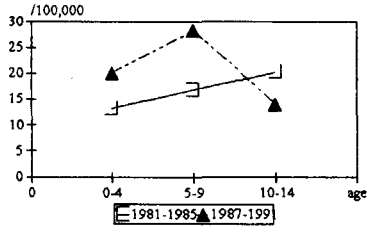

Figure 5. Cumulative mortality from leukaemia-district Braila

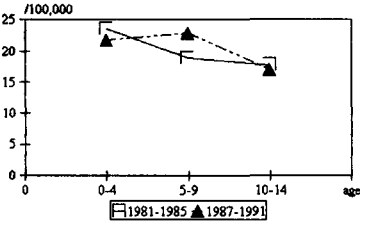

Figure 6. Cumulative mortality from leukaemia-district Olt

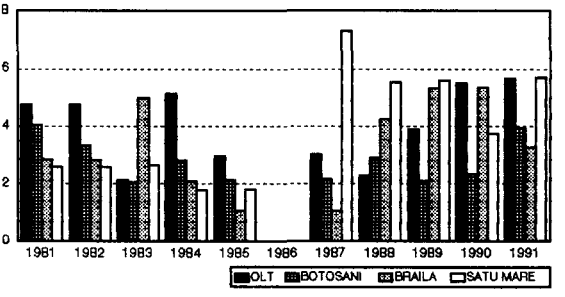

Fig. 7 Mortality from leukaemia, 0-14 years old, in four districts

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IRPA9
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PAPER TITLE CONDITION OF REPRODUCTIVE HEALTH AS CRITERION OF DETERMINISTIC EFFECTS OF LOW DOSES OF RADIATION

AUTHORS NAMES L.A. BULDAKOV, I.J. VASILENKO, S.I. DEMIN, A.M.LIAGINSKAYA, M.M. SAUROV

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MAJOR SCIENTIFIC TOPIC NUMBER

Indices of reproductive health (RH) were studied in persons exposed by long-term radiation during the living at the radiation contamination areas after the accidents in Chernobyl and Southern Urals. 8 RH indices were studied in dynamics. 40,000 people were under surveillance at the territories of Cs-137 radiactive fallout density in the range of 1 to 15 Ci/km^2, where external exposure doses were in the range of 7 to 68 mSv. Control regions population did not have additional irradiation and Cs-137 radiationuclide content in soil was less than 1 Ci/km^2.

The investigated Southern Urals population demonstrated 40-85 thousands of people to have annual radiation doses of 0.71, 7.6, 0.49, and 0.53 mSv for the 1st, 18th, 28th, and 38yh year after the nuclear facility start-up, respectively.

It was found that such RH indices like birth rate, general morbidity of newborns, frequency of hereditary development abnormalities, mortal birth rate were the higher, the larger radiation doses comparing to control and these indices were found more frequently, than that were found for lowest contamination levels caused by Chernobyl accident. Such indices like general morbidity rate in pregnant women, the rate of unfavourable outcomes of pregnancy, early delivery, newborn body mass do not correlate to doses and radioactive fallout levels.

Medical consequences analysis for living in critical zones of Southern Urals demonstrted the confidently positive correlation coefficient (increasing with the elevation of dose) for mortal birth rate and infant mortality. Otherwise, the rate of of hereditary development abnormalities and consequent mortality rate demonstrated the negative correlation to dose.

Revealing of some RH indices conditions can be used as criterion of deterministic effects of low doses of chronical radiation exposure, if other aggrevative influences were accurately taken into account.

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POSSIBLE GENETIC CONSEQUENCES FOUND IN THE RESIDENTS OF BELARUS AFTER CHERNOBYL NUCLEAR ACCIDENT

Long-time study of the dynamics of embryonal anomalies found in the residents of Belarus using genetic monitoring system for medical abortuses (the analysis of more than 30,000 samples) and genetic monitoring system for newborners (annual control of about 80,000 births) showed

1/ 59% increase in the incidence of embryonal anomalies registered in the abortuses from the areas with Cs-137 contamination of more than 15 Cu/sq km for the period of 1987 to 1993;

2/ 79% increase in the incidence of malformed newborners for the same period in the same areas;

3/ no correlation between the growth of embryonal anomalies and total and individual doses of population exposure.

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PAPER TITLE USE OF A RADIOLOGICAL ACCIDENT TYPOLOGY IN THE PREPARATION OF EPIDEMIOLOGICAL PLANS

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ABSTRACT (See instructions overleaf)

After a radiological accident, epidemiological studies should be carried out in order to assess its health impact especially but not only on cancer risks. The implementation of studies on cancer raises various questions such as : what site of cancer should be followed or studied, will a cancer excess be detectable, what should be the kind and/or the size of a study ? The preparation of epidemiological plans should help to answer these questions for a variety of possible situations. For practical reasons, these plans will be based on a restricted number of « typical situations ».

Radiological accidents wich occured in the past and accident simulations on nuclear installations or transport of nuclear material were used to build up a set of accident scenarios. For all of these, numbers of exposed people, expected dose distributions, sites and numbers of excess cancers were calculated with appropriate tools. These parameters were then used to characterize and select the so called « typical situations » which are presented in this paper.

In some of these typical situations, the predicted cancer excesses are very low, under the level of statistical detectability. In others, the predicted excesses for specific sites of cancer are higher. Plans based on typical situations will provide guidance to define the objectives of epidemiological studies (e.g. description of cancer occurrence or cancer risk quantification), select the target population groups and choose the kinds of studies (mortality, morbidity) taking into account statistical power considerations.

The Impact of the Human Genome Project on Risk Assessment

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INTRODUCTION

The radiation protection approach to risk assessment assumes that cancer induction following radiation exposure is purely random. Present risk assessment methods derive risk from cancer incidence frequencies in exposed populations and associate disease outcomes totally with the level of exposure to ionizing radiation. Exposure defines a risk factor that affects the probability of the disease outcome. But cancer risk can be affected by other risk factors such as underlying genetic factors (predisposition) of the exposed organism. These genetic risk factors are now becoming available for incorporation into ionizing radiation risk assessment. Progress in the Human Genome Project (HGP) will lead to direct assays to measure the effects of genetic risk determinants in disease outcomes. When all genetic risk determinants are known and incorporated into risk assessment it will be possible to reevaluate the role of ionizing radiation in the causation of cancer.

The distribution of genetic risk determinants in time and space is governed by the biological and social processes involved in reproduction, such as the extent of inbreeding (endogamy) . We know already that endogamy causes the genetic risk determinants to be non-randomly distributed in populations [1]. These non-random distributions of genetic risk factors need to be considered in explaining cancer "hot-spots" observed by epidemiology.

Genetic determinants of cancer risk can be defined because the relevant phenotypes and their genotypes are being mapped and sequenced. Of 6456 genetic disorders identified in Online Mendelian Inheritance in Man (OMIM), 641 are associated directly, indirectly, or by implication with cancer. These cancer risk genes, when fully described, will define the list of genetic cancer determinants for use in risk assessment. Examples which impinge upon present risk assessments include cancer prone syndromes associated with ionizing radiation sensitivity.

ANALYSIS OF GENETIC DISORDERS ASSOCIATED WITH CANCER

Rapid developments in the HGP is leading to detailed knowledge about genes, providing information about genetic risk determinants. This information is stored in molecular biological databases which mirror the current knowledge of genetics and molecular biology. Information about those genes associated with cancer seems to be most relevant to risk assessment, as cancer risk is of most concern.

OMIM is the on-line version of V.A. McKusick's Mendelian Inheritance In Man [2]. OMIM describes all known genetic disorders and traits. The access interface uses Information Retrieval Experiment (IRX) software [3]. The Genome Data Base (GDB) is directly linked to OMIM. GDB collects genetic and physical chromosome maps, information about genes arranged by chromosome location, polymorphisms (normal and mutant gene variants), probes and associated information. GDB also provides accession numbers for links to nucleotide sequence repositories such as GenBank.

Because these databases can be interconnected, it is possible to link the phenotypic description of a particular disorder with molecular mutation at the DNA level and the detailed protein changes causing the disorder. This

allows us to define in molecular terms the genetic risk determinants we need to consider in ionizing radiation risk assessment. We built a set of computer tools to access these databases, analyze the data, and use it to evaluate hereditary contributions to radiation risk. Our goal was to define a set of all known phenotypes (physical manifestations of genotypes) and genotypes (underlying genetic blueprints) and all described biochemical events associated with cancer. Such a list is expected to contain the bulk of presently known or suspected genetic risk determinants; it is from this list that the targets of molecular epidemiology will be selected for analysis.

We operationally distinguished three major sorts of genetic effects as predisposing to cancer:
.1 direct associations,
.2 indirect associations and,
.3 molecular biological traits (biochemical events) associated with cancer.

Direct associations are those genetic disorders, in which a specific gene is involved, which when mutated manifests as cancer of a particular type. The primary outcome of mutation in such a gene is a cancer itself. Examples include Retinoblastoma (MIM# 180200) and Thyroid Carcinoma (MIM# 188550) which is also associated with ionizing radiation. This group includes phenotypes in which tumor suppressor genes and proto-oncogenes are involved.

Indirect associations include genetic disorders in which, for example, DNA repair is less accurate, leading to somatic mutation manifested as an increased incidence of cancer. Examples include Bloom syndrome (MIM# 210900), Neurofibromatosis (MIM# 162200), and Ataxia Telangiectasia (MIM# 208900) which is also associated with ionizing radiation sensitivity. This group includes phenotypes in which modulator genes are involved.

Molecular biological traits are a group of OMIM entries that describe all known molecular biological events leading to or associated with cancers. Examples include the Tumor Suppressor Gene, Hela Cell Type (MIM# 191181), Oncogene MYC (MIM# 164840), Transformation Gene: Oncogene AMV (MIM# 189990), Carcinoembryonic Antigen (MIM# 114890), and the DNA Damage Inducible Gene (MIM# 126335) which is induced by ionizing radiation.

We separated the 641 OMIM entries associated with cancer into direct, indirect, and trait subsets. The results are summarized in Table 1.

Table 1

type of association with cancer	# of disorders	# of known gene locations	# of detailed gene maps	# of sequenced genes
direct	94	53	27	15
indirect	107	52	24	20
traits	440	390	166	218

In Table 1 the number of disorders of any category were defined by reading the entries, checking for linkage to GDB, and determining the details of mapping and sequence data. The entries reflect progress in the HGP; mapping at increasing resolution is followed by sequencing the actual gene. Increasing numbers of genetic disorders associated with cancer are becoming known, and genes and map locations are being determined with increasing precision. The sequence information is what is ultimately required to develop molecular assays for the presence or absence of variants of this class of genes in individuals. All these disorders are in some way connected with cancer risk; the precise nature of the connection is one of the major outputs of the HGP.

Ultimately, all genetic disorders will be tied to mapped, sequenced genes. At this interim period, however, the 253 sequenced genes are primary targets of molecular epidemiology . Thus, we are primarily interested in those

mapped and sequenced genes whose phenotypes include sensitivity to ionizing radiation.

In risk assessment, the value of having a list of genes affecting cancer risk lies in the ability to combine molecular biological assays and epidemiological methods to define cancer risk sources in individuals and thus populations. Knowing the genes and their sequences gives us the potential for direct measurement of genetic risk determinants. By measuring doses and determining genetic risk factors, a better understanding of the ionizing radiation component of cancer risk will be obtained. Thus the presence or absence of genetic risk factors in individuals will affect the risk assessment process. Ultimately it will be possible to determine risk on the individual level and tie the risk to real causes.

Consider an example of a genetically determined, ionizing radiation sensitive disorder which is expected to affect risk assessment. Ataxia telangiectasia has long been a model human genetic disease for studies in radiation research, and ultimately in risk assessment. It belongs to the indirect set of cancer associated genes. AT is described in detail in a number of recent publications [4,5,6,7,8]. Cancer predisposition in heterozygotes, who carry one mutant copy of the AT gene, is estimated to be about 3 - 4 fold higher than the general population, and breast cancer risk in carrier women may be five-fold higher than non-carriers. It has also been hypothesized that breast cancer risks in irradiated carriers may be further increased [9].

While AT is a rare disease, up to 1.5% of the world population may be carriers of one defective copy of the gene. The relevance of AT to risk assessment is that heterozygotes, or carriers, may show both cancer predisposition and moderate radiation sensitivity. This means that up to 1.5% of individuals may be significantly more radiation sensitive than other members of the population, and at altered cancer risk, owing to carrying a mutant copy of the AT gene. Non-carriers will not have the risk determinant nor the attributable risk.

Remember that AT is only one of the 641 genetic risk determinants found in our search, and can account for only a proportion of genetically mediated or influenced cancer risk. Assays to measure the actual genetic risk determinants at the DNA sequence level are required to define the presence or absence of genetic risk determinants in individuals. Such direct assays for the presence or absence of gene variants linked with cancer predisposition, susceptibility, or expression will lead to a new understanding of the magnitude of ionizing radiation risk.

The distribution of genetic risk determinants in the individuals who make up populations is governed by biological processes. We have discussed two indicators of endogamy which are associated with non-random distribution of genetic risk determinants, namely ethnicity and consanguinity [1]. The genetic consequence of endogamy is that genetic risk factors located on DNA shared between group members is more likely to lead to observed genetic disorders. The AT variants described to date show that both ethnicity and consanguinity affect their temporal and spatial distribution, and their associated cancer risks in populations.

Acquiring the ability to assay genes directly, to determine both their relationship to disease and their distribution in populations will increase the certainty of the risk assessment process by identification and direct measurement of its deterministic genetic risk factors. At the same time it will reduce the uncertainty of risk assessment by decreasing the probabilistic components of the process.

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The genetic consequens of exposure.

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The results of the study of genetic consequences of external gamma-irradiation of man and animals to 1 Sv are given. The investigation was performed in 3 groups under different conditions of exposure of the population:
- among the people of Russia and Bielorussia exposed due to the Chernobyl accident;
- among the people living on the Tetscha river basing in the South Urals;
- among the occupational contigent of "Mayak" and the members of their families;

The experimental estimation of genetic consequences was made on the offsprings of the white male rats.

The male rats were irradiated dayly for 10 -1 5 days with external gamma- radiation of different dose power. The range of the doses received by the animals was approximated to the conditions of the exposure of man to the interval from 4 to 79 cSv for a year.

It was shown that:
- the most distinct effects were observed in the groups "237" cSv and "158" cSv;
- the main contribution into the remote effects of radiation action on the reproductive function of the males gives the loss on the preimplantation stage of pregnancy. The statistically reliable increase in group "237" cSv of the stillbirths incedence and the increase (very important, to 25% compared to the control) of dominant lethality incedence attract attention. In addition, the decrease of the fertility and the decrease (not very significant) of the number of litters and first of all because of the decreased number of males in the litters
were reveated;
- the statistically reliable growth of the birth weight of the newborns in groups "237" cSv and "158" cSv that is preserved in the following periods of development;
- the influence in the dose range investigated doesn't bring the increase of the mortality of offspring or the impairment of forming sensoro- motor reflexes in any group;
- the reliable impairment of development of locomotor function in offspring of the animals in group "237" cSv and the impairment of curious behaviour at the age of 30 days in groups "237" cSv and "158" cSv;
- the change of the rat conduct in the crest- line labyrinth in the offsprings of animals received summary absorbed dose 237, 158 and 48 cSv.

Thus, when estimating the remote genetic effects of the exposure it is necessary to take into account, first of all, the integral indices.
It's reasonable to study the effects on the large number of animals with the most detailed dosages in the range to 100 cSv and using more sensitive methods. It's absolutely necessary to be supported financially for the new experimental study.

With the purpose of studying the possible unfavogable outcomes of the exposure of the population due to the Chernobyl accident, the investigation was performed in the Mogilev, Gomel and Briyansk regions. The summary doses of exposure of the man living in 226 villages examined during the period from 1986 to 1992 were estimated. The doses of external radiation varied from 0,31 to 11,0 cSv. The doses of internal radiation varied from 0,08 to 4,19 cSv. The summary doses were from 0,39 to 15,19 cSv.

The frequencies and the structure of untoward pregnancy outcomes (UPO), in general, were studied. The frequencies of UPO reflecting the character of reproduction in the population living on the contaminated territories. It's necessary to point out that the frequencies of UPO in the populations living on the territories contaminated with radionucleids are reliably higher (13,4 c 1,03 per 100 pregnant women) compared to the control population (4,31 c 0,35 per 100 pregnant women).

When studing the UPO frequencies in the dose groups (group 1 - to 3 cSv; group 2 - from 3 to 5 cSv; group 3 - from 5 to 10 cSv & group 4 - more 10 cSv), it was revealed that they received the meaning 8,0 c 1,36 per 100 pregnant women (in group 2) to 13,5 c

1,2 per 100 pregnant women (in group 4). The UPO frequencies was reliably higher only in the summary dose more than 10 cSv.

The study of antropometric indices of newborns showed the decrease of the number of children with the optimal weight at the birth. The specific weight of such children was during the preaccident period 32,3% & 34,8%, and during the postaccident period 24,1% & 26,9%, in girls and boys respectfully.

The adaptation of the the newborrns to the enviroment during the first six years after the accident was decreased. Thus, the meanings of the indices reflecting the background medico- genetic situation during the preaccident period and during the first years after Chernobyl accident were dermined.

The untoward pregnancy outcomes was studied in the families exposed due to the breaks of radionucleids waste products into the Tetscha river in the South Urals. The study showed that at the average dose on the gonads of the parents to the moment of the conception equal to 12 - 15 cSv, the UPO frequency when only the father was exposed was 11%, when the mother was exposed was 9%, when the both parents were exposed was 6,1%. The UPO frequency in the control was 5,8%.

The most reliable data were in the families of the personal at the atomic industry plant "Mayak" in the South Urals, where not only the UPO frequencies but also the offspring mortality from the moment of the
birth to the age of 45 years old were taken into consideration. The average doses on the gonads of the fathers were 144 cSv, on the gonads of the mothers were 142 cSv. The average doses during the pregnancy period varied from 0,1 cSv to 233 cSv, with the mean one 19 cSv.

In the investigation the UPO frequency increase and the growth of the offspring mortality correlated with the increase of the gonad dose of the external gamma-radiation of the mother (1.04 cases per 1 Sv). The constant of the doubling gonad dose of the mather exposure was 1,4 Sv.

Thus, the material evidence of the remote genetic effects of the human exposure was, for the first time, received and the estimate of the doubling dose on the gonads when the mother is exposed was determined to be 1,4 Sv.

IN ESTIMATE OF THE HEALTH STATUS OF THE OFFSPRING OF INDIVIDUALS SUBJECT TO OCCUPATIONAL EXTERNAL GAMMA-EXPOSURE

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The assesment of genetic effects of radiation exposure is one of the critical and important problems. It is known that normal postnatal development is genetically determined. Health status indices (physical development, stability to environmental factors and other) are under the control of many genes functioning in close interaction with each other. The injury of genes may lead to the disorder in health status occuring in the offspring of irradiated individuals. But on the other hand child's development may be subjected to negative actions of external and internal factors on the different levels of development regulation.

The purpose of the present investigation was to give a complex estimate of the health status of grandchildren of individuals subjected to chronic occupational radiation exposure before the conception and to estimate the contribution of radiation factor to the occurence of revealed defects in the children with regard to a large complex of various non-radiation factors.

The main group consisted of 1557 grandchildren of individuals occupationally exposed to chronic irradiation previous to conception. Three groups were chosen depending on which of the grandparets had come into contact with sources of ionizing radiation. The first group is represented by 830 grandchildren, who had irradiated grandfathers. The 259 children whose grandmothers had been subjected to radiation exposure were included in the second group. The 468 children whose both grandparents (grandfather and grandmother) had been irradiated were included in the third group. Total doses accumulated in the gonads by the time of conception of the parents of the observed children were calculated (table 1).

Table 1

Dose characteristic of grandparents occupationally exposed to external gamma-exposure

Group	On the side of the child	Dose before conception, mSv: Dose range	Mean equivalent dose	Percetage of persons who had exposed with doses more 1000 mSv	Dose during pregnancy, mSv: Dose range	Mean equivalent dose
Grandfathers	mother	14-4934	566 ±59.7	14.9	none	none
	father	11-8823	673± 7.7	18.9	none	none
Grandmothers	mother	13-1106	173 ±42.1	3.3	3-281	45±11.1
	father	15-2000	375± 84.4	16.6	4-660	85±25.2
Grandfathers	mother	07-5440	904±158.5	28.7	none	none
and	father	06- 6754	953± 190.0	27.0	none	none
grandmothers	mother	03-0995	267± 37.5	0	2-2184	80±49.9
	father	01-5196	588 ±179.5	13.6	1-63	19±3.0

Most of the grandmothers (77.8-81.7%) were subjected to irradiation not only before conception but also during pregnancy. The parents of observed children in some of the cases also worked at the nuclear plant. Total doses of gamma-exposure at the time of conception did not exceed the maximum permissible doses.

The control group consisted of 466 children of similar age belonging to the second generation of individuals who lived in the same town during the analogous period of time but had no occupational exposure to ionizing radiation. The children were born in same maternity hospital and visited the same preschool esteblishments, and were under observation by the same physicians as the children of the main group.

The complex estimate of the observed grandchildren's health state included dynamic study of physical development (anthropometric indices, harmonizing, age dynamics and distribution of physical development disorders) the terms of biological maturation (terms of first dentition and its exchanges; development of psychomotor skill; forming of secondary sexual characters; assessment of school maturity), the range of morphofunctional characters (state of cardiovascular system; state of peripheral blood; size of thyroid gland, ets.) the age dynamics of disease level and structure, the level and structure of congenital malformations. For the ranging of the influence of professional exposure and other non-radiation factors multufactorial analysis was used (claster, rank correlation and logistic regression). Prognostic coefficients for each index were calculated by stepwise logistic regression, and the probability of revealed disoders in observed children was found. The observed children, both the main and control groups, were born in the favorable state of the health and had no significant pathology in the neonatal period.

Anthropometric data served as a basis of estimate of the physical development. A standartized technique was used to assess height and body weight. Mean values of body weight, height and chest circumference in male and female children were calculated in every age group in period from newborn to 15 years age. Height was categorized according to the age specific mean and standard deviation. Special attention was given to the correlation between the main anthropometric values. The degree of discrepancy between actual body weiht and the height-related average weight in disharmonious physical development was calculated in percentages, and children with hypotrophy, overweight and obesity were identified.

Average anthropometric indices in children of the main group did not differ from those in children of the control. The comparison of levels and harmony of physical development in children of the main and control did not revealed significant differences either. Age dynamics of antropometric indices in grandchildren of occupationally irradiated grandparents coincided with those of controls. The range of observed disorders (intrauterine hypothrophy, obesity, overweght and hypotrophy at the age of 1-15 years) was similar in the main and control groups.

There was no relationship between occurence of disorders in the physical development of the observed children and occupational irradiation of their parents and grandparents by means of the multifactorial analysis. The relationship between the incidence of the hypotrophy and a number of non-radiation factors was confirmed. Thus, impaired uteroplacental blood circulation during pregnancy gives rise to disorders in fetus feeding and, consequently, results in intrauterine hypotrophy, specifically: increased abortion risk, combined gestosis and anemia of mother in pregnancy. The presence of number cardiovascular disease as well as the mother's age at giving birth, being below 20 or above 30 years is unfavorable. A ralationship was found between the development of hypotrophy at the age of 1 year and pathologic course of pregnancy (threatening abortion, pharmacotherapy in the second half of pragnancy), as well as with an indication on previous artificial abortions in the mother's anamnesis.

No differences in disease incidence level, structure of the morbidity, per cent of often being ill children, frequency and structure of the congenital malformations between the main and control groups were found. Acute respiratory diseases during the second year of life occurred more frequently in children whose grandparents had contact with ionizing radiation sources than in control. In other age groups no significant differences between the main and control groups were found. Increased disease incidence during the second year of life coincided with the begining of the children's visits of preschool esteblishments and the period of adaptation for the new microsocial conditions. The study of the frequency and structure of congenital malformations among grandchildren of the observed cohort did not reveal significant differences between the main and control groups (table 2).

Table 2

Rate and structure of congenital malformations (in case on 1000 children of observed cohort)

Types of malformations	Main group	Control group
Number of children	1557	466
All congenital malformations	54.6	81.5
Defects of osteomuscular system	27.0	40.7
Defects of urogenital system	20.5	23.6
Defects of gastroenteric system	2.6	2.1
Defects of cardiovascular system	2.6	8.6
Defects of skin and its appendages	0.6	2.1
Defects of neurous system and sense organs	1.3	0
Defects of face and neck	0	4.3

For the ranging of the influence of professional exposure and other non-radiation factors multifactorial analysis was used. There was no relationship between occurence of anomalies in the health state of observed children and occupational irradiation of their parents or/and grandparents. The role of factors of non-radiation character has been confirmed. The relationship of acute respiratory disease incidence at early age with a number of non-radiation factors was established: (1) low or high age of the mother at birth; (2) indications on earlier abortions; (3) gestation age of the newborn - premature or overmature; (4) occurence of allergic diathesis; (5) rachitis in infants; (6) artificial feeding in the first year of life. Inflammatory diseases of genitals of infectious etiology and late toxicosis of mothers in pregnancy predisposed to more severe course af acute respiratory diseases. Occurence of intrauterine hypotrophy, allergic diathesis and rachitis resulted inull a high rate of acute disease in children. Such 'maternal' factors as early placentation, pharmacotherapy and threatening abortions as well as early additional feeding and unfavorable living condition contribute to early sensibilization with development of signs of allergic diathesis in the first year of life. There was no relationship between of congenital malformations in observed children and occupational irradiation of their grandparents by multifactorial analysis. Such factors as low or high age of the mother at the time of the labors, chronic diseases ot mother and congenital malformations of the uterum and vagina predisposed to occurence of malformations in observed children.

Biological age of each child was estimated on the basis of level of morphological criteria development (length of body, body mass and permanent dentition development) was compared with passport age. In a half of children at age 6 years and 1/3 of children at age 7 years the biological age was detained in comparison with passport age. The differences between the main and control groups were not found. The terms of apperance of secondary sexual characters in boys and girls who were grandchildren of persons exposed to occupational radiation did not differ from those in children of control group and accorded with normal development criteria. The cases of changing the rate and terms of puberty among children of examined cohort were not found.

Thus, in dynamic observation during their 15 years of life of children whose grandparents had been subjected to occupational external gamma-exposure, it was found that the range and structure of revealed pathologies coincided with those in the control groups and corresponded to national statistics data. Analogous results were obtained when evaluating the morphofunctional status of the children examined. Occupational gamma-exposure of grandparents had no significant influence on the health status of observed children. The importance of social, biological and medical factors in the development of observed disorders was confirmed.

USE OF COSYMA IN THE DESIGN OF EPIDEMIOLOGICAL STUDIES IN CASE OF RADIOLOGICAL ACCIDENTS

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I. INTRODUCTION

In case of a radiological accident, the epidemiological assessment of health effects in exposed populations is necessary to verify risk predictions, to detect any unexpected effect and to improve the knowledge on the effects of low protracted doses of ionizing radiation (in some accidental situations, most of the people would be exposed to such low doses). This assessment is also useful to determine health needs of exposed populations, to adapt health care resources and to provide the "media" and the public with independent and validated information.

The Chernobyl accident demonstrated the necessity of a preparation for the implementation of epidemiological studies in such circumstances, in order to get the most accurate information on the health consequences. Although effects may appear years after the accident (e.g. cancers), the lack of early data collection on exposure and the number of people lost of follow-up make the evaluation of cancer excess risks very difficult. By simulating various radiological accidents with the code COSYMA [1] and predicting cancer risks for all of them, it is intended to calculate the key parameters that are necessary for the prompt implementation of effective epidemiological studies.

II. GENERAL OBJECTIVE

This study focuses on the assessment of post-accidental cancer risks. For this purpose, when designing epidemiological studies, the scientist has to answer the following questions :

* what types of cancer are mostly expected after the accident ?
* what population groups should be followed and how many subjects could or should be included in a study ?

The following parameters should be assessed for the entire population as well as for various groups (e.g. children who are more radiosensitive) :

* average doses to various organs (e.g. thyroid gland, bone marrow...) and to the whole body,
* number of exposed people in various dose intervals,
* number of expected spontaneous and predicted excess cancers for various organs and tissues in the exposed group,
* statistical power of any planned study on cancer, that is its probability to detect an excess risk.

Then, the epidemiologist has to define a dose level to classify the subjects in two groups: those with significant doses who will be considered as "exposed" and those with lower doses, "not exposed". The choice of this level could be optimized according to statistical power criteria. This means that one has to compromise with the expected risk in the followed group and the size of this group in order to be able to detect the excess risk statistically.

The work presented here aims at calculating the above parameters and the statistical power for various radiological accident scenarios. From the results, a typology of epidemiological studies will be constructed which could be used to guide the early collection of relevant data if a real accident occurred.

III. METHOD

The study was carried out as follows:

1. definition of a set of radiological accident scenarios;
2. for each scenario, simulation of N accidental situations and estimation of dose distributions, sites of expected cancers, and associated risks;
3. classification of the N accidental situations according to health effects and epidemiological criteria (number of exposed people, distribution of whole body and organ doses, types of expected cancers, excess risk values) ;
4. extraction of n typical situations among the N simulated ones;
5. statistical power calculations for the n situations to assess the probability to detect cancer excesses by epidemiological studies.

IV. DEFINITION OF A SET OF SCENARIOS

The list of possible scenarios of radiological accidents should be as exhaustive as possible. The characterization of an accidental situation is based on the type of accident (power reactor, reprocessing plant, transportation), the amount of radioactivity and the kind of radionuclides released into the environment (source term), the location of the accident (local meteorology and demography), and the set of countermeasures undertaken (evacuation, sheltering, stable iodine distribution, relocation). For plant calculations, three typical french sites were selected: one PWR located in the Loire valley, another one located in the Rhone valley, and the reprocessing plant of La Hague. The main characteristics of these sites are presented in Table 1. For transportation accident calculations, there is no particular location, and by looking at the routes of transportation a few rural and urban sites for potential accidents can be defined.

V. USE OF COSYMA IN SIMULATIONS

COSYMA was chosen among the various accident consequences assessment codes because of its power and flexibility. It performs all the calculation steps from activity release and dispersion to cancer risks, for any site of interest, using specific data input and allowing the retrieval of non-standard outputs. As a probabilistic code, COSYMA realizes a large number of simulations for one accident in the same run just by changing the weather sequence.

PWR source terms were constructed using the inventory of the COSYMA users intercomparison exercise [2] and release fractions for french nuclear reactors S1, S2, S3 (Table 2) [3]. For the reprocessing plant, two source terms were chosen arbitrarily, based on the actual radioactive materials present in the plant and on the radioactive decay of reprocessed PWR fuel (1g Plutonium and 0.01g Curium releases, Tables 4 and 5). For the transportation scenarios, several source terms still need to be defined, according to the materials transported, the modes of transportation, and the types of packaging. The post-accidental scenario has identical main options for all installations and source terms. The ingestion pathway was not treated in this study because of its unsatisfactory modeling in COSYMA. The intervention levels for the various countermeasures were drawn from ICRP 40 recommendations (Table 3) [4]. Both low and high levels were used for evacuation, sheltering and stable iodine prophylaxis. For long term actions, only the low intervention level was applied.

For the epidemiological calculations, the execution procedure of COSYMA was modified to save two non-standard output files. These files contain the individual organ doses and the individual risks for the late health effects calculated for all weather sequences. A program was written to extract the data from these files for one chosen weather sequence; it calculates the distribution of whole-body and organ doses in the population and performs statistical power calculations for prospective studies. Figure 1 shows the statistical power of a cohort study (prospective follow-up of an exposed group) as a function of the dose level defined above, for one accident in the nuclear power plant of the Rhone valley with wind oriented north. In this example, a maximal statistical power is observed for a dose level between 10 and 40 mSv. Under 10 mSv risks are low and could not be detected despite a large number of exposed subjects. When the dose level exceeds 40 mSv, the statistical power decreases despite higher risks, as the number of exposed subjects has become too low. A statistical power of 0.8 is generally accepted to conduct an epidemiological study.

VI. CONCLUSION

COSYMA is a powerful code for the assessment of a wide range of accident consequences. It has been satisfactory because of the possibility to get access to the code itself and to change options like release duration or outputs. COSYMA is a probabilistic code, but, in this application, it has been used in a deterministic way by extracting the results for one weather sequence only.

From an epidemiological standpoint, a few issues limit the use of COSYMA. Following a radiological accident, some tumors are expected, like thyroid cancers or leukemias. For these tumors, survival rates are rather good and therefore, to assess the health impact in epidemiological studies, morbidity data should be preferred to mortality data as those used in COSYMA. Indeed, the count of incident cases instead of deceased cases would increase the statistical power which is a critical parameter for the study of low protracted dose effects. In addition, it would be of great interest to look at different age classes and different periods of follow-up. It is not feasible by COSYMA which evaluates only cancer risks for the lifetime and the general population, since population and risk data are not detailed enough in the code.

VII. FIGURE AND TABLES

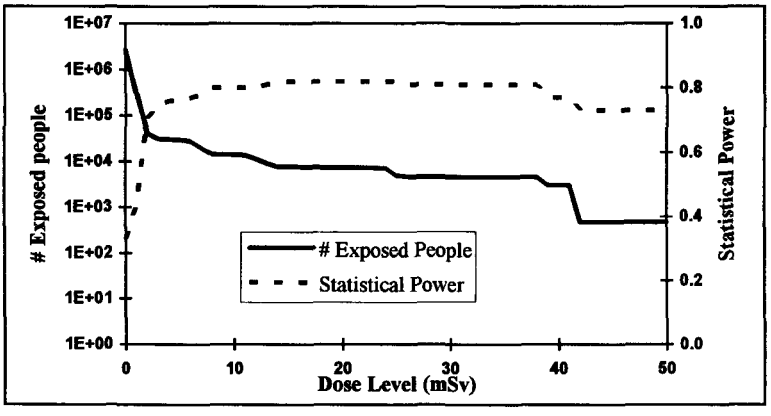

Figure 1 : Total number of exposed people and statistical power to detect an excess incidence of cancers as functions of the dose level that defines the exposed group. (Results for the thyroid gland)

Table 1 : French Sites Selection for Plants

Site	Main Wind Direction	Inhabitants within a radius of : 10 km	20 km	40 km	100 km
La Hague	West / South-West	7 841	84 098	194 158	638 487
Rhone valley	North ("Mistral")	34 977	92 372	340 463	1 941 728
Loire valley	-	20 769	56 342	430 897	1 967 423

Table 2 : Release fractions for PWRs

	Xe,Kr	I	Cs,Rb	Te,Sb	Ba,Sr	Co,Ru	La,Act
S1	8.10^{-1}	$6.6\ 10^{-1}$	$4\ 10^{-1}$	$8\ 10^{-2}$	$5\ 10^{-2}$	$2\ 10^{-2}$	$3\ 10^{-3}$
S2	$7{,}5.10^{-1}$	$3.2\ 10^{-2}$	$5.5\ 10^{-2}$	$5.5\ 10^{-2}$	$6\ 10^{-3}$	$5\ 10^{-3}$	$8\ 10^{-4}$
S3	$7{,}5.10^{-1}$	$8.5\ 10^{-3}$	$3.5\ 10^{-3}$	$3.5\ 10^{-3}$	$4\ 10^{-4}$	$3\ 10^{-4}$	$5\ 10^{-5}$

Table 3 : Intervention Levels.

COUNTER-MEASURES	INTERVENTION LEVEL (mSv)	
	low	high
evacuation	50	500
sheltering	5	50
stable iodine	50	500
relocation	50*	50*
resettlement	5*	5*

*** projected dose, integration time 1 year**

Table 4 : Source term for the Pu release.

Isotope	mass %	Activity (Bq)
^{238}Pu	1.5	$1.37\ 10^{10}$
^{239}Pu	58.0	$1.98\ 10^{9}$
^{240}Pu	23.0	$2.80\ 10^{9}$
^{241}Pu	7.5	$4.12\ 10^{11}$
^{242}Pu	5.5	$1.15\ 10^{7}$
^{241}Am	4.5	$1.83\ 10^{11}$

Table 5 : Source term for the Cm release.

Isotope	mass %	Activity (Bq)
^{242}Cm	0.5	$8.84\ 10^{9}$
^{243}Cm	2.4	$6.61\ 10^{8}$
^{244}Cm	97.1	$4.19\ 10^{10}$

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IRPA9
1996 International Congress on Radiation Protection
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Abstract No.	30869
Receipt	
Author	20729
Acceptance	
Mini-Presentation	

PAPER TITLE Epidemiological evaluation of the radiation risk among residents of the Techa river area

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The report is based on the information obtained as a result of a 40-year period of follow-up of people exposed to radiation due to discharges of radioactive wastes from "Mayak" industrial facility into the Techa river. The doses to the red bone marrow (RBM) ranging from several mSv to 4,000mSv. The registry of the exposed population (26,437 exposed people, 3,091 prenatally exposed persons, and 22,158 children of exposed parents), the registries of mortality (6,439 death certificates during the period 1950-82) and registry of cancer morbidity serve as the basis for conducting epidemiologic cohort studies. The leukemia risk estimated on the basis of the linear model of absolute risk was 0.85 (0.24-1.45) per 10,000 person-y Gy of the dose accumulation in RBM, and solid cancer risk, estimated using linear model of relative risk, was 0.65 (0.27-1.03) per Gy of dose accumulated in soft tissues. Estimates of gonadal doubling doses were made based on the incidence of stillbirth, miscarriages, early neonatal mortality and congenital lethal developmental defects. The dose range proved to be as wide as 0.2-4.8 Sv.

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April 14-19,1996
Vienna, Austria

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Abstract No.	98-874
Receipt	
Author	20734
Acceptance	
Mini-Presentation	

PAPER TITLE Epidemiologic data on the basis of the registry of workers of "Mayak" Production Association

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MAJOR SCIENTIFIC TOPIC NUMBER 2.2 (see page 7)

ABSTRACT (See instructions overleaf)

"Mayak" Production Association is the first in Russia nuclear enterprise for weapon plutonium production, that was put into operation in the Southern Urals in 1948. The registry includes 14099 male and 4776 female workers, who began working at the main plants (nuclear reactors, radiochemical plant, plutonium production plant) in 1948-1972. Vital status is known for 89% of the workers included in the registry. Epidemiologic study is being carried out among the workers with known vital status and available dosimetry data. Film badge dosimetry data are available for 10881 males and 3261females. Average cumulative external gamma dose for them is 87 cGy. Dose distribution is given in the paper. Data on internal doses from plutonium are available for 3239 males and 1097 females. 3372 male and 645 female workers died of all causes. The highest mortality is among those, who began working in 1948-1958, because of higher representation of elderly people. Preliminary epidemiologic analysis among these workers has revealed increased leukemia mortality in male workers caused mainly by external radiation exposure, increased mortality from lung cancer and liver neoplasms (in particular haemangiosarcoma) caused predominatingly by internal exposure from plutonium.

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Vienna, Austria

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PAPER TITLE SOME ISSUES at the EPIDEMIOLOGICAL HEALTH STATE RESEARCH of CHILDREN LIVING in CHELYABINSK-65 and CHELYABINSK-70

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 (see page 7)

ABSTRACT (See instructions overleaf)

Subject of report is the epidemiological study for the critical group of the Chelyabinsk-65 residents including children born in 1974-88. The children of the same age living in Chelyabinsk-70 will be included in the controlling group. The study will be realized with continuous method on the basis of registers including near 30000 children from the two towns. The study of children diet and animal and vegetable products contribution, evaluation of the critical components of children diet as radioactivity will be realized. Radiogenecal risk assessment will be based on the analysis of the general and the oncological infant mortality using the model of the linear dependence of incremental relative risk from the dose. The effective doses to children will be calculated with respect to the amount of the radioactive materials released to the atmosphere using the model of radionuclide distribution in the environment.

CHILDHOOD LEUKAEMIA AND LOW-LEVEL RADIATION - ARE WE UNDERESTIMATING THE RISK?

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INTRODUCTION

In November 1983 a television documentary claimed that the number of cases of childhood leukaemia which had occurred in the coastal village of Seascale, situated next to the nuclear complex at Sellafield in Cumbria, England, was around 10 times the number of cases expected in the period since the start of nuclear operations at Sellafield in 1950. This report was based upon just 6 cases of childhood leukaemia. The television programme generated public concern in the UK and the Government ordered an independent inquiry into the claim. In summary, this official inquiry confirmed that an excess of childhood leukaemia cases had occurred in Seascale, although the cause of this excess could not be identified (1). The inquiry recommended that a programme of scientific research should be undertaken to investigate the possible causes of the Seascale childhood leukaemia "cluster". This research has led to a variety of suggested explanations for the Seascale "cluster" and the history of these investigations illustrates the great care with which "cluster" reports need to be interpreted.

THE PROBLEMS OF EPIDEMIOLOGICAL INFERENCE

Epidemiology is the study of patterns of disease in groups of humans with the objective of identifying the causes of disease. It is an observational (i.e. not an experimental) science and as a consequence the findings of epidemiological studies must be interpreted with considerable caution. Although a statistical association between a disease and some factor may be indicative of an underlying cause-and-effect relationship, alternative explanations must always be considered. These explanations are that the association could have been produced through the effects of chance, or that certain biases are present within the study such that the association is an artefact, or that the association is due to confounding so that an underlying causal factor is related to both the disease and to the non-causally associated factor identified in the study.

The problems of interpretation of epidemiological associations have generated a considerable literature over the years. The eminent British epidemiologist, Sir Austin Bradford Hill, suggested nine criteria of causality against which epidemiological associations could be judged, and these are still used today in the scientific assessment of associations (2). Among the more important of the Hill criteria are the strength of the association, the consistency of the association across a variety of studies, evidence for a dose-response relationship (a "biological gradient"), and the evidence of biological plausibility.

The results of the studies carried out in the wake of the Seascale "cluster" must be viewed in the light of the nature of epidemiological research. They are a cautionary tale for those interested in the interpretation of epidemiological findings.

A PLETHORA OF "CLUSTER" REPORTS

The Seascale "cluster" generated other reports of childhood leukaemia "clusters" in the vicinity of nuclear installations in the UK, apparently indicating consistent evidence for a raised risk of childhood leukaemia around nuclear sites. These reports generated considerable publicity and independent investigations were conducted as a consequence. Closer examination of these findings, however, reveals that the evidence is not as persuasive as would appear at first sight (3,4). A number of the studies would seem to have been conducted on the basis of some prior knowledge of the pattern of childhood leukaemia in the vicinity of the particular installations investigated, and this prior knowledge

complicates the interpretation of results. As a consequence, secure statistical and epidemiological inference is lacking. Indeed, a recently published study of childhood leukaemia around nuclear installations in England and Wales which did not suffer from these sorts of inferential problems found that, apart from Sellafield, the evidence of any general spatial relationship between nuclear installations and childhood leukaemia was "extremely weak" (5). The "cluster" investigations carried out around nuclear sites do demonstrate the need for particular care in the design and execution of epidemiological studies carried out on an issue which is receiving wide publicity, with the consequent pressure on researchers to produce results.

POSSIBLE EXPLANATIONS

Direct Exposure to Ionising Radiation

Initial reactions to the report of the Seascale childhood leukaemia "cluster" were that the excess cases were due to an unpredicted effect of exposures to radionuclides discharged from Sellafield, or that the risk of radiation-induced childhood leukaemia had been underestimated. However, detailed radiological assessments have consistently demonstrated that the radiation doses received as a consequence of Sellafield discharges were too small by at least a factor of 100 to account for the observed excess of cases (6). By the end of the 1980s, it was widely acknowledged that direct exposure to radiation was most unlikely to be the cause of the Seascale "cluster" (7,8). This conclusion was reinforced by a recent comprehensive reassessment of radiation-induced leukaemia risk in Seascale (9).

Indirect Exposure to Ionising Radiation

In February 1990, the initial results of the West Cumbria leukaemia and lymphoma case-control study were published which suggested that the cause of the Seascale "cluster" might be indirect rather than direct exposure to ionising radiation (10). The study found a statistical association between childhood leukaemia and the radiation dose received by fathers while employed at Sellafield before the conception of their children. The authors suggested that the association was sufficient to account for the excess of childhood leukaemia in Seascale. This study received appreciable publicity and generated anxiety among radiation workers; but the association was based upon just 4 cases of leukaemia and the scientific support for a causal interpretation of association was, at best, weak. Nevertheless, this study led to one of the largest trials concerned with a personal injury claim to be heard in the English courts (11).

Evidence which has accrued since 1990 has demonstrated that the association is confined to those offspring of the Sellafield workforce who were born in Seascale, and does not extend to the great majority of the children of Sellafield workers who were born outside this village (12,13,14,15). Further, the association has not been found in any of the substantive studies carried out elsewhere (16). A causal interpretation of this association has now effectively been abandoned by scientific authorities (17,18). It would seem that this is an example of a statistical association which has been generated by the play of chance.

Population Mixing

One fascinating consequence of the Seascale "cluster" report is the series of studies conducted by Kinlen and his colleagues into the effect of unusual population mixing upon the risk of childhood leukaemia, which suggests an infective basis for childhood leukaemia. Sellafield is a large industrial complex sited in a remote rural area, and the employment demands of the site have generated unusual population movements in its vicinity. Kinlen has proposed that the consequent population mixing has produced an unusual pattern of infections which could elevate the risk of childhood leukaemia. He has now produced compelling evidence from a variety of detailed studies involving unusual population mixing in areas not associated with nuclear installations that such mixing does raise the risk of childhood leukaemia (19,20). This would appear to be the explanation for the Seascale "cluster"

which carries the most convincing scientific support. Kinlen's studies illustrate just how important it is to consider confounding as a possible explanation of an epidemiological association.

CONCLUSIONS

The Seascale childhood leukaemia "cluster" can be interpreted as indicating that the risk of childhood leukaemia arising from low-level exposure to ionising radiation has been underestimated. Indeed, several variants of such an interpretation have been advanced. These include exposure to particular radionuclides, an underestimation of the radiation risk coefficient for childhood leukaemia, and the existence of a previously unrecognised risk of childhood leukaemia from the preconceptional irradiation of fathers. However, the scientific assessment of epidemiological associations is a complex matter, and such associations must be interpreted with caution. It would now seem most likely that the Seascale "cluster" does not represent an unanticipated effect of the exposure to ionising radiation, but rather the effect of unusual population mixing generated by the Sellafield site which has produced an increase in the infection-based risk of childhood leukaemia. This episode in the history of epidemiological research provides a timely reminder of the need for great care in the interpretation of novel statistical associations.

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NUCLEAR INSTALLATIONS AND CHILDHOOD LEUKAEMIA - TESTING THE HYPOTHESES, EXPLORING THE IMPLICATIONS - A REVIEW

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INTRODUCTION:

Two hypotheses have, since 1983, been pursued in relation to nuclear installations and childhood leukaemia. One is environmental, the other occupational. The transition from one to another and the segregation of one from another have often been unclear as have been the implications to be drawn from one as opposed to the other. Examination of the evolution of the two hypotheses demonstrates a number of phenomenological patterns common to modern health scares which makes them comparable to other such events. The scientific and public policy inferences which may be drawn from these events are considered by examining the processes concerned with testing these hypotheses.

ANALYSIS OF PUBLISHED STUDIES

The sentinel reports relevant to both hypotheses derive from population sets in the vicinity of the Sellafield nuclear generating and reprocessing complex in West Cumbria[1,2]. The later occupational hypothesis derived from multiple analyses of a geographical subset (Seascale) which was being examined in pursuit of the environmental hypothesis. Thus it was drawn from what was occupationally a "sub" sub-population. In each case a similar cycle of studies was pursued in exploration of the theories. Figures 1a and 1b show the secular trends in observed/expected or in relative risks and associated confidence intervals relevant to the two hypotheses. Both sets of data show a similar pattern of regression to the mean after the sentinel study.

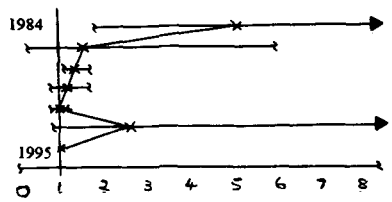

Fig. 1a Childhood leukaemia and the "environmental" hypothesis, summary of studies. O/E or RR and CIs. Unweighted. Highest exposed group in each study.

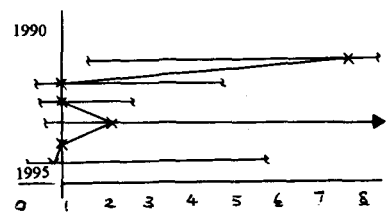

Fig. 1b. Childhood leukaemia and the "occupational" radiation hypothesis, summary of studies. O/E or RR and CIs. Unweighted. Highest exposed group in each study.

When considering the characteristics of the studies that were performed to test the two hypotheses, there is a strikingly similar pattern. Scientific work in the UK was initially concentrated on nuclear establishments in three areas, Sellafield[1], Dounreay[3] and Aldermaston/Burghfield[4]. In the geographical studies, significant excesses of childhood leukaemia were observed in all three studies. However the discharge history of the sites differed by orders of magnitude although derived doses were, by comparison with background, minute in all cases. By contrast the occupational studies carried out at Dounreay[5] and Aldermaston/Burghfield[6] did not support the preconceptual paternal irradiation (PPI) hypothesis but had vanishingly small power anyway.

Later and larger environmental (geographical) studies in the UK and elsewhere have continued to demonstrate a pattern of regression to the mean. For a number of reasons this is somewhat surprising. Firstly because the regression has continued despite the discovery of a slightly elevated risk of childhood leukaemia round potential nuclear sites generally historically. Secondly and paradoxically because such excesses of childhood leukaemia might actually be expected, at least from time to time, if an alternative, population mixing hypothesis were to be plausible.

Difficulties in discerning the crucial nature of the transition and separation of the environmental and occupational hypotheses may most relevantly be addressed by examining the key paper initiating the transition, that of Gardner[2]. This paper omitted to identify the critical importance of its derivation of the occupational hypothesis from a "geographical" sub-population of the Sellafield workforce who happened to live or to have lived in the village of Seascale. This turned out to be about 7% of the workforce and, on average, they had a slightly lower radiation dose than other workers. Thus it was left to others to infer that the most apposite test of the PPI hypothesis was by re-examining it in the whole of the population of Sellafield radiation workers from a subset of whom it had been derived[7]. When eventually performed, this test was negative. As with this study, other tests of the occupational hypothesis were uniformly negative.

The population mixing hypothesis, attributable to Kinlen, proposed that childhood leukaemia was the rare consequence of processes initiated by common viral infections and thus an excess of leukaemia would manifest in situations of "unusual" exposure to infection[8]. Specifically this would be unusually common occurrence of infection or particular liability to exposure to new strains of infection. The hypothesis was derived a priori and then tested initially in "new towns" (areas of rapid, mass resettlement) in the UK, and then in relation to military camps, wartime evacuation etc. The phenomenology is illustrated in Fig 2 and may be contrasted with Figs 1a and 1b. Of particular relevance to the nuclear debate are two other studies by Kinlen in pursuit of the mixing hypothesis. The first of these examined the temporary and recurrent migration patterns of Scottish workers serving the development of the Shetland oil industry[9]. This provided a plausible alternative explanation of the Thurso (Caithness) leukaemia cluster previously attributed to Dounreay. Similarly Kinlen's examination of large greenfield industrial, construction sites demonstrated the striking similarity of the pattern of excess of childhood leukaemia for these; including Sellafield when considered as a greenfield site[10]. It should perhaps be explained that Sellafield, in many ways, has been not so much a greenfield site but a series of greenfield sites with a rapidly turning over, serially migrant but constantly present large contractor population throughout many years of its existence.

DISCUSSION

In retrospect the leukaemia/nuclear installation debates have been characterised by early, small positive studies with a rapid fall off in positivity in later studies, large and small.

The phenomenology of the leukaemia/nuclear installation studies thus shows a pattern of secular regression to the mean similar to that seen in a number of health scares, as exemplified by the VDT - miscarriage data. This latter is shown in Fig. 3. In contrast this pattern is not seen in studies testing the population mixing concept (Fig. 2).

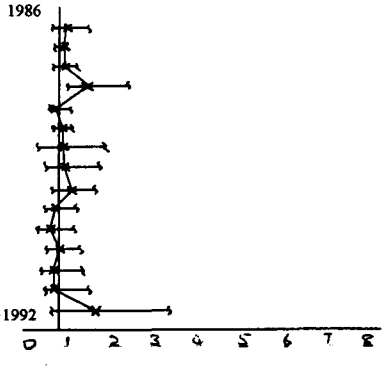

Fig. 3 Work with VDTs and spontaneous abortions, summary of studies. RR and CI for highest exposed group in each study.

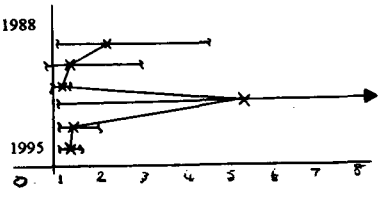

Fig 2. Tests of the childhood leukaemia population mixing hypothesis, summary of studies. O/E or RR and CI for highest "at risk" population group in each study.

The inference that may be available from the foregoing is that the population-mixing hypothesis may be describing events more plausibly than the hypotheses linked to radiation. The recognition of the phenomenology, that is the temporal trend or history of the studies relevant to a particular theory is of interest in itself but is more importantly a prescription for redoubled caution in progressing scientific research and delineating public policy in relation to it.

In scientific terms it is first necessary to re-emphasise the primarily hypothetical nature of case referent studies. Additionally, it is important to have higher regard to the long established criteria for inferring causation as propounded, among others, by Bradford-Hill. Thus it is prudent to show proper scientific caution by exercising due regard for the null hypothesis and similarly to view with caution the dissonances discernible in the biological and general scientific plausibility of novel ideas (especially across different scientific disciplines). Further it may be necessary to form a detached view of how much science should actually be done and completed before speculation is reasonably regarded as being ended and replaced by some degree of certainty. Finally, and this passes over into public policy also, the science that is done might usefully be planned so that it may be optimally focused and reorganised to serve the public interest.

In public health scares, initial "official" caution about a new hypothesis is discernible until a threshold of concern, often modulated by media attention, is overcome. A second characteristic is the subsequent caution manifested before the previously alien hypothesis, then adopted, is discarded and abandoned. It is suggested that such features, being generic, are capable of generic solution perhaps by effectively separating expert scientific advice from policy making based on that advice. Also, although this may be a counsel of perfection, some attempt might be made to avoid stigmatisation of a putative "guilty" party and putative populations at risk during the investigative phase.

By such more rigorous scientific criteria the population-mixing hypothesis performs quite well but still needs to meet a number of "robustness" requirements. Thus it is of importance to its viability that biological agents of effect be identified as well as the process by which the effect might be manifested. Further tests would usefully include the capacity to explain the Seascale cluster, demonstrate some form of dose response relationship and attempt to explain the negative findings reported round many nuclear installations.

CONCLUSIONS

The main thesis of this paper has been to demonstrate the particular phenomenology of the nuclear installations/childhood leukaemia sagas and to infer the scientific and public policy implications of the experience of investigation of these hypotheses and their confounders. If, as appears to be the case, we live in a period where science policy is more populist and thus liable to be impacted more directly and immediately by "public concern" issues, it follows that there may also be new challenges for public information and understanding and new requirements for the operation of the process of public science policy.

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ARE THERE RISKS FROM EXPOSURE TO LOW LEVEL RADIATION?

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ABSTRACT

For doses up to at least 50 mSv received at a high rate in a single event by adults (or 10 mSv for exposure of the embryo and foetus) and up to at least 15 mSv per year received routinely, there is no proof that radiation increases risks of cancer. One reason for this may be that there are beneficial or protective effects of radiation which tend to offset and perhaps dominate over the harmful effects at low levels of exposure. Another reason is that the effects are too small to be measured. It is not necessarily in the best interests of individuals or of society to assume that the linear hypothesis should be used to estimate risks throughout this range.

INTRODUCTION

With the resumption of nuclear weapons testing by France, there has been much in the news and "on the streets" in Australia about the dangers of exposure to radiation. We continually hear and read statements to the effect that there is no safe level of exposure and, regrettably, our profession has to bear some of the responsibility for this misconception.

Elsewhere in the world, a great deal of dissent and concern is being expressed regarding radiation protection regulations and practices which are based on the view that no level of radiation dose is free of risk. The first sixteen pages of the Health Physics Society's Newsletter of June 1995 provides examples. Particular concerns have been expressed that such regulations impose high costs on industry without achieving demonstrable benefits. It has even been suggested that some providers of radiation protection services have a vested interest in this situation.

Central to both issues is the ICRP's recommendation that risks of radiation induced cancers should be assumed to be proportional to dose without thresholds (1), the "linear hypothesis". It needs to be stressed that the linear hypothesis is an assumption, not a proven fact or a law of nature. Its application to low levels of radiation needs to be placed in perspective. This should not be taken as support for nuclear weapons testing or as opposition to the assessment and regulation of radiation protection practices. It simply reflects the need to apply a scientific approach to the question of risk at low levels of radiation exposure.

RISK ESTIMATION IN PERSPECTIVE

The linear hypothesis was formulated for the purpose of assessing radiation protection practices, not for estimating risks to individuals from low levels of radiation. The risk coefficients recommended by the ICRP (1) are based on the observed health effects of high doses received at high dose rates, e.g. by atomic bomb survivors at Hiroshima and Nagasaki. Radiation is more likely to be harmful when delivered at these high rates rather than spread over a protracted period. However, even at the highest rates which have been experienced, statistically significant increases in the incidence of cancers have not been observed at dose levels less than about 50-200 mSv for adults (1-2). For exposure of the embryo and foetus, detectable health effects have been observed at dose levels down to about 10-20 mSv.

The extent to which the linear hypothesis and the recommended risk coefficients apply

at lower doses and dose rates is questionable. As explained below, this is not so much a matter of thresholds as of the relative importance of different effects. However, the ICRP does not rule out the possibility that thresholds may exist (1).

It appears to be well established that radiation has beneficial or protective effects on living cells and organisms, as well as harmful or potentially harmful effects (2,3). Determination of the net effects on humans (such as changes to life expectancies or risks of cancer) requires properly designed epidemiological studies. At low levels of exposure, net effects may be too small to discern, i.e. because of statistical difficulties of measuring them against variations not caused by radiation. Hence, although biologically positive effects (sometimes called "hormesis") have been demonstrated in the form of adaptive responses of cells and organisms to damage from radiation, these effects cannot be reliably expressed as reductions in the incidence of cancer (2). On the other hand, there has been no detectible excess of cancers attributable to radiation for the same conditions of exposure.

PROTRACTED EXPOSURE TO RADIATION

A recent study (4) of protracted occupational exposures in the nuclear industry has shown that there is "no evidence of an association between radiation dose and mortality from all causes or from all cancers". At 100 mSv, there are dose-related increases in mortality from some cancers but these appear to be balanced by dose-related decreases for some other types of cancer.

The average rate of protracted public exposure to background radiation is about 2 mSv per year in Australia. In other parts of the world, it varies from less than 1 mSv per year to more than 15 mSv per year (2). Local populations living above some mineral deposits incur lifetime doses well in excess of 1,000 mSv from natural background radiation. The ICRP risk coefficient of 5×10^{-5} per mSv applied to a dose rate of 2 mSv per year gives a risk rate of 10 fatalities per hundred thousand per year, which is about 5% of the total rate of fatal cancers from all causes in Australia.

There are many variable factors other than radiation which affect the incidences of cancers. Nevertheless, on the basis of a linear extrapolation from figures in the foregoing paragraph, it would be expected that differences of dose rate over its range of natural variation would cause substantial differences in the total rates of cancers in the exposed populations. In fact, no correlation has been established between background radiation and rates of cancer (or genetic effects) in humans, suggesting that the ICRP recommendations do not apply to routine exposures within the range of background radiation rates. Effects, if they exist, are so small that they cannot be measured.

Even if radiation in the natural environment is having effects, it might be inappropriate to describe these effects as harmful. Human evolution has taken place in the presence of naturally occurring radiation and it is a fundamental tenet of evolutionary biology that organisms adapt to their environment (5). This means that, within the range of natural background, levels of survival and fitness should be expected which are optimum with respect to radiation.

APPLICATIONS OF RISK ESTIMATES IN RADIATION PROTECTION

The practice of radiological protection is concerned mainly with two of its three general principles - compliance with dose limits and the optimisation of radiation protection practices. Compliance with these principles can reasonably be equated with "safety", because the total absence of risk is fundamentally not achievable in the practice of any potentially hazardous activity and therefore cannot be a legitimate objective of regulation.

The linear hypothesis is the basis for the establishment of individual dose limits and is

intentionally conservative for this purpose. Dose limits recommended by the ICRP (1) are actually at levels where the risk can only be estimated hypothetically. These dose limits provide a high level of protection for individuals whose exposures are reliably known.

However, the postulation of the linear hypothesis has given rise to the belief that any dose of radiation, no matter how small, increases the risks of cancer and other adverse health effects. In fact, there is no evidence to support this belief except for the linear hypothesis itself, which is an assumption, and radiation appears to have biologically positive (beneficial) effects which may predominate at low dose levels. When dealing with the effects of low doses, estimated in accordance with the linear hypothesis, optimisation may thus involve balancing real costs against hypothetical benefits which are too small to be demonstrated and might well be non-existent. This application of the linear hypothesis should be viewed with caution.

Applications of the linear hypothesis to justification, the third general principle of radiological protection, should also be viewed with caution. The justification of a practice requires that it produce sufficient benefit to offset the radiation detriment it causes (1). This goes far beyond the scope of radiological protection. Clearly, however, many of society's concerns about the justification for uses of nuclear energy relate to risks, and hence to detriments, which have been overestimated - perhaps greatly overestimated. Decisions made on this basis may not be in society's best interests.

CONCLUSIONS

For doses up to at least 50 mSv received at a high rate in a single event by adults (or 10 mSv for exposure of the embryo and foetus) and up to at least 15 mSv per year received routinely, there is no proof that exposure to radiation increases the risk of cancer. Biologically positive and negative effects of radiation have been demonstrated in this range. However, because of statistical difficulties of measurement, they cannot be evaluated quantitatively in terms of changes to rates of cancer incidences. It is not likely that there is a discontinuity or cut-off to any of these effects at either high or low levels of exposure. The assumption that effects observed only at high doses and dose rates can be extrapolated to low doses and dose rates, to the exclusion of other effects, is questionable. It is not necessarily in the best interests of individuals or of society to make this assumption.

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1996 International Congress on Radiation Protection
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PAPER TITLE
POSTCHERNOBYL DOSE AND RISK ASSESSMENTS IN UKRAINE

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MAJOR SCIENTIFIC TOPIC NUMBER 2.3, 3.1-3.5 (page 7)

The experience and results in dose and risk assessment for Ukraine population after the Chernobyl accident are generalized. Four affected groups are considered: (i) *clean-up worker* (liquidators), (ii) *population* of the contaminated territories, (iii) *people had been evacuated* from Prip'jt and 30-km zone, (iv) *children with exposed thyroid gland.* Main results are follow :
(i) The exposure of *liquidators* (130 thousands persons worked in Chernobyl NPP in 1986) were estimated as 0.02-6 Sv; 6-15% of them received the doses higher than 0.25 Sv. Average individual dose of liquidators (in 1986) is 0.12 Sv.
(ii) Average ten-years individual doses of *general population* from contaminated territories are 10-160 mSv. Levels and relations of internal and external exposure pathways strongly depend on the initial radioactive deposition, type of soil, agriculture practice and diet traditions.
(iii) Doses of *evacuated people* (before the evacuation) are 1-50 mSv.
(iv) *Radioiodine thyroid exposure of children* had been reconstructed based on the some thousands of direct thyroid activity measurements made in 1986. The interval of estimated dose ranges from less than 10 mSv to more than 10 Sv. The map with the different children thyroid dose zones within Ukraine is demonstrated.

The calculations of the late health effects induced by different exposure sources for different age-groups were made. These estimations are compared with the really observed levels of oncomorbidity and mortality. The only statistically significant radioinduced effect is thyroid cancer among children exposed in 1986. All the results are presented in form of tables and maps.

DOES THE EXPOSURE FROM THE CHERNOBYL ACCIDENT ASSOCIATE WITH CANCER DEATHS IN GREECE ?

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INTRODUCTION

Exposure of the population occurs via three main pathways: external irradiation from material deposited on the ground, inhalation of airborne material and ingestion of contaminated foodstuffs. The population dose associated with these exposure pathways was evaluated just for one year, i.e. the first year after the Chernobyl accident (May 1986 - April 1987) for the following reasons: i) the specific activities of I-131, Ru-103 and Cs-134 + Cs-137 in air were peaked on May 5-6,1986. A month later, the specific activities of the above radionuclides in air declined by a factor of 1000 (1) reaching the level of 1 mBq /m3 or lower, ii) the specific activity of the long-lived Cs-137, which remained until today, significantly decreased in the foodstuffs a year after the Chernobyl accident, i.i. by a factor of 1000.

EXPERIMENTAL PROCEDURES

From the beginning of May 1986, the Nuclear Physics Laboratory of Aristotle University of Thessaloniki (Greece), initiated different measurements on fallout of the radioactive debris transported from Chernobyl.

Alpha, beta and gamma radiation monitoring systems were set into operation in order to measure the radioactivity in air, precipitation (rainwater from any rainfall event), soil, grass, milk and a large number of items of contaminated foodstuff.

All the samples were measured for radioactivity using a Ge-Li high resolution spectrometer (resolution 2 keV at 1.33 MeV) in a standard geometry 40 g plastic can of 6 cm in diameter or in a Marinelli beaker of 1 litre (volume) or about 1 kg (mass). The gamma spectroscopic system was calibrated using standard reference sources in both cases, the overall efficiency being known to accracy of better than 5 % for the plastic can geometry and about 12 % for the Marinelli beaker geometry.

EXPOSURE OF THE POPULATION

Taking into account the concentrations of the following radionuclides: I-131, Ru-103 and Cs-134 + Cs-137 in all the samples of air, precipitation, soil, grass, milk and various items of contaminated foodstuff as well as the dose conversion factors, it was estimated that 17.7 μSv was due to external irradiation from contaminated ground, 98.14 μSv was due to internal dose from inhalation of airborne material and 2.04 mSv was due to internal dose from ingestion of contaminated foodstuffs. The total dose enhancement for the first year after the Chernobyl accident was 2.16 mSv. It is concluded that 94.4 % of the total dose enhancement was due to internal dose from ingestion of contaminated foodstuffs.

RADIOLOGICAL IMPACT

Taking into account the risk factor, wich is of the order of 10^{-5} per mSv for deaths by cancer (2), the total dose enhancement, i.e. 2.16 mSv (216 mrem) would be expected to lead to 214 cases of cancer deaths appearing in Greece over a period of 15 years or more in a population of 10 million people. This predicted increase in total cancer deaths would be difficult to detect, as it is very small relative to the number of cancer deaths from all other reasons (except Chernobyl) which should be about 330000 up to the year 2001 as derived from the data of Fig.1. It is therefore not achievable to distin-

guish any cancer death due to the Chernobyl accident from cancer deaths due to all other reasons.

The collective dose is the individual dose multiplied by the population. Based on our data, an approximate estimate gives about 20000 person Sv for the annual collective dose, assuming ≈2 mSv as the annual individual dose and 10 millions the population of Greece in accord with the last census of 1991 (3).

As a consequence of the exposure to radioactivity due to fallout from the Chernobyl accident cancer deaths are expected at the digestive system, the respiratory system and the hematopoietic system. No clear rise was observed in the monthly variation of cancer deaths in all of the above mentioned cases since the Chernobyl accident, see Fig.2.

In Fig.1 the annual number of cancer deaths in Greece was presented for the period 1968-93. The increase of cancer deaths from the year 1992 to the year 1993, just one year, was 2.5 %. That percentage was the maximum since the Chernobyl accident. The data of Figures 1 and 2 originated from the Monthly Statistical Bulletin of the National Statistical Service of Greece.

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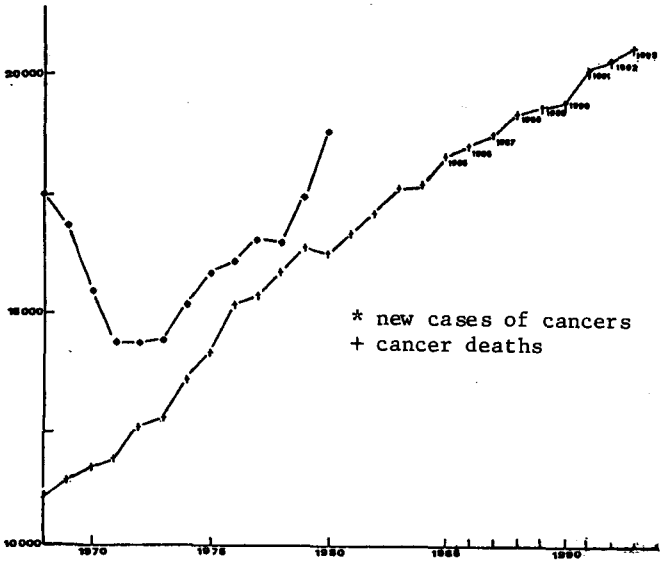

Figure 1. Annual number of cancer deaths in Greece (1968-93).

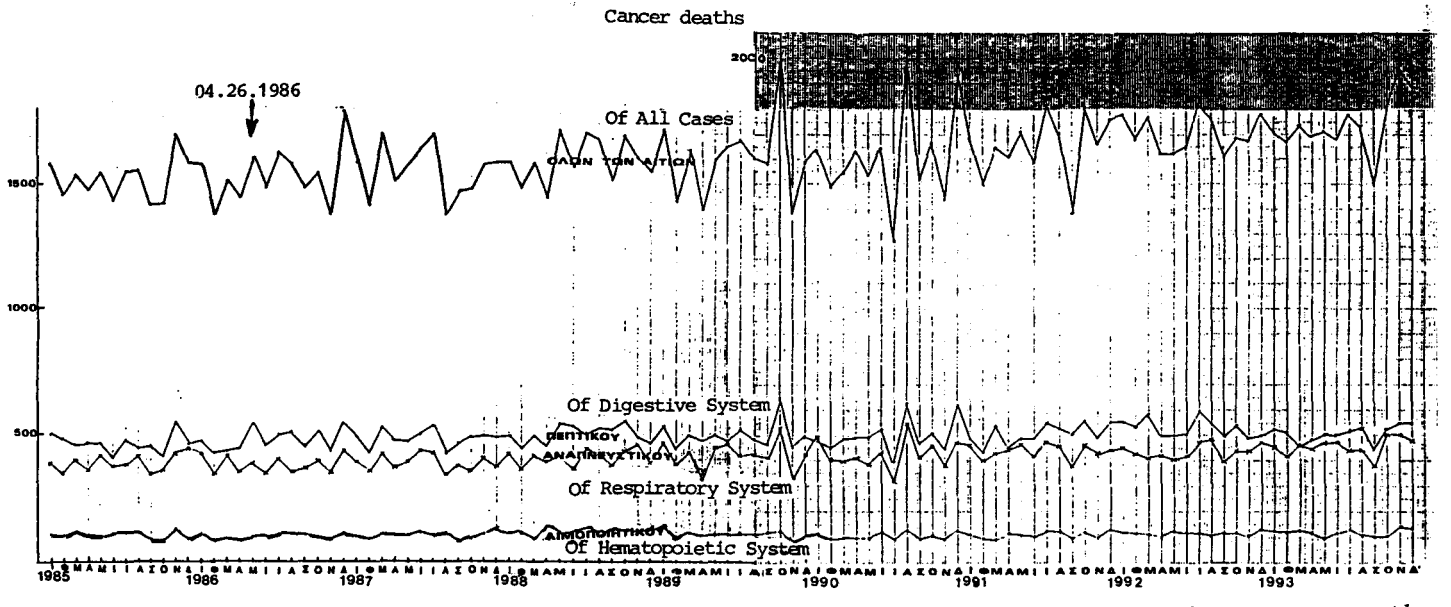

Figure 2. Monthly variation of cancer deaths in Greece for the digestive system, the respiratory system, the hematopoietic system, cancer deaths of all reasons (1985-93).

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PAPER TITLE ESTIMATION OF CARCINOGENIC RISKS FROM NEUTRONS IN MAN

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MAJOR SCIENTIFIC TOPIC NUMBER 2.2. (see page 7)

ABSTRACT (See instructions overleaf)

The comparison of the effects of high and low LET radiation exposure in man has been the subject of much debate over the years. Although the cohort of Japanese atomic bomb survivors has for a long time been the main source of data for the direct estimation of the effects of γ-rays, since the introduction of the DS86 dosimetry it has not been possible to draw any conclusions about the risks from neutrons. However, there is now growing evidence that the DS86 dosimetry needs further revision, which could result in an increase in neutron doses by up to a factor of 10 at distances beyond 1000 m in Hiroshima (Straume, et al., 1992, Health Physics, 63, 421-426, Straume, et al., 1994, Radiation Research, 138, 193-200). If the neutron doses were revised upwards in the manner suggested, the excess relative risk (ERR) for cancer incidence from low-LET radiations is likely to remain relatively unchanged. However, such a revision should make it possible to estimate the risks from neutrons. Using the recently released Japanese incidence data and assuming the dose responses are linear for both neutron and gamma radiation, it is estimated that the ERR per gray of neutrons will most likely be between 2.5 and 8.5 for all cancers except leukaemia. These values correspond to a relative biological effectiveness (RBE) of neutrons compared to gamma radiation of 5 and 15 respectively. These values are in broad agreement with the current ICRP neutron quality factors and radiation weighting factors for the energy range 0.1 keV - 2.0 MeV relevant to the Japanese bomb survivors (ICRP Publication 60, 1991).

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PAPER TITLE THE THEORY OF RELIABILITY AS A BASIS FOR RISK ESTIMATES OF THE HEALTH EFFECTS ASSOCIATED WITH RADON AND ITS PROGENY

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MAJOR SCIENTIFIC TOPIC NUMBER 1.1 (see page 7)

ABSTRACT

The contribution of radon and its short-lived decay products amounts to as much as 80% of the indoor background radiation in Russia and other republics of the FSU. No less than 10% of the lung-cancer risk in human populations may be ascribed to the radon radiation. The lung-cancer annual mortality-rate risk for adults follows the double exponential distribution function. To explain this law, the reliability-theory approach has been developed [Grodzinsky et al., 1987; Koltover, 1995]. This approach enables explanation of the kinetics of the mortality growth with age together with evaluation of effects of exposure to radon progeny. Apart from the quantitative constructing of mortality curves, the theory of reliability may also serve as the basis of investigation of molecular mechanisms of radon effects. In part, the hormetic effects of low-dose exposure to radon can be explained as resulting from its mild stress-action upon the neuro-endocrine system that trains the organism's adaptive possibilities. The analysis of fluctuations of physiological parameters is suggested as the prognostic method for testing the reliability of bio-systems while exposed to radon and its decay products.

AN EXPERIMENTAL MODEL FOR RISK ASSESSMENT OF COMBINED EXPOSURE TO RADON AND OTHER AIRBORNE POLLUTANTS

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INTRODUCTION

Combined exposure to radon and its progeny and various occupational or environmental airborne pollutants may lead to synergistic effects for lung cancer induction. In humans, an increased incidence of pulmonary neoplasia has been observed in different groups exposed to radon and its daughters, including uranium miners (1), iron miners (2) and other miners (3) especially cigarette smokers (4). These findings suggest that co-carcinogenic mechanisms may be involved in the pathogenesis of lung cancer. In laboratory animals, a co-carcinogenic effect results in increased tumour rates after combined administration of the potential carcinogens (5). A standardized protocol has been developed in Sprague-Dawley rats to identify potential co-carcinogenic agents. In this, rats are exposed to 1000 WLM of radon followed by exposure to the agent to be studied (6). Exposure to 1000 WLM of radon alone results in a lung cancer incidence of 20%. About 30% of the tumours are squamous cell carcinoma, 50% adenocarcinoma and 20% bronchioloalveolar carcinoma. In these lifespan studies, after exposure, rats were allowed to live until they died or were moribund and then killed and the latency period for tumour induction was about 700 days. The effects in rats of combined exposure to radon and various environmental or industrial airborne pollutants such as cigarette smoke, mineral fibres, diesel exhausts, minerals associated with metallic mine ores and chlorinated compounds are reviewed in relation with the possibility of combined exposure for workers in different industries.

RESULTS

Combined exposure to radon and tobacco smoke

The first experiments were carried out to investigate the effects of inhalation of radon and its daughters at various cumulative doses, before or after various passive exposures to tobacco smoke (7), using cigarettes with and without filters. For a 1000 WLM radon exposure, the incidence of lung carcinomas was slightly lower in rats exposed to tobacco smoke before radon exposure than in rats exposed to radon alone, but the distribution of the different histological types of tumours were similar in the two groups. In contrast, a highly significant excess of lung carcinomas, mainly of the squamous cell type, was observed in the group exposed to tobacco smoke after radon exposure. In this group, the incidence of lung carcinomas was almost four times greater than in the group exposed to radon alone.

The results of further studies in which rats were exposed to cigarette smoke following exposure to radon showed that for the same tobacco smoke exposure, the incidence of lung carcinomas increased with the cumulative dose of radon. In the same way, for an identical cumulative dose of radon and its daughters, the incidence of lung carcinomas increased with the cumulative exposure to tobacco smoke.

The induction of lung carcinomas was less efficient in rats exposed to tobacco smoke produced by filter cigarettes than in those exposed to cigarettes without filters. The incidence of lung carcinomas were higher, but not statistically significant in the groups exposed to radon and tobacco smoke combined than in the group exposed to radon alone and the proportion of lung carcinomas was lower in the group exposed to filter cigarettes than in the group exposed to unfiltered cigarettes. The increased incidence of lung carcinomas in the group exposed to radon and non filter cigarettes was mainly related to an increased incidence of squamous cell carcinomas. These findings suggested a stronger synergistic effect of radon and non filter cigarettes compared to that of radon and filter cigarettes.

In rats exposed to radon and tobacco smoke combined, for the same radon exposure, the incidence of lung carcinomas was greatly increased in the group exposed to radon and tobacco smoke compared with the group exposed to radon only. Tumours observed in the

groups exposed to radon and tobacco smoke were larger and more invasive than in the groups exposed to radon alone. These tumours also spread more to the pleura and the presence of intrapulmonary metastases or of multiple tumours in the lung was observed. For the same radon exposure, the mean latent period for lung carcinomas was shorter in the group exposed to radon and then to tobacco smoke compared with the group exposed to radon alone. For an identical tobacco smoke exposure of 350 hours, the mean latency period was inversely related to the cumulative radon dose. All these results showed a clear co-carcinogenic effect of exposure to radon and radon daughters and tobacco smoke in rats.

Combined exposure to radon and mineral fibres and other industrial or environmental airborne pollutants

The experimental protocol described above was used to study the potential co-carcinogenic effects of radon and mineral fibres. Acid leached chrysotile fibres were shown to exhibit less carcinogenic activity *in vivo* than untreated fibres (8). Since mesothelial cells are considered to be the target cells for the induction of tumours by mineral fibres, this experiment was designed to investigate the potential synergistic action of different kinds of unleached or acid leached asbestos fibres and other mineral dusts injected into the pleural cavity of rats after previous inhalation of radon and its daughters. In these experiments, 60 rats exposed to radon were used as controls. Ten groups of 10 rats each were exposed to the same dose of radon and then, 2 weeks later were injected intrapleurally with 2 mg of mineral dust, unleached or leached asbestos fibres, glass fibres and two varieties of quartz. No rats were exposed to mineral fibres alone. The results are summarized in table 1.

Table 1. Combined effects of radon and other environmental or industrial airborne pollutants

Experiments	Results	Conclusion
Radon + Cigarette Smoke	Increased incidence of lung carcinomas up to 4 times greater than in rats exposed to radon alone	**Multiplicative effect**
Radon + Mineral Fibres	1 / 3 Lung carcinomas 1 / 3 Pleural mesotheliomas 1 / 3 Combined pulmonary pleural tumours in excess	**Additive effect**
Radon + Diesel Exhaust	Slight non significant excess of lung carcinomas	**No clear synergistic effect**
Radon + Minerals from Metallic Mine Ores	Slight non significant excess of lung carcinomas	**No clear synergistic effect**
Radon + Methylene chloride	No excess of lung cancer	**No synergistic effect**

The potential carcinogenic or co-carcinogenic role of 4 minerals present in the ores of metallic mines was also investigated. These included, nemalite (a contaminant of Quebec chrysotile), biotite (present in many granites and in the French uranium ore), iron pyrites (present in various iron and gold ores), and finally iron-rich chlorite (present in iron, tungsten and gold ores).

The use of diesel-powered vehicles is steadily increasing worldwide. Among diesel-exhaust exposed populations, only a case control study and a retrospective cohort study in railroad workers (9), showed a significant association between diesel exhaust inhalation and lung cancer, suggesting that occupational exposure to diesel exhausts results in a small but significant excess risk of lung cancer. Experimentally, some evidence of a carcinogenic effect has been previously reported in rats after exposure to diesel exhaust containing high concentrations of diesel soot particles for periods of up to 2 years (10). The potential synergistic effects of diesel exhaust were investigated in rats after previous exposure to radon and radon daughters.

Chlorinated compounds are widely used in a variety of commercial forms for industrial and medical applications. Exposure to methylene chloride induces lung and liver cancers in mice. The aim of this study was to test the potential carcinogenic or cocarcinogenic effect of methylene chloride in rats, acting either alone, as a complete carcinogen, or as a promoter after local pulmonary irradiation by inhalation of radon.

The results of these different studies are summarized in table 1.

CONCLUSION

These results demonstrate the potential co-carcinogenic action of various environmental or industrial airborne pollutants combined with radon exposure, showing either a multiplicative, an additive or nul effect. The strongest co-carcinogenic effect was shown by combined exposure first to radon and then to tobacco smoke and resulted in an increased incidence of lung carcinomas, mainly of the squamous cell type. These results also indicated the possible application of this "radon model" to the investigation of possible interactions between exposure to two occupational and/or environmental pollutants. The importance of such an experimental model for risk assessment should be emphasized since human industrial occupational or environmental exposures are nearly always not single but multiple.

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A MOLECULARBIOLOGICAL MODEL FOR THE DOSE-EFFECT RELATION FITTING DIFFERENT EPIDEMIOLOGICAL DATA

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The effect of radiation hormesis can be observed in an increasing number of epidemiological studies investigating the cancer rate at different levels of background radiation. At this low level dose rate the mortality rate for several types of cancers decreases with increasing dose rate while others do not. With the Random Coincidence Model (RCM), a new model concerning the dose-effect relation, /1/, /2/, it is possible to explain the radiobiological effect of radiation hormesis and to give conditions when hormesis takes place and when it does not. This model describes the formation of cancer caused by a multistage process of mutations in the critical regions of tumor associated genes such as proto-oncogenes or tumor suppressor genes. The central thesis of the model says that a point mutation not only occurs by a causal coincidence of two base lesions or a misrepair of one base lesion or a DSB but also by a random coincidence of two base lesions or two single strand breaks of complementary DNA strands during the repair time of the first base lesion or single strand break.

As the random coincidence predominates the results of the RCM, the basic equations of this mechanism will be discussed now. The probability for a single base lesion or a single strand break per second (S) on each location of the DNA is the sum of the probabilities of a chemically induced nucleotide lesion ($C\alpha$) and a radiologically induced nucleotide lesion ($D\beta$)

$$S = (C\alpha + \dot{D}\beta) \tag{1}$$

The probability per second that the complementary nucleotide (or a nucleotide close to the complementary nucleotide) is also damaged before or during the repair of the first nucleotide, making the repair ineffective is described by the equation:

$$S_{fixed} = (C\alpha + \dot{D}\beta)^2 \tau \tag{2}$$

This is the fixation rate of point mutations or a misrepair of DSB caused by a randomly coincident destruction of both complementary bases or a randomly coincident caused double strand break. The coincidence takes place during the repair time. Equation (2) leads to a synergism between radiation and chemical noxes as a mixed term that does not arise if we just add the terms of both damage mechanisms:

$$S_{fixed} = \left[C^2\alpha^2 + 2C\alpha\dot{D}\beta + \dot{D}^2\beta^2\right]\tau \tag{3}$$

It is well known that cells have a response mechanism which reacts on irradiation. Works from Eckardt-Schupp /3/, Boothman /4/, Woloschak /5/, Sankaranarayanan /6/ and many others have shown this in detail. The elimination of low-dose hypersensitivity in Chinese Hamster V79-379A cells by pretreatment with x rays or hydrogen peroxide was recently shown in a work from B. Marples and M.C. Joiner /7/.

The RCM model implies that the rate of damage fixation depends on the repair time of the first base lesion or single strand break. The repair time is a function of the repair enzyme concentration as indicated by the Michaelis-Menten relation /8/. Recently Livneh /9/ gave the experimental evidence of the mutagenic pathway described above.

In the lowest dose rate region we assume a linear correlation between the mean dose rate $\dot{D}$ applied to tissue and the enzyme concentration increasing after this irradiation:

$$[E] = [E_0] + \delta' \dot{D} \tag{4}$$

$[E_0]$ is the concentration of repair enzymes in the absence of radiation and [E] is the concentration after a stimulation caused by radiation /3-7/. The factor δ' describes the extent of the production of additional repair enzymes caused by radiation and so does δ which is defined as $\delta := \frac{\delta'}{[E_0]}$. This leads to

$$\frac{[E]}{[E_0]} = 1 + \delta \dot{D} \tag{5}$$

For low dose rate irradiations as regarded in the data fits of B.L. Cohen's radon data and in those fits related to background radiation (ARIP), we refer to the equilibrium of enzyme production: the production of repair enzymes by chemicals and radiation is as large as their decay. The search for lesions by means of repair enzymes is a diffusion controlled process. In other words: the more repair enzymes are available, the faster is the finding of lesions and the higher is the velocity v of the repair process: v is proportional to the enzyme concentration and it is indirectly proportional to the repair time, thus we obtain

$$\tau = \frac{\tau_0}{1 + \delta \dot{D}} \tag{6}$$

Combining equations (3) and (6) we get equation (7), for which each term is plotted separately in Fig.1.

$$S_{fixed} = \underbrace{\frac{C^2 \alpha^2 \tau_0}{1 + \delta \dot{D}}}_{\text{term 1}} + \underbrace{\frac{2C\alpha \dot{D} \beta \tau_0}{1 + \delta \dot{D}}}_{\text{term 2}} + \underbrace{\frac{\dot{D}^2 \beta^2 \tau_0}{1 + \delta \dot{D}}}_{\text{term 3}} \tag{7}$$

Fig.1 shows the most important influence of the Random Coincidence Model on the discussion of the linear dose-effect relationship versus radiation hormesis at small dose rates. Fig. 1 clearly shows that both parties are right:

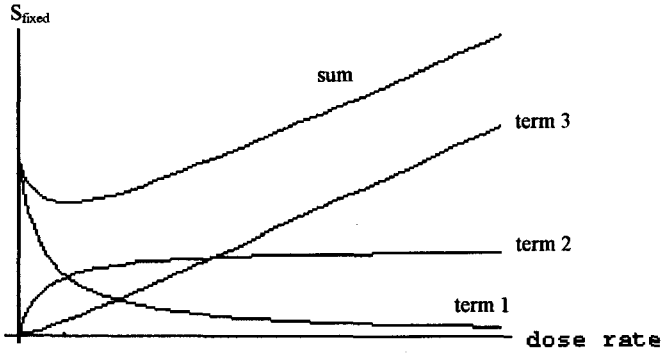

Fig.1

1) The direct radiation induced transformation rate (term 3) increases as a function of dose rate very close to linearity.

2) In addition to that, the radiation stimulated production of repair enzymes is able to reduce oxidative impact of radicals of the natural cell metabolism (term 1) and therefore causes a decrease of the natural transformation rate.

3) Of course there is a synergism of chemical and radiation impact (term 2), which indicates that under certain circumstances a multiplicative dose-risc relationship will dominate the dose-effect relationship.

4) Only if the impact of chemicals or the natural cell metabolism on the DNA is larger than the radiation induced one, the reduction of the oxidatively induced lesions is able to dominate over the increase of radiologically caused lesions: hormesis takes place (sum).

In other words: radiation hormesis can only exist if the oxidatively induced transformation rate is larger than the radiation induced one.

For experiments on radiation adaption the working principle is the same as in hormesis, just the part of the dominating oxidative and chemical impact is taken over by the "challenge dose": the increase of the repair stimulating damages due to the "inductive dose" is much more than compensated by the large decrease of those damages caused by the "challenge dose" and this decrease is a consequence of the stimulation of the repair system by the priming dose.

The Random Coincidence Model describes two different ways of the formation of point mutations:

a) point mutations where the damage of the complementary base (nucleotide) occurs before or during the repair of the first base (we talk about a random coincident fixation of the damage because the damage of the complementary base happens randomly and coincidently within the repair time τ before the first base is repaired)

b) point mutations where the damage of the complementary base (nucleotide) occurs immediately after the damage of the first base (we talk about a causal fixation). Both complementary bases (nucleotides) are damaged by the same particle or by the same cloud of radicals resulting from one particle.

For those point mutations described under a) the formalism of the model contains an expression for the chemically induced rate of point mutations ($C^2\alpha^2\tau$), one for the synergism between radiologically induced and chemically induced rate of point mutations ($2C\alpha\dot{D}\beta\tau$) and one for the coincidently occuring radiation induced rate $\dot{D}^2\beta^2\tau$ (equ.3). For point mutations described under b) we use the expression for causal damages ($\frac{1}{f_{nuc}} z_F \dot{D}\beta^2$), changing equation (3) into

$$S_{fixed} = \frac{\left[C\alpha + \dot{D}\beta\right]^2 \tau_0}{1 + \delta\dot{D}} + \frac{1}{f_{nuc}} z_F \beta^2 \dot{D} \qquad (8)$$

where the relation between the term for causal damages and the term for random coincidences can be derived by the help of microdosimetry, where z is the frequency mean specific energy per event departed by the radiation in a critical volume.

There is much epidemiological and experimental evidence that malignant change is a process which results from multiple genetic alterations. Several lines of evidence suggest that changes to more than one proto-oncogene are involved in tumors /10/. In a few human tumors, for example, changes to more than one proto-oncogene have been found /11/. Another evidence for the involvement of multiple oncogenes comes from studies in which the malignant transformation of non-immortal cell strains is facilitated by the introduction of two different oncogenes /12/. Therefore equation (8) is the source term in a system of linear differential equations, where each of the equations describes one transformation step in the successive process of carcinogenesis.

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The explanation of B.L. Cohen's radon data and the low-LET ARIP-data with the Random Coincidence Model

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Abstract The Random Coincidence Model (RCM) describes the formation of cancer if caused by a multistep series of point mutations in the critical regions of tumor associated genes such as proto-oncogenes or tumor-suppressor genes. It is the central thesis of the model that a point mutation mainly occurs through the random coincidence of two base lesions or two single strand breaks of complementary DNA bases (strands) during the repair time of the first base lesion or SSB.

Introduction The effect of radiation hormesis can be observed in an increasing number of epidemiological studies investigating the cancer rate at different levels of background radiation. At this low level dose rate the mortality rate for several types of cancers decreases with an increasing dose rate while others do not. In his epidemiological investigations, B.L. Cohen (1) measured the average radon level in homes in 1730 counties of the USA and correlated these data with lung cancer mortality. He found a clear tendency of decreasing lung cancer mortality with an increasing radon dose. Years ago N.A. Frigerio (2) got the same results for background radiation (terrestrial- and cosmic γ-radiation, K-40).

The RCM proposes the randomly coincident destruction of the complementary DNA base (nucleotide) before or during the repair of the first damaged base (nucleotide). Random coincidence means that after the first base (nucleotide) lesion the second damage on the complimentary base (nucleotide) occurs randomly before the repair of the first base (nucleotide) lesion is finished and coincidently within the repair time. This approach implies that the rate of damage fixation depends on the repair time of the first base lesion or SSB. The repair time is a function of the repair enzyme concentration as indicated by the Michaelis-Menten relation. To explain radiation hormesis we have to assume that the repair enzyme concentration and the concentration of scavengers, radical detoxification systems, is direct proportional to the dose rate.
Wallace has recently reviewed the nature of the DNA lesions caused by active oxidizing species produced both naturally and by low-LET radiation (3). Oxidizing radicals and especially OH radicals resulting from either cause produce similar types of DNA lesions (3-5). The enzymes involved in their repair are similar no matter whether the DNA damage is produced spontaneously or by radiation (6).
We therefore explain radiation hormesis as a result of the reduction of the spontaneous (chemical) damage probability by a small dose rate of ionizing radiation causing an additional genetic expression of repair enzymes and scavengers.

The Model $C\alpha$ is the chemical damage probability per second and nucleotide, caused by the natural cell metabolism and by chemical carcinogens. $\dot{D}\beta$ is the radiological damage probability per second and nucleotide. $\dot{D}\beta$ includes the possibility that radiation directly interacts with the nucleotide or indirectly by radicals.
The probability per second that the complementary base (nucleotide) is also damaged before or during the repair of the first base (nucleotide) is described by the equation:

$$S_{fixed}=(C\alpha + \dot{D}\beta)^2 \tau \qquad (1)$$

This is the fixation rate of point mutations caused by a randomly coincident destruction of both complementary bases or a randomly coincident caused double strand break. The coincidence takes place during the repair time.
v is proportional to the enzyme concentration and it is indirectly proportional to the repair time τ:

$$\tau = \frac{\tau_0}{1+\delta\dot{D}} \qquad (2)$$

with $\delta \cdot \dot{D}$ describing the additional genetical expression of repair enzymes.
Besides the three terms in equation (3), describing a randomly coincident fixation of base lesions or a randomly coincidently caused double strand break, we also have to take into account that both nucleotides may be damaged immediately one after the other by the same particle or by the same cloud of radicals originating from one particle. In this case no repair of the complementary base (nucleotide) is possible and therefore there is no τ in

the corresponding term, which is $\vartheta\dot{D}$. $\kappa(\dot{D})$ represents the probability that cells are killed by radiation in high dose regions. κ is the cell killing probability. Killed cells cannot become tumorigenic.

$$S_4=\left(C^2\alpha^2+2C\alpha\dot{D}\beta+\dot{D}^2\beta^2\right)\tau+\vartheta\dot{D}-\kappa(\dot{D}) \tag{3}$$

The following differential equation describes the time dependent behaviour of the arising of the first step in the successive process of damage.

$$\frac{dM_1(t,C,\dot{D})}{dt}=(B_0M_0-B_1M_1)\left(\left(C\alpha+\dot{D}\beta\right)^2\tau+\frac{1}{f_{nuc}}\bar{z}_F\beta^2\dot{D}-\kappa(\dot{D})\right) \tag{4}$$

$M_1(t, C, \dot{D})$ is the number of cells per individuum at time t which are in the first transformation step, i.e.: which have one point mutation on a critical, tumor relevant gene locus caused by a chemical carcinogen with concentration C or by a dose rate $\dot{D}$. M_0 is the number of all human cells which are not yet transformed. Equation (4) describes the first step necessary for a complete carcinogenesis. The number of steps necessary for the induction of cancer defines a system of coupled differential equations. Each transformation is represented by one differential equation. B_0 is the number of critical DNA bases (nucleotides) in critical codons of all tumor associated genes per cell. B_1 is the number of critical DNA bases (nucleotides) in critical codons of all tumor associated genes per cell after the first transformation.
The first sink on the right side of equation (4) ($-B_1M_1$ ...) is due to the fact that cells which have reached the first step of damage are a source term for the second step. M_1 is several orders of magnitude smaller than M_0. Therefore this sink can be neglected. In the case of constant chemical and radiological influences (C=const., $\dot{D}$=const.), the coefficients of the differential equation (4) are time independent and the equation can be directly integrated:

$$M_1(t)=B_0M_0\left(\left(C\alpha+\dot{D}\beta\right)^2\tau+\frac{1}{f_{nuc}}\bar{z}_F\beta^2\dot{D}-\kappa(\dot{D})\right)t \tag{5}$$

For $M_n(t)$ we get with $A_1:=B_0B_1B_2\ldots B_{n-1}M_0\frac{1}{n!}$

$$M_n(t)=A_1\left(\frac{C^2\alpha^2\tau_0}{\left(1+\delta_S\dot{D}\right)^2\left(1+\delta\dot{D}\right)}+\frac{2C\alpha\dot{D}\beta\tau_0}{\left(1+\delta_S\dot{D}\right)\left(1+\delta\dot{D}\right)}+\frac{\dot{D}^2\beta^2\tau_0}{1+\delta\dot{D}}+\frac{1}{f_{nuc}}\bar{z}_F\beta^2\dot{D}-\kappa(\dot{D})\right)^n t^n \tag{6}$$

where δ_s describes the extent of the production of additional scavengers caused by radiation. The term $\kappa(\dot{D})$ is approximately zero because cell killing is not a relevant effect in the low dose regions refered to in the data which are used for regression analysis. For an anambiguous data fit this equation has to be slightly transformed and we can only determine $\frac{\beta}{C\alpha}$ and $\delta=\delta_s$. For τ_0 we take 40 minutes /7/. $\frac{1}{f_{nuc}}$ can be derived as approximately 3.3 /8/. For a critical volume with a radius of 4.66 µm, the average radius of a bronchial epithelial cell nucleus, D.R. Fisher, V.P. Bond et al. /9/ received $\bar{z}_F$=0.5 Gy. Data fits are done with all 1601 data points using the Levenberg-Marquardt algorithm and the Nelder-Mead simplex search method. For n, the number of transformations, we take 4, which was received by a data fit of age dependent ARIP data for I.C.D. 140-205 (all cancers).

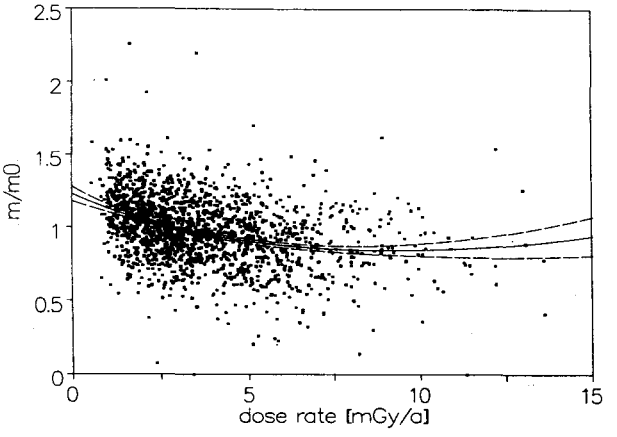

Fig. 1 Data fit of the smoking corrected, age adjusted lung cancer mortality rates (I.C.D. 163-164) for males; the 95 % confidence band is shown.

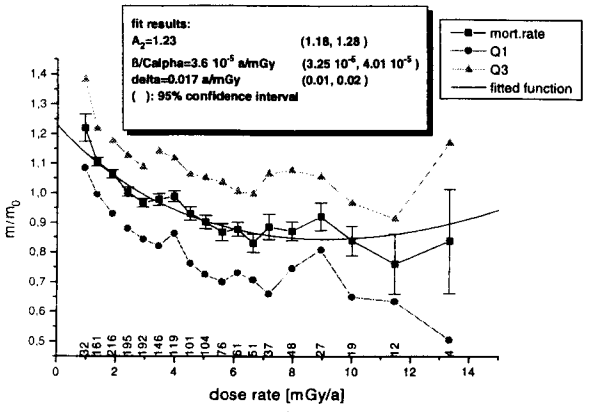

Fig. 2 Data fit of the smoking corrected, age adjusted lung cancer mortality rates (I.C.D. 163-164) for males, grouped into intervals of dose rate. The number of counties in each group is shown on the base line. We plot the mean value of m/m_0 for each group, its standard deviation, the first and third quartiles of the distribution and the fitted function.

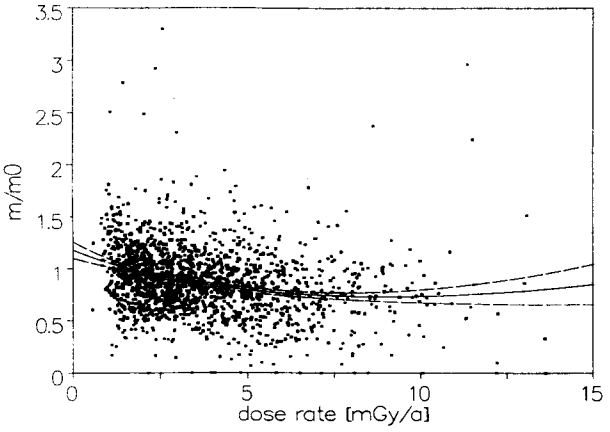

Fig. 3 Data fit of the smoking corrected, age adjusted lung cancer mortality rates (I.C.D. 163-164) for females; the 95 % confidence band is shown.

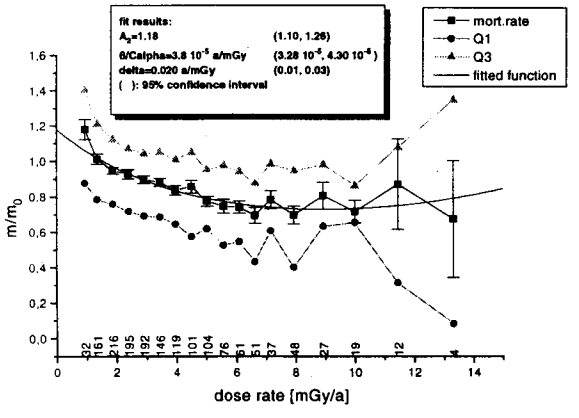

Fig. 4 Data fit of the smoking corrected, age adjusted lung cancer mortality rates (I.C.D. 163-164) for females, grouped into intervals of dose rate.

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OH WUNDER!
THE INVERSE DOSE-RATE EFFECT IS QUELLED BY THE EFFECTIVE THRESHOLD

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ABSTRACT

Three-dimensional dose-rate/time/response mathematical surfaces describe radiation effects in lifetime studies of beagles after intake by injection or inhalation of selected radionuclides (including α-emitters ^{226}Ra, ^{239}Pu, ^{238}Pu, and ^{241}Am and β-emitters ^{90}Sr, ^{91}Y and ^{144}Ce) and in people after intake of ^{226}Ra. For each effect $t_m = K_m \bar{d}^{-s}$, where t_m is the median elapsed time to death with the specified effect after intake, $\bar{d}$ is the time-weighted average absorbed radiation dose-rate to the target organ, K_m is the median distribution coefficient, and s is the negative slope parameter. Using maximum likelihood survival regression methods, s was found to be ⅕ for α radiation and ⅔ for β radiation. The relationships of competing risks of death associated with radiation injury, radiation-induced cancer, and natural aging after initial exposure or intake are graphically portrayed using three-dimensional illustrations of these dose-rate/time/response surfaces. At the higher dose rates (larger organ burdens of radionuclide) the principal deleterious effects are those associated with radiation-induced injury while at intermediate dose rates radiation-induced cancer predominates. For cancer induction, lower dose rates have a higher relative dose effectiveness than higher dose rates (*inverse dose-rate effect*) for either low LET β radiation or high LET α radiation. However, at the lower dose rates the long latency time required for development of radiation-induced cancer may exceed natural life span, yielding a *life-span effective threshold* for death associated with radiation-induced cancer at a lifetime cumulative absorbed dose to the target organ of from about 0.08 to 1.4 Gy for α irradiation or from about 28 to 130 Gy for β irradiation. The beagle results were scaled to predict human bone cancer risks for internally-deposited ^{226}Ra and the predicted life-span effective threshold of about 2.1 Gy agreed well with the U.S.A. human ^{226}Ra data. The predicted occurrence of human lung cancer from inhaled $^{239}PuO_2$ yielded a life-span effective threshold of about 1.4 Gy..

INTRODUCTION

In this report, radiation risk from internally deposited radionuclides is described by an independent risk model in which radiation risk is superimposed in time upon the natural lifetime causes of death and competing risks. In its simplest form there are three separate fatal risk distributions: (a) deaths associated with natural life span and aging including the background incidence of various types of fatal cancer; (b) those deaths associated with radiation-induced cancer; and (c) deaths associated with exposure-related non-neoplastic injury (1-7). Precise data obtained in lifetime studies utilizing beagles provide the basis for understanding these phenomena. Raabe et al. (8-9) used life-span normalization to scale response relationships from laboratory animal species to human risks.

METHODS

Selected data were reviewed from published reports of life-time studies of internally deposited radionuclides in young adult beagles at four laboratories: University of California, Davis (10), Lovelace Inhalation Toxicology Research Institute, ITRI (11), University of Utah (11), and Battelle Pacific Northwest Laboratory, PNL (12). Throughout this report, doses refer to average absorbed target organ doses from parent and decay products, where all x ray and gamma emissions are ignored, and beta emissions are also ignored where alpha emitters predominate. Because of an observed pattern of linearity between the logarithm of time to radiation induced cancer deaths and logarithm of lifetime average dose rate to the target organ, standard log-linear survival models and maximum likelihood regression methods were used to describe the response functions (4). In terms of the natural logarithms, ln, the log-linear regression line is thus given by:

$$\ln t_m = \ln K_m - s \ln d \rightarrow t_m = K_m d^{-s} \quad (1)$$

where $\bar{d}$ is the time-weighted average dose rate, t_m is the regression value of median survival time, s is the negative slope of the regression line for the specified effect, and K_m is the median risk coefficient.

RESULTS AND CONCLUSIONS

The results are summarized in Figure 1. Alpha emitting radionuclides all yielded cancer risk distributions with slopes of -1/3 and the beta emitting radionuclides all yielded cancer risk distributions with slopes of -2/3, indicating that two beta particles are required to effect the same cellular transformations effected by one alpha particle such as two-strand DNA breaks (6). In addition, the underlying slope of -1/3 indicates that the distance between these events controls the response as a function of survival time as discussed by Raabe (3).

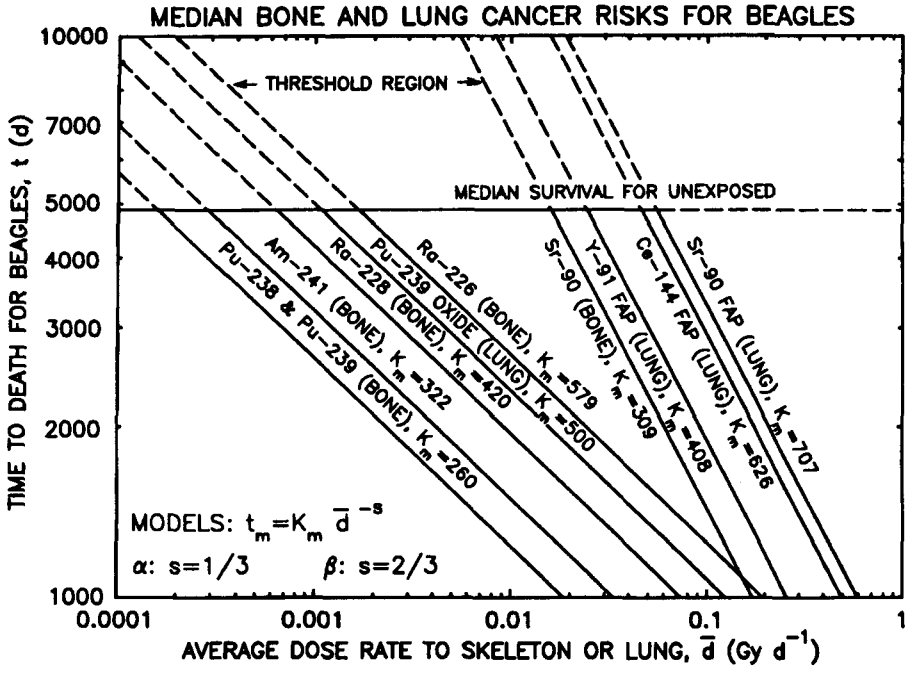

Figure 1. Median bone and lung cancer risks for beagles.

Because of the negative slopes of these distributions, the cumulative absorbed dose required to yield a specified level of cancer risk is less at lower dose rates than at higher dose rates (*inverse dose-rate effect*), but the time required for tumors to manifest tends to be much longer at lower dose rates and can exceed the natural life span, yielding a *life-span effective threshold* for fatal radiation-induced cancer. In other words, *Oh Wunder!* The old-age-related death of the individual preempts the development of radiation-induced cancer. For young adult beagles, this threshold occurs at about 0.08 Gy for ^{238}Pu and ^{239}Pu in bone, 0.2 Gy for ^{241}Am in bone, 0.5 Gy for ^{228}Ra in bone, 0.9 Gy for ^{239}Pu in lung, 1.4 Gy for ^{226}Ra in bone, 28 Gy for ^{90}Sr in bone, 70 Gy for ^{91}Y in lung, 110 Gy for ^{144}Ce in lung, and 130 Gy for ^{90}Sr in lung. Effective thresholds are predicted for human exposures using life-span normalization (8-9). For example, it is about 3.5 Gy for ^{226}Ra in bone; there were no cases of bone cancer for any cumulative dose less than 10 Gy in U.S. radium cases (13). Likewise, the predicted human occurrence of radiation-induced lung cancer (Figure 2) from inhaled $^{239}PuO_2$ has an effective threshold of about 2 Gy (8).

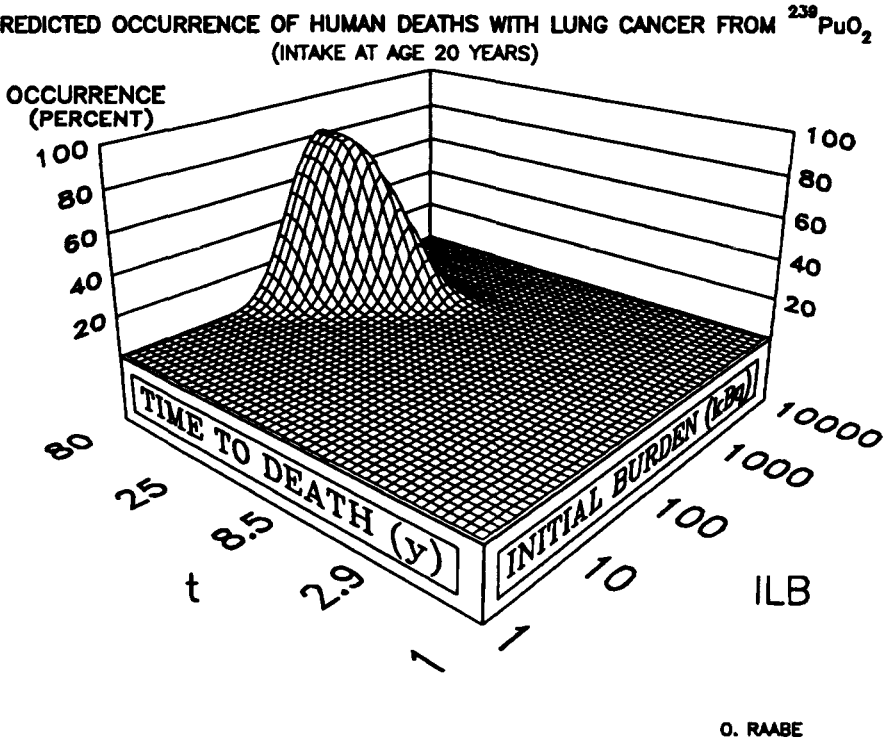

Figure 2. Predicted lung cancer from inhaled $^{239}PuO_2$ by initial lung burden (ILB).

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ACKNOWLEDGEMENTS

Research supported by the United States National Cancer Institute under Grant number R01 CA46296.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE The Mathematical Model for Risk Assessement

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MAJOR SCIENTIFIC TOPIC NUMBER 2.3 (see page 7)

ABSTRACT (See instructions overleaf)

The suggested model can be used for constructing of the objective number measure of the rising risk of the definite degree of the damage.The effect of such action is described by the binary index when taking two values **e=0** ("no" effect) and **e=1** (presence of the effect).The effect **e=e(r,a)** is the function of two arguments, where **a**-the dose(intensivity of radiation action), **r**-the resistance. In the simplest case the function **e(r,a)** may be set by the formula:

$$e=e(r,a)=\begin{cases}0, & a \le r\\ 1, & a > r,\end{cases} \text{ where } r,\ a \in [0,\infty).$$

According to this formula the resistance **r** has a meaning of the "threshold", i.e. the effect is absent as long as the action **a** will not exceed the meaning **r**. The probability $P\{r'<r\}$ of **r'** being smaller than the fixed resistance **r** is given by the function of distribution $P\{r'<r\}= F_{r'}(r)$. The randomized action of doses **a'** is introduced by the analogy with randomized resistance **r'**. It is described by the function of distribution $F_{a'}(a)$: $P\{e'(r)=1\} = 1-P\{e'(r)=0\} = 1 - F_{a'}(r)$. The randomized effect **e'(a)** can be given in the following way:

$$e'(r) = \begin{cases}0, & \text{with the probabililty } F_{a'}(r);\\ 1, & \text{with the probabililty } 1-F_{a'}(r).\end{cases}$$

"Dose-effect" curve deduced from these relations implies the next information about the action of irradiation: 1) Coinciding with the function of distribution $F_r(r)$, the "dose- effect" curve describes the distribution of meanings of the radioresistance threshold among the individuals of the studied population; 2) The "dose-effect" function being the probability of appearing of the effect **e=1** at the random taken organism determines the fraction **m(a)** of the individuals with the observed effect **e=1** under the action of the dose **a**; 3) The "dose-effect" function being the mathematical expectation of the effect value observed after the dose **a** is the measure of the mathematical expectation of the individual damage weight among the population irradiated with dose **a**.

A MATHEMATICAL FORMULATION OF RADIATION RISK

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INTRODUCTION

The purpose of radiation protection (RP) is self-evident but is not so easy to establish a target for it. The aim of RP has been set to prevent the so-called deterministic effects and also to suppress the stochastic effect to some acceptable limit. We have learnt through our experience of more than thirty years with this subject that under provision of the ordinary measures to achieve the latter aim, the former is usually automatically carried out.

Thus, the measure of the stochastic effect, radiation risk, is one of the most important two quantities in the causality relation, indispensable to radiation protection as a science. The quantity of cause is of course the radiation dose.

In this paper, issues of the definition and mathematical formulation of the radiation risk are discussed.

THE STATUS QUO

The term risk is widely used not only in the field of RP but also in many other fields. However, definition or significance of the word is not the same. In the field of RP, the word is often used to mean the probability of loosing an individual's life or mathematical expectation of the whole loss of lives in a population when it is used as the name of a quantity. While in some other fields including nuclear safety, risk is used as a synonym for the mathematical expectation of the magnitude of the undesirable consequence of event.

The risk defined as a synonym for the probability of incidence of lethal cancer, for example, is frequently understood as the rate of death due to cancer of an individual and expressed in the form of a percentage. Arithmetic treatment of this quantity we find unconvincing because total of these rates is fixed as one and the rate of carcinogenecis cannot be changed without in principle reshuffling other rates. Moreover, appearance of new death factors or increase in value of an existing other factor makes the value of the radiation risk decrease. When social desasters such as famine or AIDS occurs, the radiation risk decreases as result. Thus the risk of radiation defined with the probability is not suitable to discuss the risk of an individual.

ICRP introduced a concept of risk coefficient, risk per unit (effective) dose, which is used in the present system of radiation protection. However, as is well known, harmful stochastic effects of radiation on individuals depend on the age of exposure and their appearance accompanies stochastic "latent periods." On the other hand, risk of an exposure at time t_0 in the past ($t_0 < 0$) for the time passed is deterministic and for the future is stochastic. The function of life existence can have values of either one or zero for the past, while it can have a value of a positive real number between zero and one. Thus, even with the same exposure, evaluated risk depends on the time of evaluation even if the character of the population is stable. Therefore, the ICRP risk coefficient is not so useful for the purpose of discussing the risk to an individual.

Shortening of the mean residual life span of a population is often used as an expression of risk. This is a mathematical expectation. Extinction or depletion of the species is also important as a harmful effect of radiation on human beings and this quantity is useful for considering this problem. However, it is not suitable for discussing an individual's risks, because its value depends on the size of the population.

A MATHEMATICAL FORMULATION OF RADIATION RISK

In order to overcome the above-stated difficulties of the present situation, a reformulation of the radiation risk is tried on the premise of linear proportionality between the dose and the effect and on the newly specified concept of radiation risk.

We define the risk accompanying an act of receiving some amount of radiation dose as the expected value of shortening of the residual life span of a hypothetical standard man (or woman) at the time of exposure. The standard man (or woman) is a hypothetical individual characterized by the statistical data, i.e., the mortality rate as a function of age, $\lambda(\tau)$, with a population of interest. The data is obtained from a national census or such. His or her residual life span at age τ, $\rho(\tau)$, is calculated with the mortality function $\lambda(\tau)$ by the following relations;

$$-dN(\tau)/d\tau = \lambda(\tau)N(\tau) \tag{1}$$

$$\rho(\tau_0) = \int_{\tau_0}^{\infty} \tau\, \sigma(\tau_0, \tau)\, d\tau - \tau_0 \tag{2}$$

where τ_0 is the present time (age) and $\sigma(\tau_0, \tau)$ is the probabolity of maintaining life of an individual at age τ,

$$\sigma(\tau_0, \tau) = 1 \qquad \text{for } \tau \le \tau_0$$

$$\sigma(\tau_0, \tau) = N(\tau)/N(\tau_0) \qquad \text{for } \tau > \tau_0 \tag{3}$$

$N(\tau)$ is the population of the standard men (or women) of age τ.

Now the risk brought by unit amount of instantaneous dose is expressed as follows:

$$r(\tau, \tau') = \rho(\tau) - \rho'(\tau, \tau') \tag{4}$$

where $\rho'(\tau, \tau')$ is the new function or curve of the expected residual life span for an instantaneous exposure of unit amount of dose at the time (age) of τ'. The function $\rho'(\tau, \tau')$ can be evaluated if the data on the distortion of the function $\lambda(\tau)$ for an instantaneous exposure of unit amount of dose at time of τ' is available.

$$\dot{\varepsilon}(\tau) = \delta(\tau'), \tag{5}$$

where $\delta(\tau)$ is the Dirac's delta function and $\varepsilon(\tau)$ is a time spectrum of the dose,

$$\dot{\varepsilon}(\tau) \equiv dE(\tau)/d\tau$$

Thus the risk (for the future only by nature) evaluated at the time τ for the past exposure $\dot{\varepsilon}(\tau')$ is expressed as

$$R(\tau) = \int_{\tau}^{\infty} r(\tau, \tau')\, \varepsilon(\tau')\, d\tau' \tag{6}$$

Here, the risk or expected shortening in residual life span is proportional to the dose is assumed. That for the past is zero due to the fact of survival at the time of evaluation.

CONCLUSION

Risk for radiation exposure is defined as the expectation of the shortening of the residual life span. With this definition, the concept of risk is only for future life, since the fact of main-

taining life at the time of evaluation negated the possibility of losing the life in the past.

A mathematical formulation is shown under the premise of the linear proportional relation between the dose and the end effect (newly defined risk) and on the new concept of "Standard Man/Woman" whose characteristic on mortality is the same of the population of interest.

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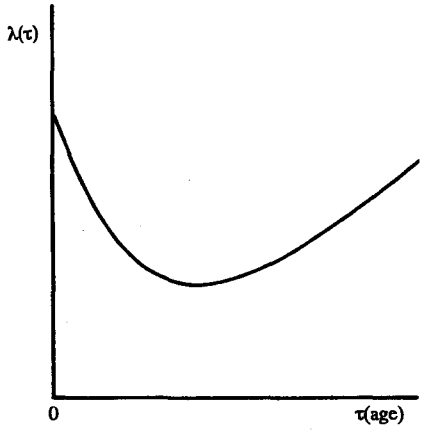

Fig.1 Mortality Function

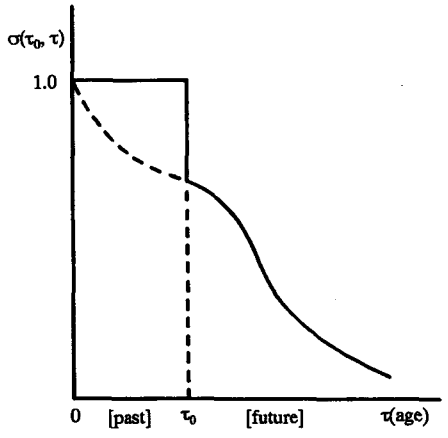

Fig.2 Existence Function

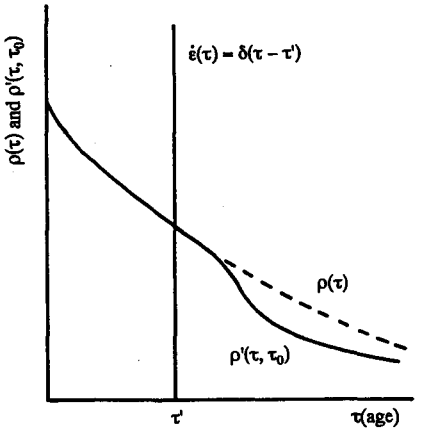

Fig.3 Expected Residual Life Span

USE OF RISK PROJECTION MODELS FOR THE COMPARISON OF MORTALITY FROM RADIATION-INDUCED BREAST CANCER IN VARIOUS POPULATIONS

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Lifetime risk estimates for radio-induced fatal breast cancer were compared in Japanese, North American and French populations for several epidemiological studies. In the case of breast cancers, some differences appear between the epidemiological studies according to the reference population. Thus, this comparison aims at pointing out how this affect the lifetime risk estimates when Japanese and Occidental populations are used.

METHODS

Six studies on mortality from breast cancer after irradiation were selected (Table 1.).

Table 1. Characteristics of epidemiological studies on breast cancer after irradiation

Reference	Study	Follow-up period *(mean follow-up)*	Exposed population *(deaths from breast cancer)*	Age at irradiation	Relative risk $(Gy)^{-1}$	Excess risk $(10^4 \text{ PY Gy})^{-1}$
Shimizu et al. (1990)	Hiroshima/ Nagasaki	1950-85	45,557 *(155)*	0-9 10-19 20-29 30-39 40 +	2.90 3.34 2.21 2.26 1.11	0.32 2.23 1.21 1.54 0.18
UNSCEAR (1988)	Hiroshima/ Nagasaki	id	id	all	2.19	1.20
Darby et al. (1987)	Ankylosing spondylitis	*(23 years)*	2,334 *(26)*	25-55	2.24	-
Davis et al. (1989)	Fluoroscopic examinations in Massachusetts tuberculosis patients	1925-86 *(25 years)*	3,329 *(62)*	all	1.57	3.14
Miller et al. (1989)	Fluoroscopic examinations in Canadian tuberculosis patients	1950-80 *(24 years)*	8,371 *(163)*	10-14 15-24 25-34 35+	4.46 1.77 1.25 1.10	6.06 3.05 1.72 1.23
National Academy of Science BEIRV (1990)	Hiroshima/ Nagasaki + medical studies	-	-	<15 ≥15	*Linear model considering age and time since exposure*	- -

The risk coefficients have been transferred to the 3 studied populations (Japan, USA, France), using demographic data from the 3 countries, risk projection models (additive or multiplicative), and assuming an individual acute exposure of 10 mSv. The calculations have been performed with the PC-based software "Assessment System for the Quantification of Radiation Detriment" (ASQRAD).

RESULTS

The excess lifetime risk from radiation-induced breast cancer was compared between the 3 populations, using the risk coefficients either from the Japanese A-bomb study (Table 2) or from the study based on the Canadian fluoroscopic examinations of patients treated for tuberculosis (Table 3).

Table 2. Number of lifetime excess deaths from breast cancer per 100,000 persons individually exposed to 10 mSv acute radiation (Shimizu et al. 1990) for France, USA and Japan demographic data

Age at exposure (y)	Japan 1988	USA 1987	France 1987
Additive model			
10	13.5	12.9	13.3
20	6.1	5.8	6
30	6.3	5.9	6.2
45	0.48	0.44	0.46
Multiplicative model			
10	18.5	75	71.8
20	9.5	38.8	37.1
30	9.4	39.6	38.1
45	0.6	2.9	2.8

Table 3. Number of lifetime excess deaths from breast cancer per 100,000 persons individually exposed to 10 mSv acute radiation (Miller et al. 1989) for France, USA and Japan demographic data

Age at exposure (y)	Japan 1988	USA 1987	France 1987
Additive model			
10	36.7	35	36
20	15.5	14.6	15.1
30	7	6.6	6.9
45	3.3	3	3.2
Multiplicative model			
10	27.3	111	106
20	6.1	24.7	23.7
30	1.9	7.9	7.6
45	0.54	2.7	2.6

The lifetime risk estimates are similar in the 3 populations when the additive projection model is used because of the similarity of the life tables of the 3 countries. However, using the multiplicative model, large differences appear due to the discrepancies in the baseline breast cancer rates between the Japanese and Occidental populations. For this reason, transfer of risk coefficients fitted on the Japanese population data must be considered with caution when applied to an Occidental population. The risk projections obtained from different epidemiological studies have been compared in the French population (Table 4). Lifetime risk estimates are largely dependent on the risk coefficients considered. Nevertheless, for each source of data, they are higher with the multiplicative one.

Table 4. Number of lifetime excess deaths from breast cancer per 100,000 persons individually exposed to 10 mSv acute irradiation (French demographic data, 1987) for additive and multiplicative models

Age at exposure (y)	Shimizu et al. (1990)	UNSCEAR (1988)	Darby et al. (1990)	Davis et al. (1989)	Miller et al. (1989)	BEIR V (1990)
Additive model						
10	13.3	7.1	-	18.7	36	-
20	6	6	-	15.6	15.1	-
30	6.2	4.8	-	12.5	6.9	-
45	0.46	3.1	-	8.1	3.2	-
Multiplicative model						
10						
20	71.8	36.5	-	17.5	106	19.7
30	37.1	36.5	-	17.5	23.7	6.6
45	38.1	36	37.5	17.2	7.6	5.6
	2.8	30.8	32	14.7	2.6	2.9

CONCLUSION

The choice of a transfer model is all the more difficult that mechanisms of radiation-induced cancer are not yet well understood. If breast cancer death rates are higher in the American population than in the Japanese one because Americans tend to be exposed to more breast cancer initiators early in life and if radiation also acts as an initiator, then the additive model might be expected to fit well. On the other hand, the multiplicative transport model would be more appropriate if the difference between populations could be due to differential exposure to promoters later in life. As far as breast cancer is concerned, the characteristics of the reference population, where the risk coefficients have been defined through epidemiological studies, are of particular importance when these risk coefficients are transferred to another exposed population. This is due to the possible large variations in breast cancer rates between various populations.

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MONTE CARLO SIMULATION OF Z ≥ 2 ION TRACK DISTRIBUTION IN BONE TISSUE

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ABSTRACT

Heavy ion tracks distribution up to 2 GeV / amu in bone tissue was simulated in order to assess the biological effects of the radiation. Calculation of ion tracks distribution in the target was materialized using the code TRIM (1).
Average energy transferred by ion tracks in nuclear as well as in atomic collision is obtained and pattern of ion tracks is analyzed for comparison. The computer program provides particularly high computer efficiency, when a new analytical formula used in determining nuclear scattering angles (2). Heavy ion therapic parameters, such as the width of the spread-out Bragg peak (Figure 1), Q values and H dose equivalent are obtained and compared with existing data (3).

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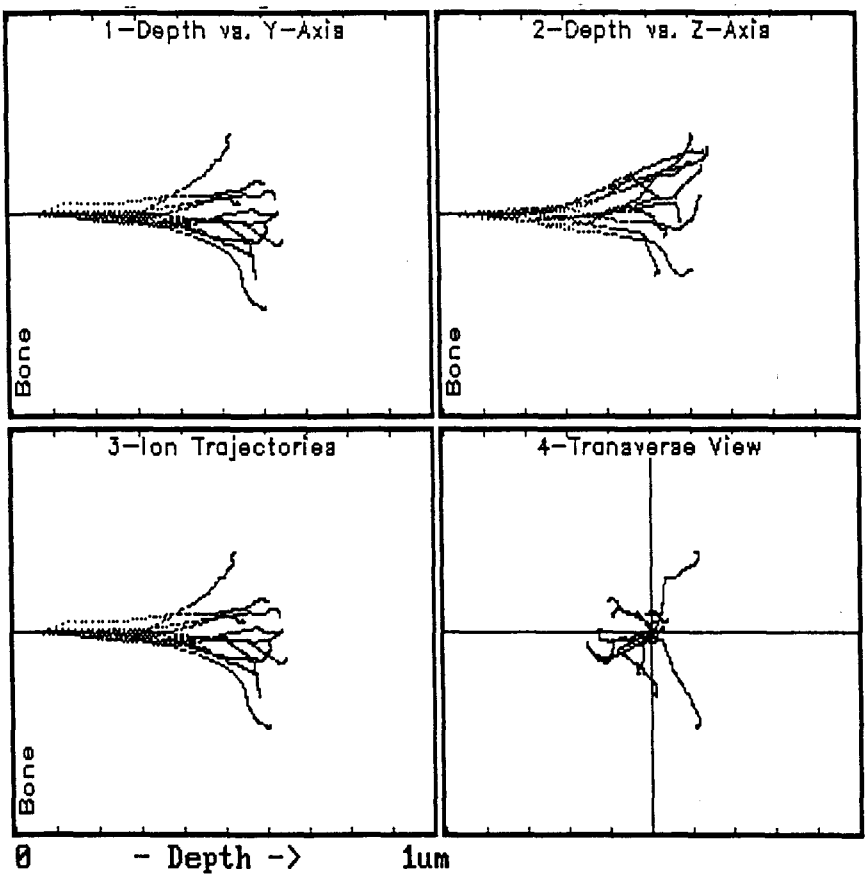

Figure 1. 100 keV He ion tracks distribution simulation in bone tissue.

Biological effects of the combined impact of low radiation doses & chemical agents

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This report examines biological effects of the combined and separate impact of low ionizing radiation doses on the organism and molecular levels. Synergetic effects were found. Radiation effects of thyroid hormones synthesis, peroxide oxidation of lipids, formation of proteins are examined.

THE DOSE RATE AS A MODIFIER OF BIOLOGICAL EFFECTS IN CONDITIONS OF THE COMBINED ACTION OF RADIATION-CHEMICAL FACTORS

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INTRODUCTION

The proper evaluation of the risk of the delayed effects in the case of equal absorbed doses but different dose rates in the conditions of the additional impact of the factors of nonradiation nature is a complicated biological and mathematical problem.

Insufficiently correct evaluation of the multi-factor effect of harmful substances on organizm and ascribing the registered negative aftereffects only to the impact of the radiation can result in both the overestimation of the radiation danger and the inefficient investments of funds into the system of countermeasures for reduction of risk of irradiation [1].

Taking into account the real character of the impact of the set factors of the radiation and nonradiation nature on organism, the effect of the dose rate has been studied in the conditions of additional influence of chemical compounds on the process of structural damages of chromosomes and induction of lung adenoma. Nitrogen oxides (one of the dominating atmospheric pollutants) and urethan, which is the pulmonotropic mutagen, have been chosen as chemical substances.

MATERIALS AND METHODS

The studies have been carried out using the mice of Af line in two series differ in the method of irradiation. In the first series mice have been exposed to the whole body single irradiation of Co gamma rays at a dose of 0.35 Gy and a dose rate of 1.0 Gy/h. In the second series the mice have been irradiated at a total dose of 0.35 Gy, but at a dose rate of 0.008 Gy/h (0.0389 Gy · 9 days).

Each series consisted of 8 groups: 1 - control, 2 - irradiation, 3 - inhalation of nitrogen oxides at a concentration of 120 mg/m^3 for 30 minutes, 4 - administration of urethane intraperitoneally (1 mg/g weight), 5 - inhalation of nitrogen oxides followed by administration of urethan an hour later, 6 - inhalation of nitrogen oxides in the same concentration an hour later after irradiation, 7 - administration of urethan an hour later after irradiation, 8 - the mice have been exposed to the action of three factors in the following sequence: irradiation + nitrogen oxides + urethan.

The cytogenetic analysis has been carried out on marrow cells in line with methods given in [2,3]. The cancerigenic effects have been determined according to the number of the induced lung adenoma in 20 weeks after the impact [4].

In groups of the combined effect of the factors, the coefficient of synergy (SC) has been determined with correction for the background [5].

RESULTS AND DISCUSSION

Chromosome aberrations (ChA)

The change of the frequency of the chromosome aberrations is shown on fig.2.

The spectrum of ChA has been presented by the fragments (single and pair), intrachromosome (cyclic rings) and interchromosome (symmetrical translocation) exchanges. The cells with multiple damages has also been registered.

The analysis of the spectrum has shown, that deletions have been the dominant type of aberrations in all experimental groups. The largest quantities of fragments have been found in the groups of the combined effect under a single exposure to radiation (up to 95% of all types of aberrations).

With prolongation of the dose, the fraction of simple deletions has been smaller and accounted for 75%. In the groups of the combined effect with lower dose rate, the aberrations of the chromosome type dominated. The number of the cyclic rings in case of the action of three factors against the background of the prolonged exposure to radiation has been four times as many as in the case of a single exposure to radiation under the same combination of factors (18.5% - 4.8% of the total number of ChA, respectively). In addition, the cells with multiple aberrations (more than 6) have been registered in this group.

The stable aberrations (total symmetrical exchange) have been under the three-factor effect with the background of prolonged exposure to radiation. The fraction of these aberrations in the total number of ChA has been 2.7% and 7.4% of the total number of ChA, respectively.

Lung adenoma

The induction of lung adenoma is shown on fig.2.

The effect of nitrogen oxides on mice under a single exposure to radiation was manifested as the inhibition of the process of adenoma formation in comparison with the separate effects. On the contrary, the prolongation of radiation exposure dose with inhalation oxides has been characterized by the increase of the frequency of adenoma - 2.06 (SC = 1.73).

In the group with combination of a single exposure to radiation and administration of urethan, the number of adenoma has reached the maximum value 14.06 adenoma/mouse, and SC = 4.46. The administration of urethan after the prolonged exposure to radiation has increased the cancerigenic effect, but the number of adenoma has been lower in comparison with the group of a single exposure to radiation, amounting to 4.31 (SC = 1.44).

In the group of three-factor effect and a single exposure to radiation, the number of adenoma/mouse has amounted to 6.67 (SC = 1.82). The prolonged exposure to radiation has increased these figures to 10.73 adenoma/mouse, and SC up to 3.48. With three-factor effect, SC in the group with the prolonged exposure to radiation has exceeded the indices of the similar group under a single exposure to radiation by a factor of two, and the growth of adenoma - by 1.6 times.

The treatment of the mechanisms of aberrations found out under the combined effect of radiation-chemical factors is significantly more complex as in the formation of ChA and tumor induction both the direct damage of DNA and the efficiency of functioning of the reparation system of a cell are important. The effect of the additional factors at this period can both strengthen the processes of restoration and facilitate its retardation especially in the cases, when the coefficient of synergy exceeds unity. The latter evidences the inclusion of additional mechanisms into the realization of registered changes, which are out of the scope of the additivity mechanisms.

The following mechanisms can be supposed as the above mentioned:

- Apart from the direct effect of each of the agents able to induce the DNA damage, possible additional influence on latent damages, which have been induced by the effect of the first agent, should be taken into account, including their transfer into the cytogenetically registered form, that is, ChA under the effect of the second agent [6, 7].
- Formation of a new substance from the active chemical compounds possessing mutagenic and cancerigenic properties intensifying the effects of precursors [8, 9].
- Stimulating effect of the ionizing radiation on the synthesis of carcinogenic agents [10].

It follows from the obtained data, that under the multi-factor effect, the reduction of the dose rate can result not in the increase, but in the decrease of frequencies of tumors. Therefore, the delayed consequences, associated with the dose effects, in particular, the processes of cancerigenesis, can be underevaluated in the real ecological conditions. Presently available observations on the alteration of the sensitivity of the irradiated organism to the effect of the nonradiation factors, as well as the radiation-chemical synthesis of mutagens from the containing in organism and the entering precursors confirm the possibility of existence of such a process [11, 12, 13].

Unsufficient evaluation of the modifying effect of radiation on the changes of the sensitivity of organism to the action of the chemical compounds, the level of which are considered to be safe, and attributing the registered modification only to the affect the scale and the direction of the countermeasures on elimination of the radiation risk.

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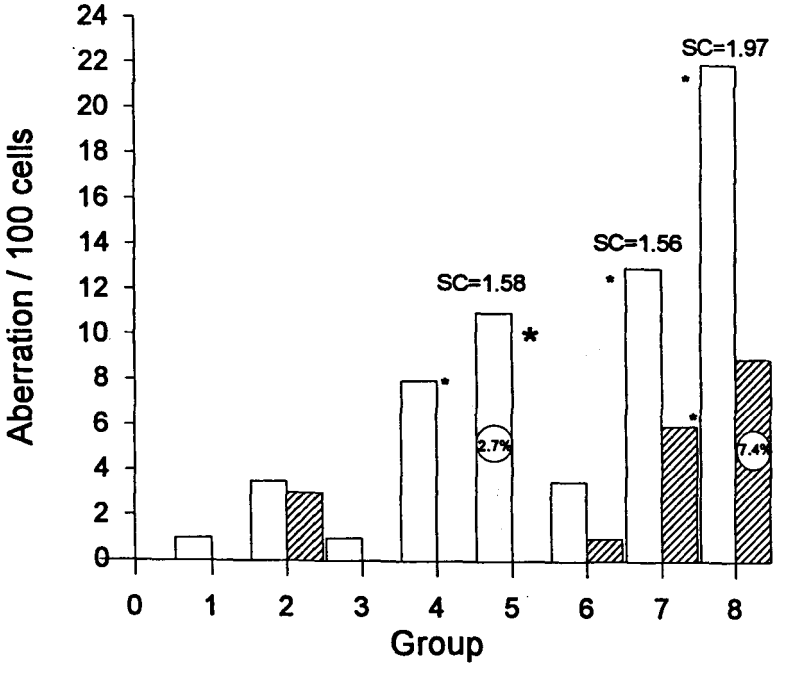

Fig.1. Frequency of chromosome aberrations in mice marrow cells
Y axis - number of aberrations on 100 metaphases
X axis - groups of experimental animals (in Materials and Methods)
- Total single γ-irradiation - prolongated γ - irradiation
- Presence of nonreparable aberrations
* - Difference is statistically reliable to control (P<0.05)

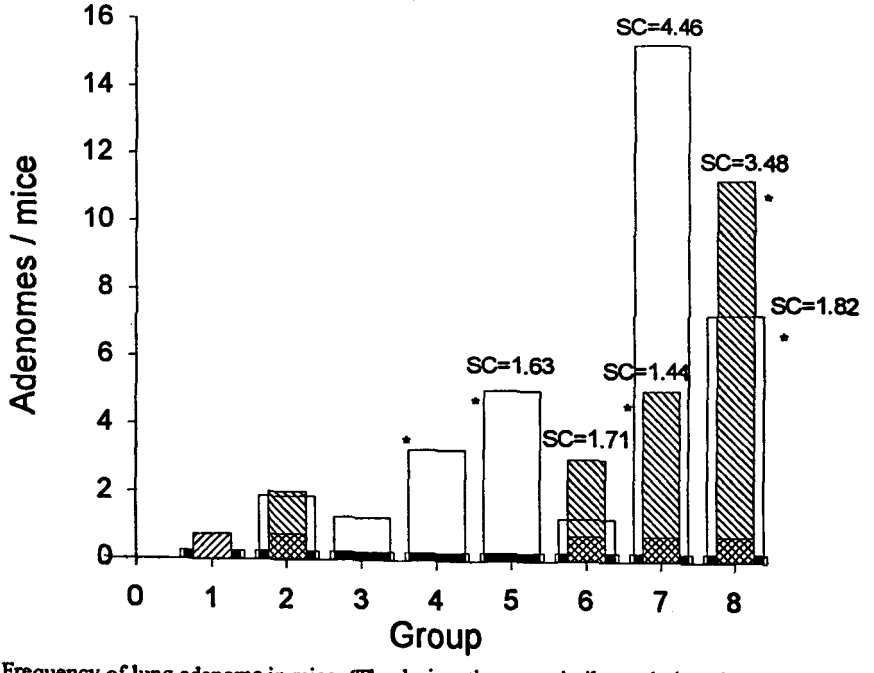

Fig.2. Frequency of lung adenoma in mice (The designations are similar to designations of Fig.1.)

- control of the group of total single γ- irradiation
- control of the group of prolongated γ - irradiation

ASSESSMENT OF AGE-SPECIFIC RADIATION RISKS TO PATIENTS WITH NON-UNIFORM DOSE DISTRIBUTION

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ABSTRACT - In medical radiology, a typical radiation dose distribution within the patient's body is extremely non-uniform, and the patients' typical age distribution is clearly different from that of the whole population. The effective dose and the nominal probability coefficients, as defined in the 1990 Recommendations of the ICRP for radiological protection and for assessment of risks in general terms, apply to workers and to the whole population. For estimating the risks from a known exposure to a known population, e.g. to patients in certain X-ray examinations, it is better to use specific data relating to that exposed population.

The modified relative risk model of the BEIR V (1990) report allows assessment of the radiation risk as a function of the age at exposure and the time after exposure, separately for males and females. Fitted parameters are given for five specific organs or organ groups. The model is directly applicable if the dose distribution is uniform within each of the organ groups. Otherwise, some extra information or extra assumptions are needed for risk assessment. In this work, the BEIR V model is used with Finnish statistical data on cancer and mortality. Some approximative assumptions are presented and discussed, applied to selected X-ray examinations, and compared with uniform exposure of the population.

INTRODUCTION

In the 1990 Recommendations of the ICRP (1), average coefficients for the probability of radiation-induced fatal cancer are defined for twelve organs or tissues and for the remainder group. The tissue weighting factors and the effective dose are then defined, considering the estimated loss of life expectancy and the contributions of non-fatal cancers and genetic effects. The quantities used in the definitions are averages calculated for several populations, both sexes and a wide range of ages. Because of this averaging, there is no explicit dependence on age, time or sex in the definition of the effective dose.

The BEIR V report (2) presents a modified relative risk projection model with explicit dependence on the age at exposure, time after exposure and sex. The model can be applied together with national mortality and cancer mortality rates and age distributions, and with the age distributions of specific groups. It is one of the risk projection models used by the ICRP in preparing the 1990 Recommendations. With some modifications, it is also used by the NRPB (3). The use of the radiation risk projection models - and the associated uncertainties and problems - have been discussed by many authors (1-7). If the dose distribution within the body is strongly non-uniform, a special problem arises from the application of collective groups of organs (8,9).

METHODS

In principle, the radiation-induced age-specific excess mortality rate, R(a;e,D) is the sum of all organ-specific mortality rates, $r_k(a;e,d_k)$ caused by the organ doses, d_k:

$$R(a;e,D) = \sum r_k(a;e,d_k)$$

where a is the age, e is the age at exposure and D is a symbol for the configuration of all organ doses. In practice, certain organ-specific functions, r_k, may be available for some organs, and one or more organ groups may have given group-specific functions, r_g :

$$R(a,e,D) = \sum r_k(a,e,d_k) + \sum r_g(a,e,d_g)$$

The problem is how the group-specific doses, d_g, should be estimated if the dose distribution is not uniform within each of the organ groups. The problem can be examined within any one of the groups; in the following, the group subscripts are deleted: $r \equiv r_g$, $d \equiv d_g$, and $\sum r_k$ is the sum of organ-specific mortality rates within the group under consideration:

$$r(a,e,d) = \sum r_k(a,e,d_k)$$

In the following, some simplifying assumptions are made, and the resulting relations are presented for the absolute (A) and relative (R) risk models.

In functions r and r_k, the dose dependence is assumed to be separable;

A: $r(a,e,d) = f(d)\, g(a,e) = \sum f_k(d_k)\, g_k(a,e)$

R: $r(a,e,d) = f(d)\, g(a,e)\, r_0(a) = \sum f_k(d_k)\, g_k(a,e)\, r_{0k}(a)$

Functions r_0 and r_{0k} are the baseline mortality rates of the population, and $r_0 = \sum r_{0k}$.

The age and time dependence is assumed to be the same for all organs in the group: $g_k \equiv g$;

A: $f(d) = \sum f_k(d_k)$

R: $f(d)\, r_0(a) = \sum f_k(d_k)\, r_{0k}(a)$

A simple linear dose response function, $f = \alpha d$ and $f_k = \alpha_k d_k$, is assumed for all organs in the group;

A: $\alpha\, d = \sum \alpha_k d_k$

R: $\alpha\, d\, r_0 = \sum \alpha_k d_k r_{0k}$

If the dose distribution is uniform, $d_k \equiv d$ for all organs;

A: $\alpha = \sum \alpha_k$

R: $\alpha\, r_0 = \sum \alpha_k r_{0k}$

These relations apply to any dose configuration because the risk coefficients, α and all α_k, are independent of the organ doses. For practical calculations, more information or extra assumptions about the unknown α_k coefficients are needed. The four equations above are then solved simultaneously with the extra relations.

First assumption: All coefficients within the group are the same, $\alpha_k \equiv \alpha'$;

1A: $\alpha = n\, \alpha'$ and $d = \sum d_k / n$ (n is the number of organs in the group)

1R: $\alpha = \alpha'$ and $d = \sum d_k r_{0k} / r_0$

If, in the R model, the absolute risk coefficients are required to be equal ($\alpha_k r_{ok} = C$) then

1R': $\alpha\, r_0 = n\, C$ and $d = \sum d_k / n$

Second assumption: The ratios of the coefficients are furnished with specific weighting factors, w_k ;

$\alpha_k = w_k \alpha'$ and $\sum w_k = 1$

2A: $\alpha = \alpha'$ and $d = \sum w_k d_k$

2R: $\alpha\, r_0 = \alpha' \sum w_k r_{0k}$ and $d = \sum w_k r_{0k} d_k / \sum w_k r_{0k}$

If, in the R model, the absolute risk coefficients need to have constant ratios ($\alpha_k r_{ok} = w_k C$), then

2R': $\alpha\, r_0 = C$ and $d = \sum w_k d_k$

In the following examples, the BEIR V relative risk model is applied to Finnish demographic data, mortality and cancer mortality rates, patients' age distribution, and estimated organ doses in a typical chest CT examination. The lifetime risk projections are calculated using two of the assumptions above: 1R and 2R'. The weights, w_k for the 2R' case, are calculated from the nominal probability coefficients of ICRP 60 (1). For comparison, the same lifetime risk quantities are also calculated from a hypothetical uniform dose distribution corresponding to the effective dose of the same chest CT examination.

RESULTS

The most prominent organ doses (in mSv) in the chest CT examination are: lungs 18; breasts 16; red bone marrow 2; digestive organs: oesophagus 15, stomach 2, liver 3, gall bladder 1, pancreas 3, spleen 3; and other organs: adrenals 3, kidneys 1, skin 5, thymus 25, thyroid 7, muscles 5, bones 9 mSv. According to the ICRP 1990 Recommendations (1), the remainder organ dose is 13 mSv, and the effective dose 6 mSv. These values are rounded averages of a sample collected from Finnish hospitals. The weighted group-specific doses according to the 2R' case are 3.2 mSv for the digestive organs and 7.5 mSv for the BEIR V group of other organs.

The lifetime risk projections: the excess lifetime risk (ELR), the risk of exposure-induced death (REID), and the loss of life expectancy (LLE) are calculated according to Thomas et al. (4). The REID values, calculated as a function of age at exposure, are shown in Figure 1. The mean values of REID and LLE, according to the relevant age distributions, are presented in Table 1.

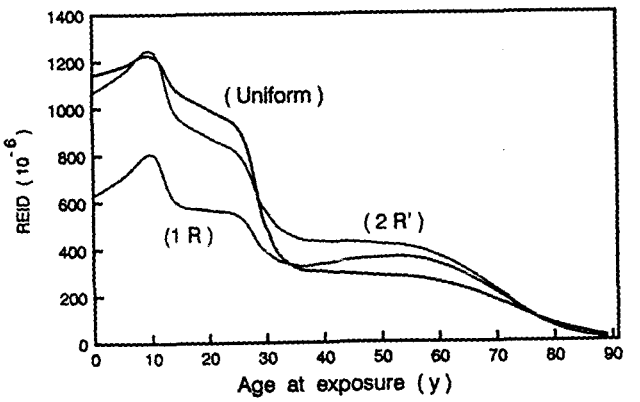

Figure 1. The REID for Finnish females in chest CT examinations, calculated using two different assumptions (1R and 2R' as defined in the text) and a uniform dose distribution for comparison.

Table 1. Mean REID (10^{-6}) and LLE (10^{-6}y) for Finnish females in chest CT examinations and with uniform dose distribution.

Target group:	CT Patients		CT patients		CT patients		Population	
Assumption:	1R		2R'		Uniform		Uniform	
Quantity:	REID	LLE	REID	LLE	REID	LLE	REID	LLE
Leukaemia	16	200	16	200	47	590	37	740
Breast cancer	44	910	44	910	16	340	32	790
Respiratory	180	2090	180	2090	60	700	50	660
Digestive	50	480	70	660	130	1240	270	2720
Other cancers	40	420	110	1250	90	1000	170	2120
All cancers	330	4100	420	5100	340	3870	560	7030

DISCUSSION

The problem of assessment of the group-specific dose is eliminated if the dose distribution is uniform, but its significance increases with increasing non-uniformity. The problem is closely related to the problems in the definition of the effective dose with respect to the group of remainder organs (1,8,9). The difficulties arising from the definition of the remainder (8,9) may be emphasized if the dose distribution is strongly non-uniform. For example, in CT examinations of the head, the dose to the brain is by far the highest organ dose. The brain is one of the remainder organs, and the interpretation of the definition of the remainder has a strong influence on the effective dose and risk assessment. The goals in the risk assessment of specific groups differ in some respects from the goals in defining the effective dose; it may be appropriate to modify the rules defining the remainder according to the purpose, rather than to apply them literally.

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IRPA9
1996 International Congress on Radiation Protection
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FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE In-Utero exposure to plutonium: distribution and carcinogenesis studies.

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MAJOR SCIENTIFIC TOPIC NUMBER 2.3. (see page 7)

The Black Committee report on the raised incidence of childhood cancers in West Cumbria revealed the paucity of data on the effects and sensitivity of the mammalian embryo/foetus to high LET radiation exposure in-utero.

Uniquely in this study, mice were exposed at low activities to isotopes of plutonium chronically throughout pregnancy to model possible environmental contamination. Animals exposed to ^{241}Pu-citrate were used to investigate the uptake and distribution of the radionuclide. These biokinetic data were used to estimate radiation doses in a life-span study of the offspring of 255 female mice exposed to (0-89 kBq.kg^{-1}) ^{239}Pu-citrate. A range of activities known to result in leukaemia in adult mice. Distribution studies showed that only 0.1% of infused activity was incorporated into the pups at birth. Autoradiographs prepared of pups in-utero showed localisation of activity in the foetal yolk sac, placenta, foetal liver and skeleton. Activity and concentration increased in the foetal liver, the major site of embryonic haemopoiesis, throughout gestation despite increasing organ mass. Doses were estimated to the foetal liver and bone in-utero and for lifespan.

No significant differences due to plutonium exposure were observed in numbers of animals born or litter survival. Full independent (blind) pathological analysis was undertaken on dead animals and those killed when moribund. Kaplan Meier analysis showed no significant differences in lifespan. Analysis was also undertaken by tumour type and significant differences were not observed. One case of myeloid leukaemia was observed which is not significantly different to previously reported control incidences.

The major findings of this work are:- i) no increased cancer incidence as a result of alpha particle exposure at low levels throughout gestation was observed, ii) chronic exposure results in tenfold less uptake than single injection studies, iii) the contribution of the lactation pathway to neonatal uptake was higher than in previous studies, and iv) no mass effect, as previously reported by others, was seen. The significance of these results in the context of environmental plutonium contamination is discussed.

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PAPER TITLE The application of Monte Carlo and Magnetic Resonance Image phantoms in establishing calibration factors for in vivo measurement systems

AUTHOR(S) NAME(S) D.P. Hickman, J. Clark, D.A. Kruchten, M. Firpo

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MAJOR SCIENTIFIC TOPIC NUMBER 3.1. (see page 7)

ABSTRACT (See instructions overleaf)

At the IRPA8 meeting in 1992, Lawrence Livermore National Laboratory (LLNL) presented a new method for obtaining calibration factors for radiation measurement systems used to measure internally deposited radionuclides *in vivo*. This method utilizes Magnetic Resonance Imaging (MRI) to determine the anatomical makeup of an individual. Monte Carlo methods are then employed to simulate the transport of radiation throughout the phantom. A supplemental interface code entitled MRIPP (Magnetic Resonance Image Photon Phantom) has been developed and tested to determine applicability for generating efficiency factors for in vivo measurement systems. MRIPP assists in modeling the detection equipment of the *in vivo* measurement system into the Monte Carlo code, allows for the modeling of several different phantoms and phantom/source configurations, and provide calibration factors. MRIPP provides a useful tool for the calibration of in vivo measurement systems, enhances evaluation of different sized phantoms and detection configurations, and reduces the cost involved in procuring surrogate human structures. This presentation will demonstrate the use of MRIPP and provide results demonstrating the degree of accuracy that can be achieved using MRIPP.

This work was performed under the auspices of the U.S. Department of Energy by Lawrence Livermore National Laboratory under Contract No. W-7405-Eng-48.

INVESTIGATIONS ON FREQUENCY DISTRIBUTION OF WHOLE-BODY RETENTION PARAMETERS OF CAESIUM IN HUMANS

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INTRODUCTION

The fission product caesium radionuclides, ^{137}Cs and ^{134}Cs, play an important rule in internal dose received by members of the population as the consequence of reactor accidents or nuclear weapon tests. Better knowledge on the biokinetic parameters of these radionuclides leads to more realistic dose estimate. Since these parameters, apart from their age and sex dependence have also individual variability, it is worth to investigate quantitatively and characterize statistically these variations.

METHOD

In this study the individual variability of the parameters of the systemic retention function were investigated which has the following form according to the ICRP Publ. 30 and 56 recommendations:

$$R(t) = a \exp [- 0.693\ t/T_1] + (1-a) \exp [- 0.693\ t/T_2]$$

where the coefficient a and the half lives T_1 and T_2 are independent parameters to be investigated.

The study was based on whole-body potassium measurements assuming the validity of close correlation between potassium content and the above given retention parameters of caesium according to a predicting model suggested by R.W. Leggett [1]. The basic investigated data set was obtained by measurements of the whole-body counter of our institute. Altogether the whole-body counting results of 286 adult male and 84 adult female subjects were investigated together and separately in two age groups. namely from 20 to 40 and above 40 years of age. The frequency distribution of the three independent parameters of the given two-exponential function was studied by fitting normal and log-normal distributions. According to the goodness-of-fit in most cases slightly better fit were obtained when log-normal distribution was assumed, however the differences were not significant.

RESULTS

The most characteristic results of these statistical investigations obtained for the three parameters as mean values and for their standard deviations assuming normal distribution together with the ICRP recommended parameter values are shown in the following table:

Table 1. Caesium retention parameters and their found uncertainties

Sex	Age-group [year]	a	σ_a [%]	T_1 [day]	σ_{T1} [%]	T_2 [day]	σ_{T2} [%]
m	20-40	0.091	25	1.5	29	113	12
m	40-	0.104	25	1.7	29	106	12
f	20-40	0.160	24	4.8	15	102	16
f	40-	0.160	18	4.8	12	101	13
ICRP	adults	0.1	-	2	-	110	-

As it is seen the calculated mean values of the three parameters are quite close to the ICRP recommended values and their standard deviations varied from 12% to 25% depending on the investigated age and sex groups.

The investigated data sets represented the final results of whole-body counter measurements which contained the spread of data of different origin. To consider the contribution due to the applied scanning WBC technique (reproducibility, geometry, counting statistics, body size, etc.) and of the short term variation of whole-body potassium to the total frequency distribution pattern, the data obtained by repeated measurements on selected individuals were also investigated. The measurements of 5 male and 10 female subjects have been repeated 15-27 times within a relatively short time period in which no significant age and weight variation could be expected. The frequency distributions were investigated and spread of retention parameters was characterized by their standard deviations. The averaged standard deviations are shown in the next table:

Table 2. Averaged standard deviations of empirical distribution of retention parameters determined by repeated measurements of single individuals

Sex	σ_a [%]	σ_{T1} [%]	σ_{T2} [%]
m	13	14	6
f	13	8	9

These figures are characteristic to our measuring conditions (scanning bed technique, least squares spectrum resolution, etc.), while another whole-body counting method can provide another feature of these data. For instance the data measured by a four detector stretcher arrangement on selected individuals in Karlsruhe show about half of the values of the above given Table 2 [2]. However considering even our data this source of error is reducing the parameter total uncertainties (see Table 1) by 2-5% only. On this reason due to the biological

variability the data given in Table 3 are more realistic figures for adults disregarding age dependence:

Table 3. Derived uncertainties of retention parameters attributed to biological variability

Sex	σ_a [%]	σ_{T1} [%]	σ_{T2} [%]
m	21	25	10
f	16	11	12

For comparison we investigated data sets measured on adult subjects from the territory of the previous Soviet Union measured by the WBC laboratory of GSF Frankfurt using single detector chair technique [3]. The standard deviations of the caesium retention parameters derived by the given method are shown in the next table.

Table 4. Uncertainties of retention parameters determined in Frankfurt on adults from the former Soviet Union

Sex	σ_a [%]	σ_{T1} [%]	σ_{T2} [%]
m	21	23	12
f	21	13	16

The figures in Table 4 are quite close to the values obtained by using our measurements, which confirm the validity of the outcome of our investigations. It is to be mentioned that G. Schwarz and D.E. Dunning found much larger values for these distribution parameters that is 40, 32 and 27 percent for σ_a, σ_{T1} and σ_{T2} respectively [4].

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European Intercomparison of *in vivo* Monitoring Systems

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1. INTRODUCTION

With the implementation of the Internal Market workers are guaranteed freedom of movement within the European Union. With regard to the practical application of the Council Directives on the Basic Safety Standards [1] and on the operational protection of outside workers exposed to ionizing radiation as a result of their work in controlled areas, the so-called 'Outside Workers' Directive [2] lays down that internal doses shall be included in the Union's radiation monitoring system (Art. 4, annex I, part III). Therefore it is especially important to ensure a common quality standard in performing measurements to attain consistent results and facilitate effective interpretation. Furthermore, article 25 of the Basic Safety Standard Directive requires that measurement instruments for radiation protection are regularly checked and tested to ensure their effectiveness and correct use. One of the established methods for the fulfillment of this requirement is by intercomparison, normally performed at a national level. International intercomparisons provide the opportunity to harmonize radiation protection measures at a wider level. The European Commission has therefore decided to fund an Europeanwide Intercomparison of *in vivo* Monitoring Systems which is organized and carried out by the Institut für Strahlenhygiene (part of the German Bundesamt für Strahlenschutz, BfS). Taking part in this intercomparison are 44 institutions in 19 states (the 15 Member States of the European Union and Hungary, Czech Republic, Switzerland and Norway). In addition to the purely technical part of the intercomparison, the methods and underlying assumptions of deriving dose estimates from the measured activities are also examined. The institutions are asked to calculate doses on the basis of given (but not complete) information on fictitious *in vivo* measurements and state the assumptions and equations used. To complete the overview, the various regulations regarding incorporation monitoring are also reported.

2. OBJECTIVES OF THE INTERCOMPARISON

The study seeks to compare aspects of *in vivo* dose evaluation between the nominated representatives of the 15 European Union member states and 4 non-EU countries.

The primary objective of this study is to compare the performance of European whole body counters under normal working conditions. To carry out this intercomparison, each participating institute measures a phantom filled with unknown, mixed radionuclides. Paramount to the success of the study is that the normal measurement routine for humans is used at all times, in particular that the normal measuring time is not exceeded. A representative of the BfS is therefore travelling with the phantom to supervise the measurement routine and to ensure that the phantom is assembled in the correct way. In addition to the whole body measurement, the following aspects of dose evaluation are considered in this study: (i) the measurement of I-125 and/or I-131 in a thyroid phantom (optional), (ii) the calculation of committed dose for cases where either full or partial work history is known (case studies), and (iii) the legal mechanisms in each country which regulate detection and measurement of incorporated radionuclides in the workplace.

3. THE PHANTOMS USED FOR THE INTERCOMPARISON

3.1 WHOLE BODY PHANTOM

The Whole Body Phantom used for this intercomparison was developed by the 'Research Institute for Industrial and Sea Hygiene' and produced by the 'Scientific and Technical Centre -Protection Ltd.-' both in St Petersburg, Russia. The St Petersburg phantoms are made from up to 120 small, interchangable, plastic bricks, held together with hollow aluminium connection pieces. The plastic bricks are supplied in two sizes representing 1 kg and 0.5 kg of body mass. Four holes are drilled through the length of each brick so that rods containing known amounts of user defined radionuclides may be inserted (see Fig. 1).

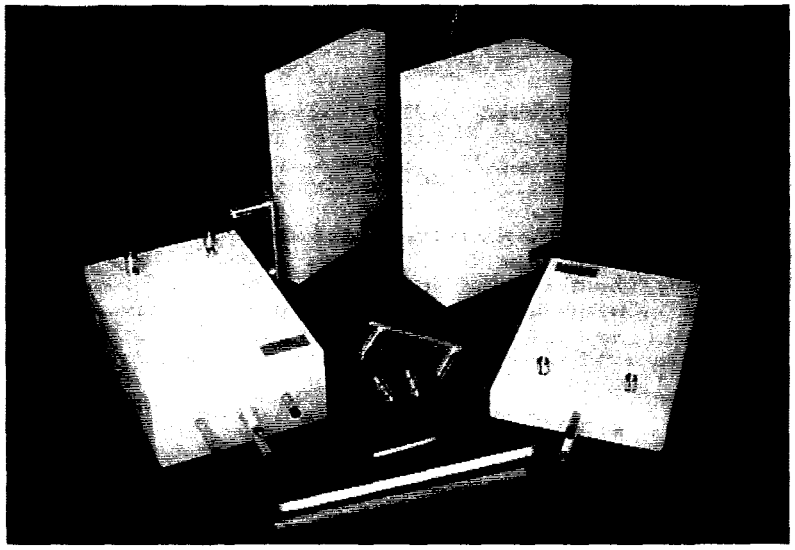
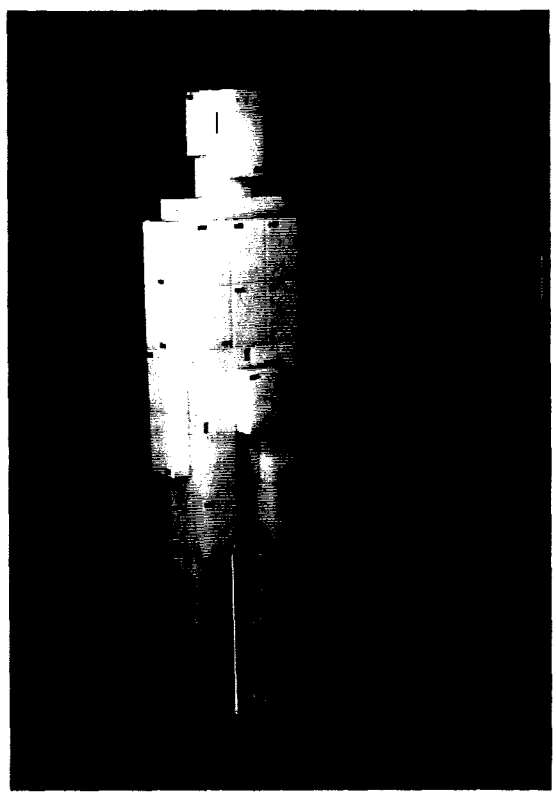


Fig. 1 1 kg and 0.5 kg bricks showing rods and connecting pieces

The St Petersburg Phantom can be assembled into representations (in terms of scattering) of 12 kg, 24 kg, 50 kg, 70 kg, 90kg and 110 kg persons (P1, P2, P3, P4, P5 and P6 respectively) in either stretch, chair or bending geometry. The stretch geometry may also be used in a "standing" position, when supported by a custom built aluminium stand. For the ongoing intercomparison, the St Petersburg phantom P4 is being used, representing the 70 kg reference man (Figure 2).

Fig. 2: Phantom P4 standing geometry



3.2 THYROID PHANTOM

The acrylic glass thyroid phantom which is made available to the participating institutions was developed as part of a research project on intercomparison of iodine counters in Germany funded by the German Federal Ministry of Environment, Nature Conservation and Nuclear Safety [3]. Intensive test measurements led to the conclusion that the chosen design provides an accurate representation of the thyroid gland for the purpose of measuring radio-iodine.The phantom was involved in a German intercomparison, and has been used by the majority of German iodine counters for calibration. The phantom is made of acrylic glass and incorporates 6 holes into which iodine samples may be placed. The holes are positioned in three pairs of two and each pair, representing the two lobes of the thyroid gland, is a specified distance from the surface of the phantom. For this intercomparison, the holes which are 20 mm from the surface of the phantom (smallest neck thickness) are used. The vials provided to the participating institutions are smaller in diameter than the holes in the phantom, and so acrylic glass tubes are used to maintain a uniform measurement geometry. The four holes which are not used during a measurement are filled with acrylic glass blocks.

4. RESULTS

More than half the measurements have been carried out and preliminary results show that most participating institutions measured the activities within 20% of the actual values. However, some institutions could not identify all radionuclides in the phantom.
The experimental part of the intercomparison is expected to continue until mid 1996. The analysis of the results and the evaluation of the answers to the questionnaires on legal requirements and dose assessment should be concluded by the end of 1996 and will be presented in spring 1997.

Acknowledgement

This work is supported by the European Commission, DG XI.C.1.

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April 14-19,1996
Vienna, Austria

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PAPER TITLE THE INTERNAL DOSIMETRY INTERCOMPARISON PROGRAM OF THE U.S. DEPARTMENT OF ENERGY

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MAJOR SCIENTIFIC TOPIC NUMBER .3.1. (see page 7)

ABSTRACT (See instructions overleaf)

In 1992, the U.S. Department of Energy (DOE) performed a pilot study to determine the feasibility of performing an internal dosimetry intercomparison[1]. An intercomparison was performed because a conventionally true value for an internal dose evaluation is not available. Intercomparison of results using a common set of test questions provides a measure of the consistency of internal dose calculations, and DOE facilities are encouraged to participate in internal dosimetry intercomparisons. Participants were asked to evaluate five actual exposure scenarios. Results of this study demonstrated that the relative standard deviation of the internal doses was approximately 30-50% (about 180% in one case) of the mean values. Factors contributing to the significant discrepancies were identified to include: 1) the interpretation and statistical treatment of the bioassay data; 2) the biokinetic models applied; and 3) the computational tools used. This study pointed to the need for further intercomparisons to fully evaluate the problem of internal dose assessment consistency. A second intercomparison study is currently underway, and it is planned that similar intercomparisons will be performed every three years. Test exposure scenarios that have been used in the present study are directly related to the current operations taking place in DOE facilities. They cover different single and multiple intakes of plutonium, chelation treatments and other realistic exposure situations. Results of this study, along with the improvements that have been made in the intercomparison technique, will be reported.

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* Work supported by the U.S. Department of Energy under Contract DE-AC06-76RL01830

DIRECT INTERNAL DOSIMETRY - A NEW WAY FOR ROUTINE INCORPORATION MONITORING OF γ-EMITTING RADIONUCLIDES

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INTRUDUCTION

The conventional procedures for *in vivo* monitoring of γ-emitting radionuclides involve the determination of body or organ burdens using whole body counting techniques and the subsequent estimation of intake and committed dose equivalent, respectively. For these procedures many information are required, such as time and pathway of intake, physical and chemical form of incorporated materials, metabolism etc. At routine monitoring these information are hardly available, thus resulting in significant uncertainties for the estimation of intake and committed dose equivalent. Besides, it is extremely difficult or even impossible to define general criteria such as lower limit of detection or confidence interval for intake and committed dose equivalent, respectively. Thus it is problematic to define minimum requirements for incorporation monitoring devices or to compare different monitoring procedures with respect to the quality of their dose estimates. In order to meet these difficulties, at Karlsruhe Research Center a new method for direct internal dosimetry has recently been developed.

MEASURING PRINCIPLE

The method refers especially to those radionuclides which commonly are detected with standard whole body counters, i.e. radionuclides emitting γ-rays with relative high abundancy (>10 %) and relative high energy (>100 keV) If such a radionuclide is deposited in some organ or region of the body, there is a well defined correlation between the photon flux at particular points of the body surface and the dose equivalent rate due to the incorporated radionuclide. This correlation may be used for direct dose assessment with an adequately designed detector system according to the following general equation:

$$H'_{eff}(S) = C(S) \cdot \sum_{i=1}^{n} [\alpha_i \cdot R_i(S)] \qquad \text{with} \qquad C(S) = \frac{\sum_{T} [w_T \cdot SEE(T,S)]}{\sum_{i=1}^{n} [\alpha_i \cdot \varepsilon_i(S)]}$$

$H'_{eff}(S)$	effective dose equivalent rate due to a given radionuclide deposition in the source organ S
$C(S)$	calibration factor of the detector system consisting of n detectors at well defined measuring points for the same deposition in S
$R_i(S)$	response of the detector i for the deposition in S
α_i	weighing factor for the detector i ($\sum\alpha_i=1$)
$SEE(T,S)$	specific effective energy for the source organ S and the target organ T according to ICRP 30
w_T	weighing factor for T according to ICRP 30
$\varepsilon_i(S)$	counting efficiency of detector i for the deposition in S

In general the calibration factor C(S) defined by the above equation depends both on the radiation emitted by the radionuclide and on the pattern of the deposition in the body. This dependence, however, can be minimized by adequate optimization of the detector system, i.e. optimization of type, number, arrangement, size and lateral shielding of the detectors, material and thickness of radiation entrance windows, electronic settings of amplification and discriminator levels etc.

„INDOS" DETECTOR SYSTEM

The INDOS detector system has been developed both for routine incorporation monitoring and for special monitoring at incidents or accidents. Thus, the following practical aspects were taken into account:

- The measurement should be performed with a simple counting detector system without spectrometry.
- The measurement should be performed automatically without the need of any trained staff.
- The measuring time should be very short (≤ 20s), thus allowing for monitoring a large number of persons with a high frequency (1 measurement/week).

The detector system which meets these conditions is shown schematically in Fig. 1. The main components of the system are four plastic scintillation detectors, being positioned in front of the thyroid (detector 1: 6.5 x 6.5 x 10 cm^3), in front of the respiratory tract (detector 2: 16 x 16 x 10 cm^3), over the thighs (detector 3: 20 x 20 x 10 cm^3) and under the gastro-intestinal tract (detector 4: 20 x 20 x 10 cm^3). The detectors are operated by standard electronics consisting of a high voltage power supply. a preamplifier and a main amplifier with a single channel analyzer. The detector pulses are fed into four counting channels of a PC. The PC is connected to a chip card unit for input of personal data (i.e. personal identification, body weight, body height, chest circumference) and for output of measuring results. For individual adjustment of the measuring geometry the seat of the detector system can be moved by computer controlled stepping motors in the horizontal and vertical direction according to the subjekct's body proportions (chest circumference for horizontal movement and body height for vertical movement, respectively). The individual adjustment allows for measurement of all subjects with body heights ranging from 150 cm up to 200 cm.

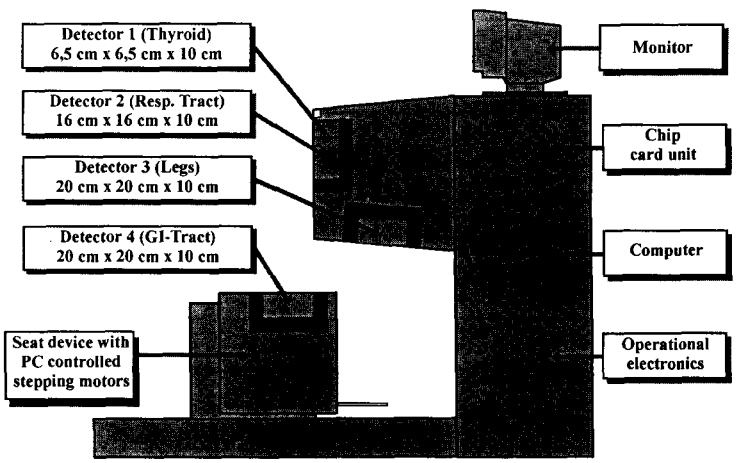

Fig. 1:
The INDOS
detector system

In order to use the detector system for dose estimations in any meausuring geometry, a mathematical calibration procedure was applied. This procedure is based on a set of semi-emprical formulas for the calculation of the detector response as a function of the photon energy of the source, the coordinates of the source position and of the thickness of the tissue between the source and the detectors. The measurements in general are performed using calibration factors for the reference man having a body weight of 70 kg and a body height of 170 cm. Since the SEE values of ICRP Puplication 30 also refer to the reference man, this procedure will yield realistic results for all persons having simaler body propertions. Moreover, this procedure will also yield very good results for persons with different body proportions, because the calibration factors and the SEE values show a very simaler dependence on the body size. Thus, this method of internal dosimetry to some extend has a built-in correction of the body size effects. This is an important advantage of the direct method.

The background counting rate is governed by radiation emitted from the natural radionuclides, i.e. K-40 contained in the materials of the environment of the detector system. The contribution of the natural airborne activity is relatively small and so the background of the detector system is very stable when counting is done susequently without subject in the same environment. When counting with a subject, however, the background of all detectors is reduced significantly due to the absorption of the environmental radiation in the

body. The absorption effect has been studied on the basis of more than 300 measurements with about 40 male and 20 female subjects with different body proportions. These measurements show the backgrount counting rate to be in very good approximation a linear function of the body weight (detector 1 and 4), the chest circumference (detector 2) and the body height (detector 3), respectively. The mean deviation of the measured background counting rates from the respective fit values are ±26 cts/20s (detector 1), ±115 cts/20s (detector 2), ±142 cts/20s (detector 3) and ±140 cts/20s (detector 4), these values being less than twice the standard deviation due to counting statistics.

EVALUATION

For evaluation of the measurement, first the individual background counting rates of the four detectors are calculated on the basis of the biometric parameters of the subject. These hypothetical background values are subtracted from the measured counting rates. If the resulting net counting rate of any detector is significant larger than the respective standard deviation (>3.29σ), the net counting rates of all four detectors are analyzed with a special logic algorithm in order to select one of the four deposition cases listed in the first column of Tab. 1. As can be seen from Tab. 1, the measured dose qanatities are the effective dose equivalent in the first two deposition cases, the lung dose equivalent in the third deposition case and the thyroid dose equivalent in the fourth deposition case, respectively. The calibration factors and the values for the lower limit of detection refer to Co-60 in the first three cases and I-131 in the last case. For unknown mixtures of radionuclides the choice of these standard calibration factors result in a mean calibration error of about 50 %. Otherwise the calibration error amounts to about 20 %.

Tab. 1:
Parameters for evaluation of the measurement and corresponding values of the lower limit of detection for one 20 s measurement (95 % confidence level)

Deposition case	Measured dose quantity	Weighing factors ($/\alpha_1/\alpha_2/\alpha_3/\alpha_4$)	Calibration factor (μSv/d per cps)	Lower limit of det. (μSv/week)
Homogenious whole body deposition	Effective dose equivalent	0/0.34/0.33/0.33	0.018	1.6
Inhomogenious trunk deposition	Effective dose equivalent	0/0.14/0.36/.50	0.062	6.3
Dose relevant lung deposition	Lung dose equivalent	0/1/0/0	0.72	9.6
Dose relevant thyroid deposition	Thyroid dose equivalent	1/0/0/0	4.7	140

CONCLUSIONS

The INDOS detector system offers the following advantages with respect to routine incorporation monitoring:

- The measurement is performed automatically and there is no need for trained staff.
- The measuring time is short und thus a relative large number of persons may be monitored with a relative high measuring frequency.
- First estimates of the individual effective dose equivalent rate are available immediately after the measuremt.
- The direct determination of the dose equivalent in principle is more precise than the conventional procedures for internal dosimetry, because (i) the retention of radionuclides in the body may be measured explicitely and (ii) the dependence of the dose equivalent on the body proportions is corrected implecitely.
- The measuring procedure is comparable to the external dosimetry with respect to accuracy and lower limit of detection. Thus, the results of internal and external dosimetry can be summed up in an easy and reasonable manner.
- The detector system can be installed in any building; it also can be installed as a mobile unit in a car or a container for long distance transportation by aircraft or train.
- Last but not least, the cost for monitoring with INDOS is much lower than for the conventional monitoring procedures using whole body counters.

The Canadian National Calibration Reference Centre for *In-Vivo* Monitoring and the United States Department of Energy's International *In-Vivo* Intercomparison.

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INTRODUCTION

The Canadian National Calibration Reference Centre for *In-Vivo* Monitoring (1) and the United States Department of Energy (DOE) collaborated to offer an intercomparison program to whole body counting facilities in 1993. The HML fabricated the phantom shell and Battelle Pacific Northwest Laboratory filled the shell with radioactive tissue-substitute polyurethane to simulate a uniform fission-product distribution in soft tissues.

The choice of the four-year-old phantom (2) was made for two important reasons: following the Chernobyl reactor accident monitoring many children was necessary and it follows that any whole body counter that has to be used in emergencies should have a good range of calibrations for it to be effective; costs- the smaller phantom required much less tissue-substitute to fill it and the shipping costs were also lower for this small phantom.

The phantom contained ^{137}Cs 4.487 ± 0.076 kBq, ^{88}Y 18.90 ± 0.54 kBq and ^{40}K 851 ± 43 Bq (01-May-93, 1200 PST). Performance of the participating facilities was evaluated according to ANSI N13.30 "Performance criteria for radiobioassay" (3) which states that "For testing purposes ...(bias) shall be within -0.25 to 0.50" and "The relative precision ... shall be in absolute value less than or equal to 0.4".

PARTICIPANTS

The program began with 27 facilities in the intercomparison program; however, this number grew as word of the program spread. As a result, the original 27 facilities enrolled in the program grew to 35. Each counting system was coded so some facilities had multiple codes, for a total of 43. Some participants counted the phantom on several different whole body counters. For example, one facility had seven codes assigned to it.

Time estimates for the length of the program were made assuming that it would take one week to perform measurements and two weeks for shipping/handling the phantom to the next facility. Thus, the length of the intercomparison was estimated to be 86 weeks. The program began in April 1993 and should have ended in December 1994; however, such factors as shipping delays, equipment breakdown, and customs contributed to the lengthening of the program. The program officially ended on 1-Aug-95, when the phantom arrived back at the HML from the last participant's laboratory.

RESULTS

The phantom, designated as P4C (Phantom 4 year old, item C), was filled with soft-tissue substitute material and an unknown quantity of multiple radionuclides. The activity in the phantom was homogeneously distributed throughout all sections and was proportional to the volume of that section. The phantom also contained ^{40}K homogeneously distributed in an amount similar to a Reference Child, to produce an accurate Compton background in the resulting spectra.

Each facility was asked to determine the identity and amount of the radionuclide(s) in the energy range 200 - 2000 keV. Each facility was asked to make an estimate of the "worst case" precision (WCP) (4) and estimate their Minimum Detectable Activity (MDA).

DISCUSSION

All facilities correctly identified the radionuclides in the phantom although some facilities did not report the activity for ^{88}Y as they had no calibration factors for this radionuclide. Most did not report an activity value for ^{40}K. Selected results are shown in and the Figs. 1 - 3. The complete set of data can be found elsewhere (4). The average shipping time was 6.3 days and the average time at a facility was 16.3 days; however, this time is not realistic due to the downtime of the counting system at one facility. If the 187-day layover at one facility is

replaced with the actual time taken to do the measurements (seven days), the average facility time becomes 11.3 days. Total shipping plus facility time was 17.6 days, which is less than the assumed three weeks used for planning purposes. The 21-day time-frame will be used for the next intercomparison scheduled for early 1996.

Bias results: Fig. 1 shows the bias results for ^{137}Cs as a function of counting geometry. Other data showed that size dependency was not a simple function of the type of counting geometry but must be a mixture of factors such as distance of the phantom from the detector(s), scan length (where applicable), size of detector, calibration coefficients etc. A similar analysis was performed on the ^{88}Y results (not shown) and the same trends were seen.

It is interesting to note that most of the facilities overestimated the activity in the phantom; however, when size corrected calibration factors were applied the results (not shown) become much more normal for ^{137}Cs. Most facilities underestimated the ^{88}Y when using size corrected calibration factors. The data in the final report (4) will be useful for facilities that wish to redefine their calibration factors.

Fig. 1 shows that no one type of counting geometry appeared better than another. Representatives of all counting geometry types fell within the N13.30 acceptable limits mentioned above.

Precision results: Fig 2 shows the WCP results for ^{137}Cs. Normal precision values for these facilities will be much less than the values reports here. All facilities fall within the N13.30 guidelines for precision. Similar to the bias results, there appeared to be no one counting geometry that was superior to another.

MDA results: Figure 3 shows the MDA as a function of counting geometry. As the photon energy rises both the counting efficiency and the background count rate drop. Therefore, one would expect the ^{88}Y MDA's (not shown) to be lower than the ^{137}Cs values. The data showed this to be true for most facilities; however, for a few facilities the reverse was true, and the ^{88}Y MDA was higher than the ^{137}Cs value. As before, there was no correlation with counting geometry. One would have expected to see a correlation between ^{137}Cs MDA and counting time; however, the data showed that this was not true. Although there was a general drop in MDA as counting time was increased there was much scatter in the data. The data clearly showed that there was a large difference between ambient background count rates and/or detector shielding at the participating facilities (4).

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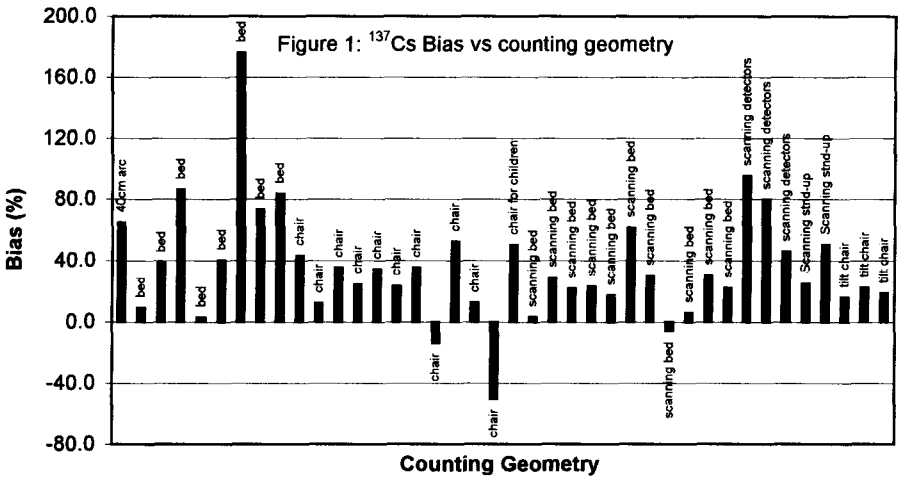
Figure 1: 137Cs Bias vs counting geometry
Bias (%)
200.0
160.0
120.0
80.0
40.0
0.0
-40.0
-80.0
40cm arc
bed
chair
chair for children
scanning bed
scanning detectors
Scanning stnd-up
static bed
tilt chair
Counting Geometry

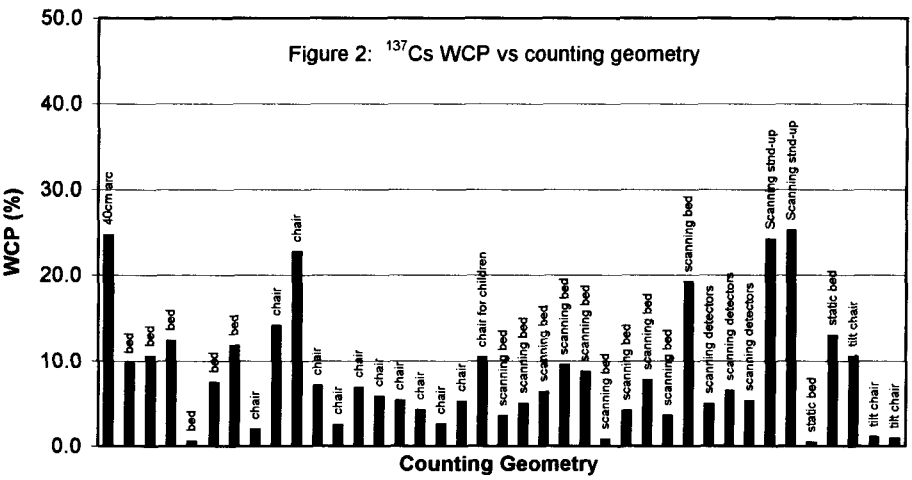
Figure 2: 137Cs WCP vs counting geometry
WCP (%)
50.0
40.0
30.0
20.0
10.0
0.0
40cm arc
bed
chair
chair for children
scanning bed
scanning detectors
Scanning stnd-up
static bed
tilt chair
Counting Geometry

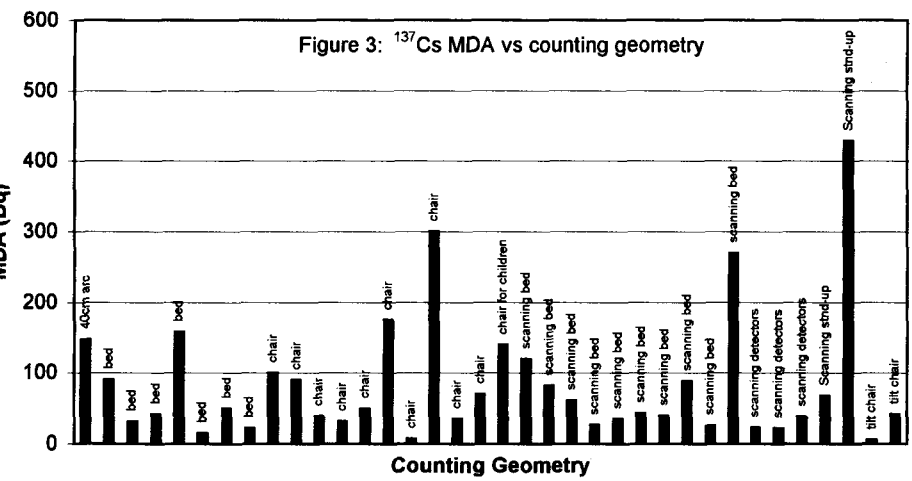
Figure 3: 137Cs MDA vs counting geometry
MDA (Bq)
600
500
400
300
200
100
0
40cm arc
bed
chair
chair for children
scanning bed
scanning detectors
Scanning stnd-up
tilt chair
Counting Geometry

THE NEW WHOLE-BODY COUNTER AT THE NRIRR, BUDAPEST

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P.O.B. 101

INTRODUCTION

A CANBERRA Model 2270 whole-body counter equipped with NaI(Tl) detectors in chair geometry operates at the NRIRR since 1981. This counter is suitable mainly for screening of internal contamination in body and thyroid of occupational origin because of the relatively small size of detectors and the poor shielding (1). However the counter was also used to determine the ^{137}Cs body burden of population living in the region of Budapest following the Chernobyl accident (2). The experiences gained from this study and preparations for the investigation of contamination due to accidental releases from a NPS made clear, that beside the improvement of shielding a new whole-body counter should be equipped with high resolution HpGe semiconductor detectors too. The new shielding was made from steel of thickness of 15 cm with the inner sizes of 1.6, 2.0 and 2.0 meters allowing the operation in scanning geometry too, with moving detectors under and over the bed.

METHODS

The requirements of moving and positioning of detectors, i.e. quasi continuos controlled linear driving, manual setting of vertical and rectilinear position could have been satisfied most simply by the use of step motor and chain driving of the detector holder stand. The detection limit of the measurements can be reduced by the summing up of the spectra taken by the detectors moving under and over the bed simultaneously. However, to obtain good result the complete co-operation of the two detector systems, conformity of amplification and linearity are essential.

The calibration of the new whole-body counter was performed by homogeneous phantoms built from 1 L plastic bottles for the gamma-energies from 22 up to 1836 keV and for body masses of 15, 30, 45, 60, 75 and 90 kg for both standing above the chest and scanning detectors. Neck phantoms made from different materials (plastic and wood) and having different absorbent layer thickness (10-19 mm) were used for the calibration of thyroid geometry.

RESULTS

The efficiencies of the two-detector system for different gamma-energies and for the most common scanning geometries can be seen in Fig. 1. The highest efficiency was observed at 122 keV energy, while the efficiencies at very low and high energies (22 and 1836 keV) are less by a factor of 4 to 5. The total efficiency is lower for the 90 kg, 35 cm (bed-detector distance) geometry than the 15 kg, 25 cm one by a factor of about 3. The efficiencies for the detectors standing above the chest are higher than of the similar scanning geometries by a factor of 1.3 to 1.4, because of the shape and activity distribution of the phantoms.

The efficiencies of the thyroid geometry are illustrated in Fig. 2. The observed ratio between the highest (plastic, 10 mm absorbent layer, 1 cm neck-detector distance) and lowest efficiencies (plastic, 19 mm absorbent layer, 5 cm neck-detector distance) was higher than 3 .The results on the effect of the material and thickness of the absorbent layer and of the detector-neck surface distance on the efficiency yield valuable information for assessing the uncertainties of measurements of human thyroids as well.

The typical ranges of the detection limits derived for scanning and standing whole-body and thyroid geometries for the measuring times of 1000 and 1500 s can be seen in Table 1. The highest LLD was estimated for ^{40}K, because this natural radionuclide is present also in the background.

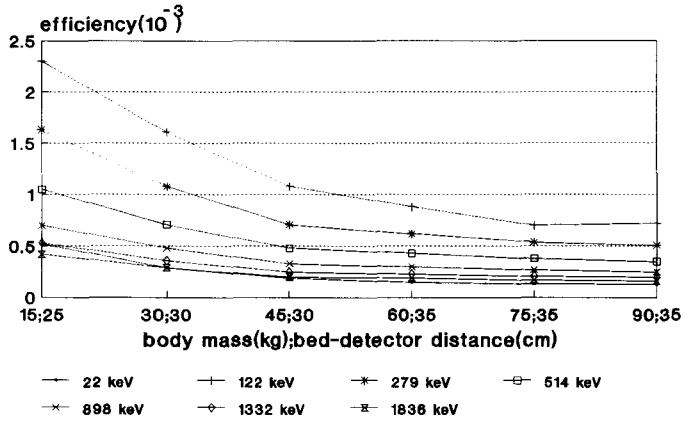

Fig 1. The counting efficiencies for different measuring arrangements (phantom masses; bed-detector distance) for the two-detector scanning geometry

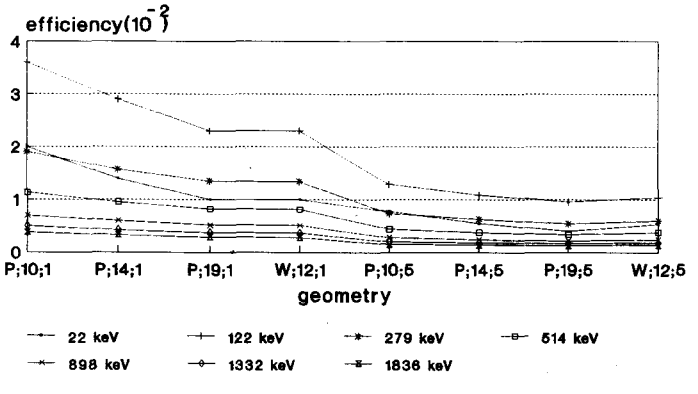

Fig 2.The counting efficiencies of different thyroid measurement geometries (phantom material; thickness;neck-detector distance) for the upper detector

Table 1. Typical ranges of LLD values in Bq for several important radionuclides for body and thyroid measurements of different geometries

geometry	^{125}I	^{131}I	^{134}Cs	^{137}Cs	^{40}K
1 det. chest	-	-	20-55	25-60	500-1200
1 det. scan.	-	-	25-70	30-80	600-1500
2 det. chest	-	-	15-35	15-45	300-800
2 det. scan.	-	-	15-45	20-50	350-950
thyroid	3-10	3-10	-	-	-

The retention of ^{134}Cs radioisotope for two radiation workers (KA 42 yr., 65 kg; SD 54 yr., 85 kg) was investigated following the ingestion of 1 kBq activity. The results of our measurements agreed well with the whole-body activities measured at the National Institute of Radiation Protection (SSI), Stockholm 7 and 10 days following the intake (Fig. 3.). The estimated components of the biological half-times were 32 and 120 days for subject KA, while in the other case a single exponential fitting was only acceptable.

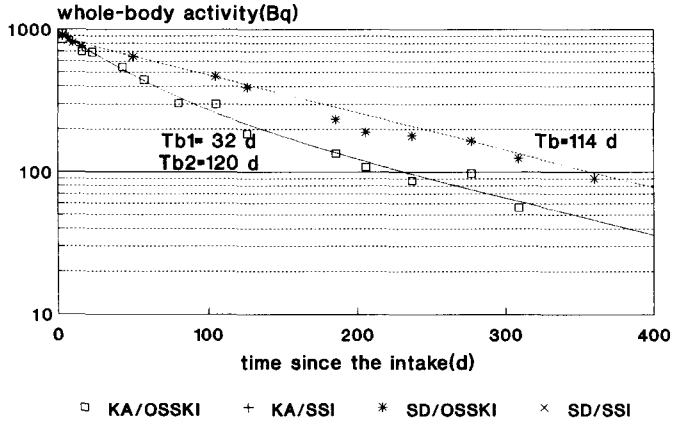

Fig 3. Whole-body retention of ^{134}Cs measured at the NRIRR and SSI (Stockholm)

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RADON INFLUENCE IN THE ESTIMATION OF BACKGROUND OF WHOLE-BODY COUNTING

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INTRODUCTION

Because of the existence of natural background radiation, it is important for the precise whole-body counting to keep the fluctuation of background radiation slightly. It is considered that radon and its daughters are one of the remarkable components of natural background radiation for its large fluctuation. Therefore, we investigated (a) the relationship of meteorological factors and radon concentration, (b) the effects of ventilation and filtration on radon concentration, and (c) the comparison between radon concentration and count rate of whole-body counter (WBC).

METHOD

1. Outline of WBC at the University of Tokyo

The WBC at the University of Tokyo is consisted of 20 cm thick iron shield room with 4 segments of 50x50x15 cm plastic scintillators and a 8inch ϕ x 4inch NaI(Tl) scintillator (1). Two ventilation systems are equipped; one is for an operation room (4.5m^3/min, air intake from top of the building, 21m height), the other is for an iron room (0.5m^3/min, air intake from the ceiling of iron room). The operation and iron rooms are situated at a semi basement floor.

2. Radon concentration measurement

Radon concentration was measured with PMT-TEL measurement system (provided by Pylon Co.) by the air-flow electrostatic sampling method. The air flow rate by the built-in pump was 2,000 cm^3/min. The radon concentration was averaged for 10 minutes or 1 hour. The conversion factor of count rate to concentration was 1.61 Bq/m^3/cpm from a reference manual of the system (2).

RESULTS AND DISCUSSIONS

1. Relationship between meteorological factors and radon concentration

The relationship between meteorological factors, which are temperature, humidity and atmospheric pressure, and radon concentration was investigated. As for atmospheric pressure, there was a significant correlation. Measured radon concentration level was low in case of higher atmospheric pressure. The regression equation was as follows:

$$RC = -0.270\ AP + 293$$

where RC is radon concentration (Bq/m^3), AP is atmospheric pressure (mb) (r^2=0.726). This tendency agrees with the previous report (3).

2. Effects of ventilation and filtration on radon concentration

Figure 1 shows the change of radon concentration in the operation room by the method of ventilation. When the ventilation was not operated, radon concentration reached about 40 Bq/m^3. In the case of air intake from the ground level, radon concentration remained at rather high level (about 20 Bq/m^3). When air was intake from the top of building with 21m

height, the radon concentration was kept at low level (about 10 Bq/m^3). Under this condition, HEPA filter could not work to remove radon effectively.

As shown in Fig. 2, the ventilation of the operation room decreased radon concentration effectively and it took about 6 hours to settle a stable condition from the beginning of ventilation. This tendency has a good reappearance. The averaged radon concentration in the operation room was 8.5 Bq/m^3 in case of well-ventilation. The calculated value was 10.4 Bq/m^3, obtained by the following method.

The flux density of radon, J_D (Bq/m^2/s), from one side of a wall (or a floor) was calculated by the equation (4) shown below,

$$J_D = C_{Ra} \lambda_{Rn} f \rho [D_e/(\lambda_{Rn} \varepsilon)]^{0.5} \tanh d [D_e/(\lambda_{Rn} \varepsilon)]^{-0.5},$$

where C_{Ra} is the activity concentration of ^{226}Ra in the building element (Bq/kg); λ_{Rn} is the decay constant of ^{222}Rn (2.1 10^{-6} /s); f is the emanation fraction, ρ is the density (kg/m^3); D_e is the effective diffusion coefficient (m^2/s); ε is the porosity, and d is the half-thickness (m). The ventilation rate is 4.5 m^3/min. The volume of an operation room is 197 m^3. Because air exchange through doors and other openings was not taken into account in this calculation, the actual amount of air exchange was larger. Therefore, the calculated value, 10.4 Bq/m^3, would be more small.

The comparison of the fluctuation of radon concentration in the iron room with/without ventilation of the operation room is shown in Fig. 3. The averaged radon concentrations per every ten minute were measured for 40 minutes. In the case of no ventilation, radon concentration was high (30 - 45 Bq/m^3) and its fluctuation was large. On the other hand, radon concentration was in low level and its variation was small. These results showed the air exchange between an operation room and an iron room could occur easily and only the ventilation of an iron room was not effective. Therefore, it is necessary and important to keep radon concentration in an operation room low by the ventilation of an operation room.

3. Comparison between radon concentration and count rate of WBC

The effect of radon concentration on WBC count rate with plastic scintillator is shown in Table 1. The count rate in the lower energy range below about 800 keV of WBC corresponded with radon concentration. The decreases of Bi-214 and Pb-214, which are radon daughters, by ventilation were observed in the spectrum measurement with a high-purity Ge semiconductor detector. On the other hand, there was no change of the count rate in the upper energy range above about 800 keV. That shows no consideration is needed in this high energy region on the discussion of radon influence.

CONCLUSION

The behavior of radon in WBC room was analyzed and the desirable condition of ventilation and filtration for whole-body counting was discussed.

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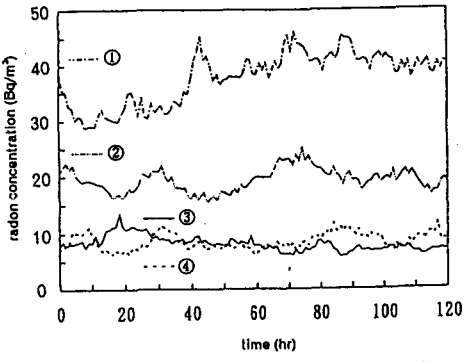


Fig. 1 Change of radon concentration by the method of ventilation; ① no ventilation, ② air intake from ground level, ③ air intake from 21 m height, ④ ③ through HEPA filter.

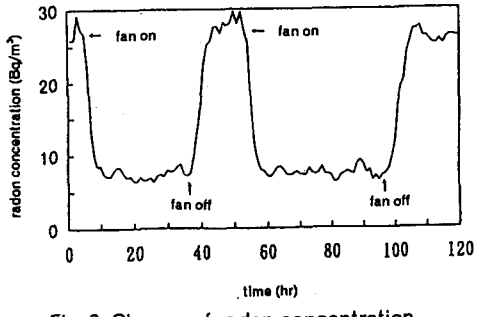
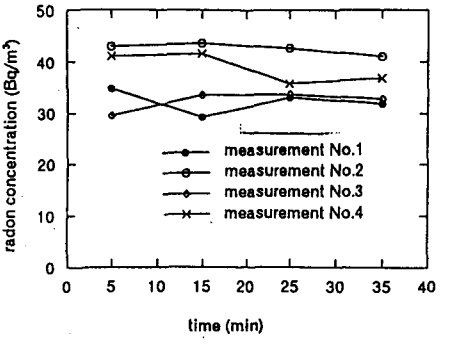
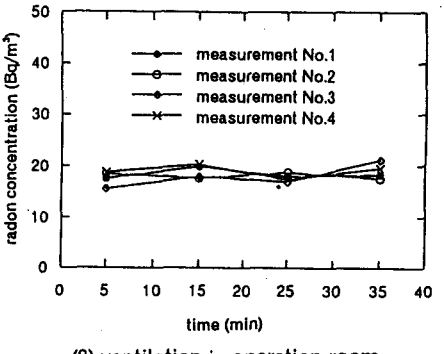


Fig. 2 Change of radon concentration with/without ventilation



(1) no ventilation in operation room



(2) ventilation in operation room

Fig. 3 Comparison of the fluctuation of radon concentration with/without ventilation

Table 1 The effect of radon concentration on WBC count rate

ventilation rate	radon concentration	WBC count rate (counts/10min)	
(m^3/min)	(Bq/m^3)	lower bin	upper bin
0	36.6 ± 4.2	39286 ± 293	11748 ± 104
4.5	18.4 ± 0.6	36894 ± 182	11140 ± 95

I.R.P.A. 9
VIENNA 14-19 April 1996

POSTER P 31-9

TREATMENT OF THE X AND γ RAYS LUNG MONITORING SPECTRA OBTAINED BY USING HP-Ge DETECTORS IN CASE OF EXPOSURES TO URANIUM

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INTRODUCTION

X and gamma whole body counting examinations make it possible, when focusing to an organ, to determine the retention of radioelement. The critical organ when performing the radiotoxicological monitoring of personnel exposed to non-transferable compounds of uranium is the lung. This is true for both natural and enriched uranium because of their biological inertia and of the very high ionising power of their alpha radiation. *In vivo,* X and gamma spectrometry take advantage of the low excretion rate of these compounds to allow the quantification of the pulmonary retention of the inhaled products. These examinations are based on the analysis of the X and gamma radiation emitted by the radioelements retained in the lung and the ensuing quantification of their activity.
HyperPure coaxial Germanium detectors (HP-Ge) are used because of detection features which are specific to uranium. In the 10 keV to 400 keV energy range, these detectors combine low electronic background, satisfactory energy resolution and a high peak/background ratio. Their detection limits depend on spectrum treatment performance and on the modelling of the morphological parameters of the rib cage. For various reasons, the estimation of radioactive contamination in the lungs of a human being poses a delicate spectrometry problem: activity levels are low, a significant proportion of the radiation emitted is absorbed by the rib cage and counting time is limited for practical reasons as well as because of the equipment background effect. Consequently, for most of the spectra, the 185 keV ^{235}U absorption peak does not clearly stand out of the average level of spurious counter pulses. It is therefore difficult to affirm whether or not a peak exists, and in case of contamination, it is, *a fortiori,* difficult to estimate the number of pulses emitted by the radioelement at that energy level during the measurement operation.
The aim of our study, which calls into question both acquisition conditions and interpretation methods, is to improve the efficiency of these measurements.

MATERIALS AND METHODS

1 - The whole body counting equipment

The system consists of four LOAX detectors movable and adjustable on a rail. The coaxial N-type low background detectors are made with high purity germanium. The energy detection range extends from 3 keV to 3 MeV and covers the isotopic energies required for medical assessment in Pierrelatte facilities. Two detectors are placed along the body axis above each lung to cover as much lung area as possible. The absolute efficiency of the measuring system is calibrated with a Livermore phantom. Two background measurements are performed every day. The background measurements are made on an empty shield, the counting time being the same as for the monitoring measurements. The calculations, which take into account the morphological parameters of the worker, are made using automatic spectral analysis, peak search and specific area. The results are analysed using the sum of the counts given by the four detectors, once the background of the room has been subtracted.

2 - The method adopted

The starting point for this study was a critical appraisal of the measurement principles. It is clear that, before looking at spectrum interpretation problems, a certain number of questions need to be asked: Do the four spectra need to be summed or should they, on the contrary, be treated separately ? Can the backgrounds really be exploited ? If so, what counting time should be allotted for them ? Should lung acquisition time be increased, or can it be reduced ?

In replying to these questions, an important consideration is to take into account all the available information: the resolution of the detector, for example, is a useful piece of information since it indicates what form the eventual peaks will have. The second stage involves the analysis of the spectra. Basing ourselves, respectively, on statistical test theory and evaluation theory, we propose a method of diagnosis and a peak quantification algorithm. A computer programme written in Turbo-Pascal has been developed in order to implement the corresponding algorithms.

3 - Modelling

A whole body counting spectrum cannot be considered to be a series of regular peaks, even if the background is totally abstracted. Adjusting a Gaussian function to a peak is not a trivial affair, because the peak is the result of random phenomena. As such, in comparison with the "ideal" Gaussian function, it may contain discrepancies of varying importance.

Peak formation modelling is a random phenomenon when we consider that the number of counts that the detector - which is submitted to a certain apparent activity over a given period - shows in each of its channels obeys a Poisson distribution in which the average is equal to the product of the following numbers:

- ⇨ the apparent activity (activity x percentage of non-absorbed photons x solid detector angle fraction)
- ⇨ the counting time
- ⇨ the fraction of the theoretical Gaussian area intercepted by the channel

Throughout this study, to simplify the calculations, we have taken the numbers of pulses received in each channel as being independent random two-by-two variables. The probability of a spectrum corresponding to a given source can be calculated, thanks to this approximation, from the probabilities of the number of counts that each channel can receive.

RESULTS

1 - The background

Physical modelling of the background has to be ruled out, because there are too many possible explanations for the spurious counts. The study will accordingly be statistical, based on a two-hour long background measurement on an empty cell. Subtracting the background does not bias the measurement. However, for low counting rates, the relative error may not be negligible.

2 - Do the treatments have to be separated ?

The argument advanced to justify the summing of the four spectra is that, for cases of low contamination, it appears to be necessary to add up the spectral information. This argument holds for peak presence searches, but it is not valid in the cases where the peak area has to be quantified. Figure 1 shows the magnitude of a spectrum corresponding to a level of activity close to the equipment detection limit.

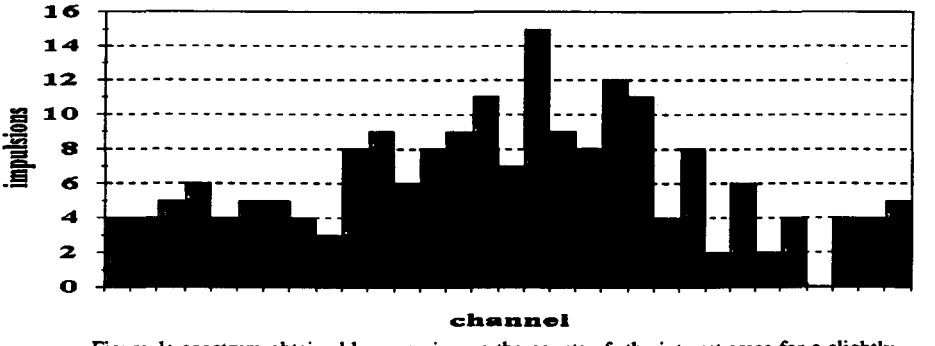

Figure 1: spectrum obtained by summing up the counts of the interest areas for a slightly contaminated worker

The peak obeys a Poisson law where the average is proportional to the resolution of the detector. Separate quantitative treatment also shows up the distribution of the contamination in the lungs.

Table 1 summarises the distribution in terms of the spectral information provided by each detector during measurement on workers whose results were higher than the detection limit. Examinations need to be studied detector by detector.

Table 1: Distribution of the areas measured on real cases in function of the detectors

detector	1	2	3	4
% of the area cover by each detector	24.2	25.4	24.2	26.2

3 - Estimating the presence of internal contamination

This analysis falls within the scope of statistical test theory. Two multiple hypotheses can be posited: the presence of spurious pulses only, or the simultaneous presence of spurious pulses and photonic uranium isotope emission. Since the numbers of pulses counted in each channel are considered to be independent variables, test theory enables us to show the existence of an unbiased, convergent Uniformly Most Powerful test, with an acceptance region corresponding to a 5 % type I error.

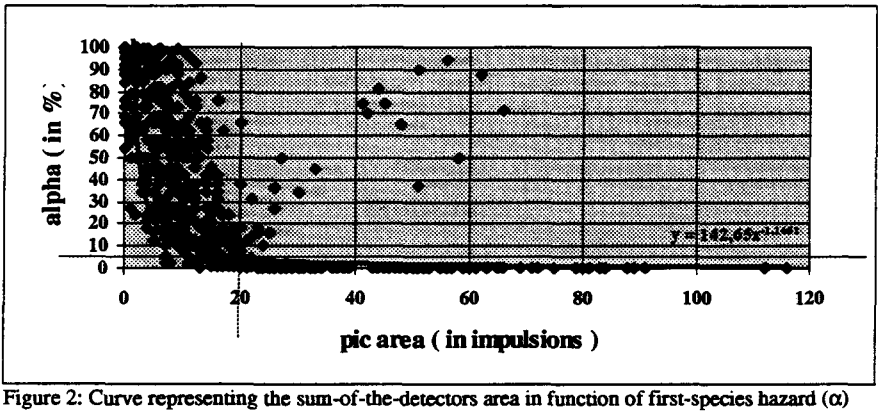

Figure 2: Curve representing the sum-of-the-detectors area in function of first-species hazard (α)

Figure 2 shows four distinct zones.
Negative results : weak area (less than 20 pulses), α between 5% and 100%
Positive results : area with more than 20 pulses, α less than 5%
Results with interferences : weak area and α less than 5%, or area with more than 20 pulses and α between 5% and 100%

CONCLUSIONS

A lung monitoring counting spectrum can be described as a random phenomenon. Channel-by-channel Poisson-type modelling was verified for cases of pure background. When carrying out spectral analysis for qualitative research, one must work with the sum of the detectors. The quantification must be calculated detector by detector. Statistical tests make it possible to certify that one or several peaks are really present in the organism.

The calculations are currently made with automatic spectral analysis, peak search, specific area, statistics and probability of the real presence of analytic photo peak taking into account the morphological parameters of the worker. The results are analysed detector by detector, with and without the background of the room. Detection limits obtained in Pierrelatte in monitoring measurement conditions were assessed for variable tissues covering the range of subjects to be examined. For each subject , the calculations are made taking into account the equivalent tissue thicknesses derived from individual morphological parameters.

This method makes it possible to quantify lung activities with a detection limit of 3.9 Bq (^{235}U; thirty minutes counting time ; reference man parameters) and to monitor exposure to the different compounds of uranium.

THE GAMMA CAMERA AS AN EMERGENCY WHOLE-BODY COUNTER FOR CONTAMINATED PERSONS

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INTRODUCTION

In case of a severe accident with release of large amounts of radionuclides into the environment there will be a great need for equipment to measure contaminated persons. Gamma cameras with large scintillation detectors are available at many hospitals. It would be of great importance if these could be used for measurements of gamma-emitting nuclides as Cs-137, Cs-134 and I-131 in the acute situation as well as in the control work which will be needed for a long period of time. The aim of this work is was to investigate the possibility of using gamma cameras in accident situations and to provide suggestions of measurement procedures. Simplicity was in this context judged to be essential.

MATERIALS AND METHODS

Two gamma cameras have been used. Camera A is an older camera with a 400 mm diameter 9 mm thick NaI(Tl) detector with the possibility to easily connect an external multi channel analyzer (MCA). Energy ranges can thereby be set even for energies above that of Cs-137 (662 keV). With the addition of an extra counterbalance the camera could be handled freely with and without a collimator. Camera B was more modern with a 400 mm diameter 6 mm thick detector and limited possibilities of setting energy ranges. No extra counterbalance could be used so the camera had to remain fixed when used without a collimator.

All calibrations were made with phantoms constructed from various numbers of 300 ml freezing sachets filled with known water solutions of Cs-137, Cs-134, I-131 and potassium. Tap water was used for background meaurements. The sachets were arranged in the form of rectangular boxes of various sizes, as well as three body-shaped phantoms simulating a four year old child (14 kg) as well as two adults of different sizes (60 kg and 90 kg).

At measurements the phantom was placed on a bed with a flat surface. The distance phantom surface - detector was 5 cm, 20 cm, and 35 cm respectively. At measurements with the body shaped phantoms the camera head was placed in two fixed positions one centered over the throat (thyroid) and one over the center of the body. At measurments with the box phantoms the camera head was fixed over the geometrical center of the phantoms.

Camera A was used with High Energy General Purpose (HEGP) collimator, Low Energy General Purpose (LEGP) collimator and without a collimator. The energy ranges investigated were 50 - 450 keV, 450 - 550 keV, 550 keV - 750 keV. Camera B was only used without a collimator and the investigated energy ranges were 50 - 450 keV and 450 - 511keV, the maximum possible energy to be pulse height analyzed for that camera.

Minimum detectable activities (MDAs) were calculated as three standard deviations of the background counts for 1 minute measuring time. In sensitivity values, cps/Bq, cps refer to the net counts from the present nuclide when background counts, from measurement with background fantom in the same energy interval, is subtracted. All MDA values, in Bq, and all sensitivity values, in cps/Bq, refer to total activity in the phantoms.

RESULTS

The highest sensitivity is achieved in the large low energy range, 50 - 450 keV, with camera A without collimator. In all three energy ranges the sensitivity decreases with increasing distance phantom surface - detector. The sensitivity for the Cs-137 phantom at constant distance phantom surface - detector decreases with increasing length, width and thickness of the phantom. The background increases with increasing distance phantom surface - detector and decreases with increasing length, width and thickness of the phantom. The variations of sensitivity and background with phantom size is larg at 5 cm distance phantom surface - detector. At 35 cm distance the variations are much smaller and for "adult" sizes of phantoms,

especially in the low energy range, almost constant. MDA values for adult phantoms are in the low energy range 400 Bq (5 cm distance) to 1000 Bq (35 cm distance). The MDA values in highest energy range, covering the Cs-137 full energy peak, are 1,3 - 1,8 times higher than in the lowest energy range. In the intermediate energy range the MDAs are higher by a factor of 3 - 4 than in the low energy range.

With LEGP collimator variations of sensitivity and background with distance phantom surface - detector, and with phantom size, are similar but more pronounced than without collimator. The level of the background values are lower, but the sensitivity values are also lower, which gives higher MDA values by a factor of 3 - 4. Values of MDA with use of the HEGP collimator are approximately 10 times higher than without collimator.

MDA values of I-131 in thyroid at measurement over the phantom throat at 5 - 35 cm distance are in the low energy interval 100 Bq - 450 Bq without collimator, 800 Bq - 2500 Bq with LEGP collimator, and 12000 Bq - 17000 Bq with HEGP collimator.

All calibration measurements of potassium show that K-40 is not detectable (refering to three standard deviations of the background counts).

The sensitivity for camera B is in the low energy interval about 20 - 40% lower for both I-131 and Cs-134,137. In the intermediate energy range 450 keV - 511 keV, compared to 450keV - 550 keV for Camera A, the sensitivity is reduced by a factor 2 to 5.

Table 1. Sensitivity, background and MDA values (1 min. measuring time) for camera A at 35 cm height above adult phantom Whole-body Cs-137 measured over the center of the phantom; I-131 in thyroid measured above thyroid. Comparison with standard measurements at the whole-body counter (WBC) in Göteborg (one - two 12,5 x 10 cm NaI(Tl) detectors).

Energy range (keV)	Collimator	Sensitivity, Cs-137 [cps/Bq]	Sensitivity I-131 in thyroid [cps/Bq]	Background countrate [cps]	MDA Cs-137 (Bq)	MDA I-131 (Bq)
50 - 450	No	2.1E-02	4.7E-02	3200	1000	450
450 - 550		8.5E-04	2.0E-04	80	4100	17000
550 - 750		2.2E-03	4.7E-04	110	1800	8500
50 - 450	LEGP	2.4E-03	3.4E-03	460	3500	2500
450 - 550		2.3E-04	8.4E-05	35	10000	27000
550 - 750		6.4E-04	1.6E-04	65	4900	20000
50 - 450	HEGP	1.5E-04	4.1E-04	330	24000	17000
550 - 750		8.0E-05	5.4E-04	40	16000	4500
WBC Göteborg (1,2)		1.2E-03	2.2E-02	0.5 (Cs-137) 0.4 (I-131)	230	3

DISCUSSION

The most important parameter is the background. For the normal standard gamma camera, the background constitutes the major part of the pulses in the camera, without or with a contaminated person. This makes a stable and known background extremely important. A "normal" change in background - due to changes in the general background or due to a person passing close to the camera making a shield for a part of the background or due to contaminated persons or items in the neighbourhood of the camera or due to a different body shape of the measured person or another distance person to camera head than taken into account - can alter the measured value with thousands of Bq. Lead shields around the camera will decrease the background and its variations. Moreover a lead shield under the camera head reduces the photon field from below, including scattered photons, and hence reduces the background (3).

A realistic goal is to keep the background within six SD of the mean (i.e. double MDA values). This requires stable background conditions and a background calibration measurement performed in the present outdoor contamination situation. If the background is kept within these limits an activity measurement at the MDA level means that the activity in the body is

somewhere between 0 Bq and an activity corresponding to two MDAs. Hence the accuracy of activity measurements increases with increasing activity in the body. However the MDA values are in most cases low when considering the doses achieved from internal contamination. For a person of 70 kg a constant body burden of 160 kBq Cs-137 gives a yearly effective dose of 5 mSv. For a child of 10 kg the corresponding figure is 24 kBq (4). An intake of 120 kBq I-131 giving 35 kBq at maximum in thyroid gives a thyroid dose of 50 mSv (5).

Suggested measurement procedure

To maximize the sensitivity of the gamma camera, a camera with a thick chrystal and without collimator is to prefer. The most sensitive energy interval is in all situations for all nuclides 50 - 450 keV or larger. To minimize the changes in background with various body sizes, a not to short, but fixed, distance body surface - chrystal should be used e.g. 30 - 40 cm. Since the background without phantom/person varies with distance of the camera head above the floor, the most stable condition is a fixed camera head above a flat bed possible to change in height. A gamma camera without a collimator is difficult to handle so a convenient height may be difficult to achieve without special arrangements. The person making the measurement should be placed at a fixed place at least a few meters away from the camera. No other persons should be allowed in the closest area.

For calibration it is most simple to use simple box phantoms made of larger plastic containers. The size of the phantom should be approximately, length 150 cm - 170 cm, width 30 cm - 40 cm, and height 15 cm - 20 cm. For adults one phantom for background and one for each radionuclide is sufficient. The background phantom could be filled with water or, rather, a mixture of water and 1,5 g - 2 g potassium/kg phantom. For children one or more phantoms sizes may be needed. For calibration of I-131 in thyroid a small container (appr. 20 ml) with I-131 placed under a few cm of water will be sufficient. It is necessary to make often repeated background measurements without a phantom to correct the background values measured with water phantom for changes in the background photon field.

CONCLUSIONS

The gamma camera can be a good and important tool for assessing the internal contamination of gamma-emmitting nuclides in an accident situation. The background is a large source of uncertainty and must be kept low, if possible, and known and stable at all times. To avoid variations in background and sensitivty with body size, which severely influence the accuracy of the measurements, the camera should be used without a collimator, a large low energy interval should be chosen and the distance persons surface - detector should be at least 30 cm. Measurements can then be performed with simple calibration procedures. In the case of low contamination levels near the theoretical MDA values, use of collimator, short distances measured persons surface - detector, or simultaneous measurements of e.g. I-131 and Cs-137,134, somewhat more extensive calibrations are necessary.

ACKNOWLEDGMENT

This investigation was supported by The Swedish Radiation Protection Institute (SSI P 55:88)

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A METHOD TO LOCALIZE SMALL ACTIVITIES IN THE HUMAN BODY BY A WHOLE-BODY COUNTER

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INTRODUCTION

In nuclear medical research and diagnosis it is often important to be able to localize small amounts of radioactivity in the human body. A complete representation of the distribution of a radioisotope in a region of the body is highly desireable but would require scans in two orthogonal planes to achieve a three-dimensional portrayal. Since most scanning devices are limited to plane portrayal we have developed an alternative technique which will be presented in this paper. Thereby we combine single longitudinale profile scans using a special detector configuration with a computer model, where we take the special geometry into account.

MATERIALS AND METHODS

Description of the experimental equipment

The detector system of the high-sensitivity whole-body counter (HWBC) used for the measurements consists of four 8" x 4" NaI(Tl)-crystals. Two of these linear scanning detectors are placed above and two below the bed and can be arranged independently in three dimensions. For an accurate measurement of the spatial distribution of the radioactivity special focussing lead collimators were designed. These multi-slit collimators can easily be brought into position in front of the four NaI(Tl)-detectors when required. This equipment is placed in a closed massive shielding chamber (total weight of the order of 74 t) to ensure a low background activity and thus a high sensitivity of the system. A central computer controls the scanning motion and the data aquisition separatly for each detector, the evaluation of energy spectra, the storage of data and the output of results.

Collimated profile scanning

To achieve spatial resolution a special detector configuration was chosen shown in Fig.1. Various point sources [^{133}Ba (355 keV), ^{131}I (364 keV), ^{22}Na (511 keV), ^{137}Cs (662 keV), ^{60}Co (1173 keV, 1333 keV)] of known activity were used to measure the counting efficiency systematically for different point source positions. Thereby each radioisotope was placed at fixed positions inside an elliptical cylindric Presdwood phantom (major axis 34 cm, minor axis 26 cm, height 50 cm). This phantom was positioned on the bed, and the profile of count rates was measured (scanning time 600 s, scanning length 190 cm) by the profile scanning system using the special measuring arrangement. The profiles thus obtained for each scanning detector expressed in cps/Bq represents the efficiencies vs. the longitudinal localization of the point source. Based on the counted efficiencies by different vertical and transversal point source positions a computational model to estimate spatial positions was developed, in which the inverse square law and the attenuation factor due to the absorbtion and

scattering processes of gamma rays in body tissue equivalent medium were taken into account. So the energy peak efficiencies for each detector can be approximated by the expression:

$$\varepsilon_D(x_p,z_p) \approx \frac{C}{r(x_p,z_p)^2} \cdot e^{-\mu(E)\cdot s(x_p,z_p)}$$

ε_D	detector efficiency [cps/Bq]
C	scaling factor [cm^2 cps/Bq]
r	distance between source and detector [cm]
μ(E)	attenuation factor [cm^{-1}]
s	thickness of the scattering medium (Preswood phantom) [cm]
x_p	transversal position of the source [cm]
z_p	vertical position of the source [cm]

This model allows to determine the unknown position of a point source inside the Presdwood phantom and its accuracy was verified for various radionuclides over a wide gamma energy range. The energy peak efficiencies of the detectors for ^{131}I (364 keV) are shown in Fig.2. as a function of vertical point source location with/without transversal displacement within the phantom. Also the geometrical and arithmetic means are indicated and it can be seen that a almost complete independence of response on the depth of the point source was obtained for the geometric mean. The variation of sensitivity with depth is largly compensated by the use of opposite detectors. Because of the attenuation, when a radioisotope in a tissue equivalent phantom is scanned, the sum of the counts from all detectors showes a minimum for the source at the mid point of the phantom. Similar results have been obtained for other gamma energies but the absolute values of the averaged energy peak efficiencies may differ considerably.

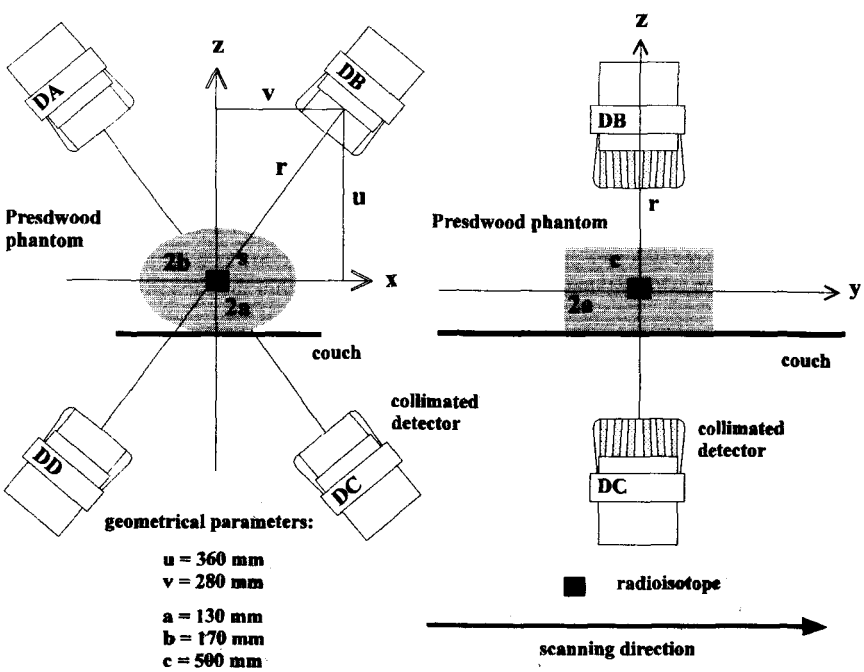

Fig.1. Measuring arrangement using a Presdwood phantom.

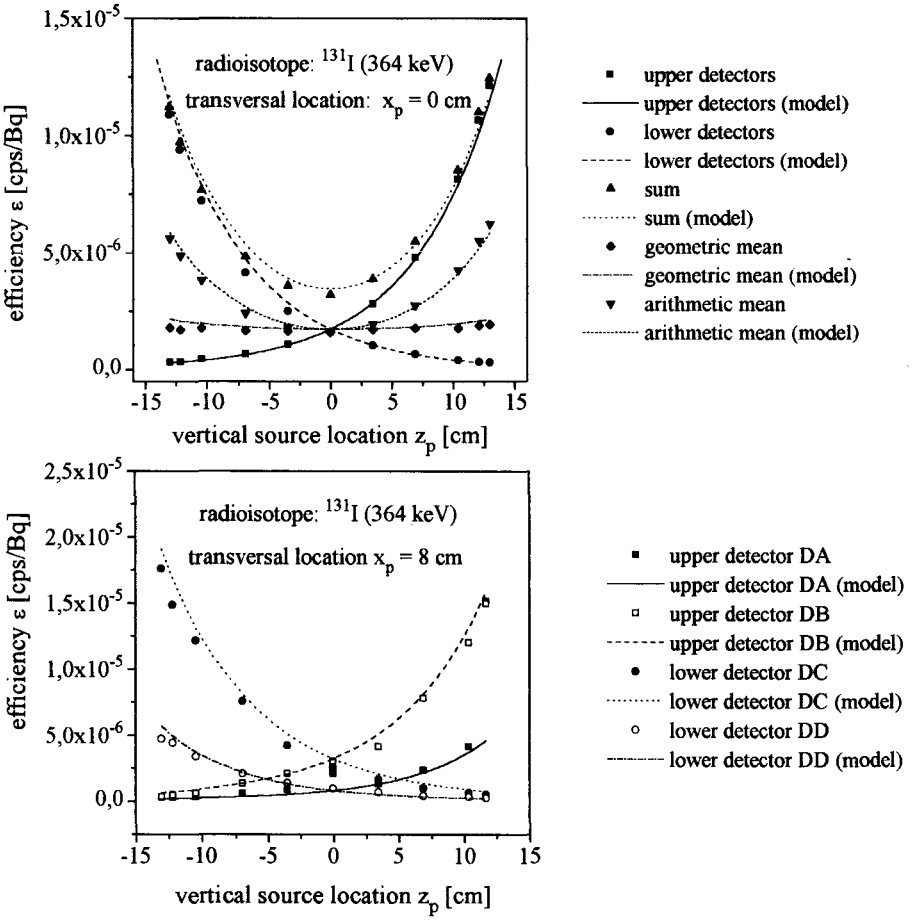

Fig.2. Lower and upper detector efficiencies and their combinations as a function of vertical point source location with/without transversal displacement within the Presdwood phantom.

DISCUSSION

By the usually used method (linear profile scanning) only the distribution of radioactivity in one dimension can be visualized. Perpenticular to this plane there is no resolution. To achieve a complete representation of distribution in the human body a model based on the detector configuration was developed. The computational model presented in this paper was crosschecked for various radionuclides over a wide gamma energy range. Because of the excellent agreement between the measured and the model calculated efficiency values and the fact that the geometrical mean of the efficiency values for all detectors is almost independent from the actual position of the distribution of radioactivity it is possible to calculate the centre of the radioactivity from the detected counts.

The importance for medical applications of this method follows from the combination of the possibility to localize radioactivity and the typical characteristics of the HWBC. So count rates can be directly attributed to organs of the human body and it is possible to study the metabolism. In this connection the HWBC over a wide energy range (60 keV - 2 MeV) ensures a very high accuracy even when lowest activities are provided.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE "WINDOWS" CODE FOR THE INTERPRETATION OF THE RADIOACTIVE BIOASSAY MEASUREMENTS BASED ON U.S. NUCLEAR REGULATORY COMISSION DATA AND THE INITIAL INTAKE/DOSE ASSESSMENTS USING U.S. INTERNAL DOSIMETRY CODES ON EUROPEAN INTERCOMPARISON DATA #

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MAJOR SCIENTIFIC TOPIC NUMBER 3.1 ..... (see page 7)

ABSTRACT (See instructions overleaf)

An European intercomparison study (Gibson,1992) uses the bioassay data for 5 cases of contaminations by inhalation, wound or injection with 7 radionuclides to estimate the initial intakes and doses in 9 laboratories (U.K., Germany, France, Spain and Switzerland).

To extend the area of this intercomparison we made a study to establish the degree of fitness for several computer codes currently used for estimation of initial intakes and doses at Lawrence Livermore National Laboratory (LLNL), USA. This study provided equivalent results within the inherent errors of the data and the method of dose estimation. In our initial intake assessments, the results are more constrained, showing that the model parameters were the roughly the same, while in the European study model parameters were independently chosen by the dosimetrist. The range of the coefficients of variation for the dose estimates show that the choice of the dose conversion factors was fairly consistent among the European laboratories and U.S. dosimetry codes.

To show that it is possible to fulfill a gap in the interpretation of the bioassay measurements, we made at the LLNL a short "demo computing program" to use the US Nuclear Regulatory Commission data and recommendations (CR4884 and DG8009). We present the algorithms, inputs and outputs for our new "Windows" program for the interpretation of radioactive bioassay measurements.

Work partially performed under an I.A.E.A. fellowship at the LLNL.

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PAPER TITLE CPHR-Whole Body Counter Unit. Calibration results.

AUTHOR(S) NAME(S) Cruz Suárez Rodolfo; López Bejerano Gladys; Nogueira Oliveira Carlos; Bastos Becker Paolo.

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MAJOR SCIENTIFIC TOPIC NUMBER ...3. (see page 7)

ABSTRACT (See instructions overleaf)

A high sensitivity whole body counter has been installed at the Centre for Radiation Protection and Hygiene (CPHR-Cuba).The detectors system consists of a 8"x4" NaI(Tl) and a 3"x3" NaI(Tl) scintillation detectors located in a low background room. The room is made of low intrinsic radioactivity steel plates (less than 1 Bq of ^{60}Co per kg of steel), with internal dimensions 2500 mm w by 2500 mm l by 2600 mm h and plate thickness of 162 mm. Internal walls are lined with 3 mm of Pb, 1.8 mm of Sn and 1.5 mm of Cu for background reduction between 10 keV and 3 MeV. The gamma ray spectra are analyzed automatically using a special purpose software package and a personal computer. In order to calibrate the detection system for high energy photon emitters a structure based on the BOMAB phantom which comprise ten elliptical containers was assembled. This structure approximate the physical shape of a human body for 5, 10, 15 years old and an adult person. Phantoms are filled with plastic bags containing radioactive solution of ^{57}Co, ^{22}Na, ^{137}Cs, ^{226}Ra, ^{134}Cs, ^{54}Mn, ^{133}Ba, ^{60}Co, ^{40}K, simulating an uniform distribution. Each photon was measured with NaI(Tl) 8"x4" detector using a tilted chair geometry. Detection efficiency, FWHH and minimum detectable activity as function of energy, for counting time of 30 minutes was calculate for each radionuclide. The calibration factors as a function of weight of the phantoms were calculated too.

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PAPER TITLE

A PBPK Model for Carbon; for describing metabolism in humans for use in internal dosimetry.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

3.1 Internal Dosimetry

ABSTRACT (See instructions overleaf)

The metabolism of carbon has been extensively studied and is well known. Despite this, models currently being used to describe intake, distribution, and retention following a pulsed intake are not adequate for internal dosimetry. Carbon-14 is encountered in the nuclear industry primarily as CO_2, and in nuclear medicine and biomedical research as organic carbon compounds. Environmental ^{14}C will become bound in a variety of hydrocarbons. On occasion, ^{14}C will exist as an insoluble aerosol (inhalation Type S). The dose per unit intake from these different compounds will have a large range, as will the relationship between intake and excretion. This paper describes a physiologically based pharmacokinetic (PBPK) model that has been developed for use with bioassay to evaluate the dose from an intake of ^{14}C. The model has been used to calculate committed effective and equivalent dose following inhalation intakes of compounds containing ^{14}C as gases and vapors, and as Type F, M, and S compounds, as well as from ingestion intakes. Data useful in relating measurable quantities (chest contents, and urine and fecal excretion) to organ contents are given.

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RESTRICTIONS FOR APPLICATION OF EFFECTIVE DOSE IN RADIATION SAFETY

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INTRODUCTION

ICRP in Publications N 26 (1) and N 30 (2) recommended to calculate the annual limit on intake (ALI) of a radionuclide as the smallest value of the annual intakes corresponding to committed effective dose of 50 mSv or committed organ dose equivalents of 500 mSv. Recently ICRP don't limits the value of committed equivalent dose in organs and tissues on ALI calculations. The recommendations of ICRP Publication 61 (3) are the first one, where the annual limit on intake for any radionuclide is based only on a committed effective dose.

It is well known, that the effective dose, E is the sum of the weighted equivalent doses in all the tissues and organs of the body, H_T. The weighting factors for different organs have wide variation in magnitude, e.g. 0.2 for gonads and 0.01 for bone surfaces or skin (3). When one is exposed to radionuclides, uniformly spread in the body, e.g. ^{137}Cs, ^{106}Ru, equivalent doses in all organs and tissues are not different from the effective dose. When one is exposed to organotropic radionuclides irradiated only some organs or tissues (e.g. ^{93}Zr - skeleton, ^{238}U - kidneys, ^{131}I - thyroid), the effective dose become the weighted equivalent dose in the only organ or tissue. In those cases the equivalent dose in organ differs from the effective dose by a factor of the inverse value of tissue weighting factor put to that organ. For ingestion of ^{131}I, as an example, committed equivalent dose in thyroid is close to 200 mSv if effective dose is equal to 10 mSv (value of tissue weighting factor for thyroid is equal to 0.05).

METHODS AND RESULTS

Ratios between committed equivalent dose in organ and committed effective dose both for inhalation and ingestion of 1148 compounds of 383 radionuclides relevant to radiology were calculated with computer code "R-MAN" (4). Computer code "R-MAN" is based on ICRP Recommendations (Publications NN 30, 38, 48, 56, 60 (2, 5-8), publications of US Oak-Ridge National Laboratory (9, 10).

Results of the ratio calculations for some organs are presented in the tab. I. Thus, in accordance with (3), ALI is a value of the annual intake corresponding to committed effective dose of 20 mSv. The list of radionuclides for which H_T/E in some organs exceed 25 includes compounds of ^{93}Zr, ^{109}Cd, ^{113}Cd, ^{113m}Cd, ^{125m}Te, ^{146}Sm, ^{147}Sm, ^{151}Sm, ^{152}Gd, ^{171}Tm, ^{181}Hf, ^{210}Pb, ^{210}Bi, ^{227}Ac, ^{228}Ac, ^{228}Th, ^{229}Th, ^{230}Th, ^{232}Th, ^{231}Pa, ^{230}U, ^{232}U, ^{233}U, ^{234}U, ^{235}U, ^{236}U, ^{238}U, ^{237}Np, ^{238}Np, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{244}Pu, ^{241}Am, ^{242m}Am, ^{243}Am, ^{244m}Am, ^{243}Cm, ^{244}Cm, ^{245}Cm, ^{246}Cm, ^{247}Cm, ^{248}Cm, ^{250}Cm, ^{249}Bk, ^{250}Bk, ^{249}Cf, ^{250}Cf, ^{251}Cf, ^{252}Cf and ^{254}Es.

One mast to expect in relation to the data presented in the tab. I that intake of radioactivity equivalent to ALI of some organotropic radionuclides may result in a committed equivalent dose of 920 mSv in bone surfaces or 520 mSv - in kidneys or 400 mSv

Table I: Comparison of the committed equivalent and effective doses for inhalation and ingestion of 383 radionuclides

Target organs and tissues	Fraction (%) of radionuclides for which H_T differs from E by factor of between:						Maximum of H_T/E ratio, in brackets - the inverse value of tissue weighting factor
	0 and 2	2 and 4	4 and 8	8 and 16	16 and 32	32 and 64	
Bone Surfaces	77	4.7	3.1	3.7	8.4	3.1	46 (100)
Kidneys	91	3.0	2.3	2.3	1.4	0	26 (40)
Thyroid	95	2.0	0.6	0.8	1.6	0	20 (20)
Spleen	98	1.4	0.4	0	0.2	0	20 (40)
Bladder	96	3.2	0.7	0.2	0	0	12 (20)
Liver	87	6.3	6.2	0.5	0	0	10 (20)
LLI wall†	61	15	22	2.0	0	0	8.3 (8.3)
Lungs	56	11	27	6.0	0	0	8.3 (8.3)
Pancreas	99.6	0.3	0.1	0	0	0	7.8 (40)
Red Marrow	86	11	3.0	0	0	0	6.6 (8.3)

† Wall of lower large intestine

- in thyroid and spleen or 240 mSv - in bladder or 170 mSv - in lower large intestine wall and lungs or 160 mSv - in pancreas or 130 mSv - in red marrow.

CONCLUSIONS

It stands to reason that correct applying of effective dose is bounded by share of permissible equivalent doses (*PED*) in some organs:

1/50th of PED in bone surfaces;

1/25th of PED in kidneys;

1/20th of PED in thyroid or spleen;

1/10th of PED in liver or bladder wall;

1/8th of PED in lower large intestine wall, lungs or pancreas.

Therefore, the annual limit for effective dose of 20 mSv is in good agreement with the permissible equivalent dose for any internal organ of 100 mSv established by ICRP Recommendations 1990 (8).

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STOCHASTIC APPROACH TO THE ESTIMATION OF RADIONUCLIDE BODY BURDENS

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It is common for estimation of radiation body burdens to use a compartment model. In these models are proposed the following simplified assumptions: 1) it is may introduced a good determined metabolism coefficients α_{ik} for each radionuclide, and 2) the law of radionuclide entering γ in organism are known in detail. Then the problem reduced to the analysis of system n independent coupling differential equations in the case of model with n compartments:

$$\frac{dq_i}{dt} = \gamma_i + \sum_{k=1}^{n} \alpha_{ik} q_k, \qquad (1)$$

where q_i is radiation loading in i-th compartment , α_{ik} are metabolism coefficients between i and k compartments and i and k are a numbers of compartments (i,k = 1,...,n). The equation (1) decided under initial conditions $q_i(t = 0) = q_i^0$. Thus the such approach to this problem is deterministic.

In reality, the situation is rather complicate. As shown the measurements, the radioactivity contamination of soil on the territory due to the accident on Chernobyl plant is inhomogeneous. This inhomogeneity is due to both the circumstances of accident and just statistical regularity of nuclide's precipitation on soil surface. Therefore, the foods (of the same kind) that are work out on this territory will to have different degree of pollution. The same result is due to the reason that transfer coefficients of radionuclides from soil to plants are not well determined parameters. In this situation the entering of activity $\gamma(t)$ in living organism is not deterministic process and rather stochastic one and in the many cases it may be approximated by impulse Poisson process:

$$\gamma(t) = \sum_j \xi_j g(t - t_j), \qquad (2)$$

where a random quantity of the impulse amplitudes ξ_j are statistic independent and characterised by a certain probability density function $w(\xi)$, and the random time moments t_j distributed on consider time interval (0,T), so that their number N is distributed according to Poison law with the parameter $\langle N \rangle = \nu T$. The function $g(\tau)$ ($g(\tau) = 0$ if $\tau < 0$) describes the form of impulse and is not random one.

Another factor that determines the stochastic of activity entering in a critical organ is temporal fluctuations in parameters α_{ik} that determine the process of metabolism. The such fluctuations may be due to random variations of organism state. In this case the dynamic of nuclides distribution on organism compartments is essentially determines by both relations between the characteristically times $t_{ik} = \alpha_{ik}^{-1}$ of nuclides transfer between compartments and times t_f, that characterise the temporary length of this fluctuations and the value of such fluctuations $\delta\alpha_{ik}$ of the metabolism parameters.

The above mentioned is shown that it allowed to speak only about the probability W(C(t)) that the amount of radionuclides entered in organism (organ) at moment t is equal to C(t). The quantity W(C(t)) is very useful one since it permits to find the probabilities of radionuclide body burdens in the given interval (C_1,C_2) and probabilities of $W(C_1 < C < C_2) = \int_{C_1}^{C_2} W(C)dC$ radioactivity loading on an inhabitant above, for example, the middle loading for reference man.

Then it may to obtain in the pair correlation approximation:

$$\frac{\partial P_t(x)}{\partial t} + \sum_k \frac{\partial}{\partial x_k}\{\sum_i \alpha_{ki} x_i P_t(x)\} + \sum_i \langle \gamma_i \rangle \frac{\partial}{\partial x_i} P_t(x) = \sum_i \sum_k \frac{\partial^2}{\partial x_i \partial x_k}\{\langle \delta\gamma_i(t)\delta\gamma_k(t)\rangle P_t(x) \quad (3)$$

This equation describes a distribution of burdens on all n compartments. The initial conditions for equation (10) are $_{t=0}(x) = \langle \prod_{i=1}^{n} \delta(x_i - q_i^0) \rangle$, where the sign <...>denotes the average on the ensemble of the all possible realisations of the dependencies $\gamma(t)$.

The (3) is equation of Einstein-Fokker-Plank type that describe the temporary evolution of burden on compartments in diffusion approximation. Since the equation (3) has coefficients that linear dependent on coordinates x_i it is may to claim that the system of random quantities x_i are obey the normal law of distribution and consequently for the unknown function P may to write down:

$$P_t(x_1,\ldots,x_n) = \frac{1}{\sqrt{(2\pi)^n \det(\lambda_{jk})}} exp\{-\frac{1}{2}\sum_{j=1}^{n}\sum_{k=1}^{n}\Lambda_{jk}(x_j - \xi_j)(x_k - \xi_k) \quad (4)$$

where the elements of correlation matrices $\lambda_{jk} = \lambda_{kj}$ and $\left(\Lambda_{jk}\right) = (\lambda_{jk})^{-1}$. The averages $\xi_i = \langle x_i \rangle$ and correlation coefficients λ_{jk} depend on variable t only. For determination of ξ_i and λ_{jk} we have the following equations ($\gamma_i = \langle \gamma_i \rangle(1 + \delta\gamma_i)$, $<\delta\gamma_i>=0$):

$$\frac{d\xi_l}{dt} - \sum_k \alpha_{lk}\xi_k = \langle \gamma_l \rangle ; \qquad \frac{d\lambda_{lm}}{dt} - 2\sum_k (\alpha_{lk}\lambda_{km} + \alpha_{mk}\lambda_{lk}) = 2\langle \delta\gamma_l \delta\gamma_m \rangle,$$

Let us to consider a simple example of compartment model, namely two- compartment model that used for describing of ^{137}Cs, ^{90}Sr and ^{131}I metabolism[1]. In this case $\alpha_{12} = k_1$, $\alpha_{21} = k_2, \alpha_{11} = -(k_2 + \lambda_p)$, $\alpha_{22} = -(k_1 + k_3 + \lambda_p)$ and $\gamma_1 = 0$, $\gamma_2 = \gamma$. Here k_1 and k_2 are the coefficients of exchange by matter between compartments 1 and k_3 is the coefficient of coming out. λ_p is constant of radioactive decay. For the long-life nuclides ($\lambda_p^{-1} \geq t \geq k_i^{-1}$) we have: $\langle q_1 \rangle \approx {}^{\langle \gamma \rangle k_1}/_{k_2 k_3}$. For the distribution function $P(q_i)$ of burden in compartment 1 we obtained:

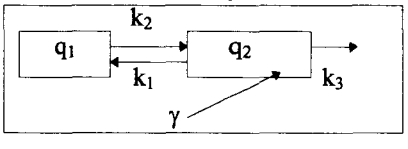

Fig.1. Two-compartment model of metabolism

$$P_t(q_1) = \left[16p^3 L_{22}\det(l_{ik})\right]^{-1/2} \exp\left[-\frac{1}{2}\frac{\det(L_{ik})}{L_{22}}(q_1 - \langle q_1 \rangle)^2\left[1 + \mathrm{erf}\left\{\sqrt{L_{22}}\left[\langle q_2 \rangle - \frac{L_{12}}{L_{22}}(q_1 - \langle q_1 \rangle)\right]\right\}\right]\right],$$

where $rf(\eta) = \frac{2}{\sqrt{\pi}}\int_0^{\eta} e^{-t^2} dt$. In order to obtain the distribution function $P_t(q_2)$ it is sufficient to change the index 1 on 2.

Given real radioactive contamination of foods, with this method it is possible to calculate the distribution of body burdens among inhabitants. For example, knowing contamination of milk in Olevsk (Ukraine), we are calculated the distribution body burdens among inhabitats in this village. This distribution is very close to experimental data(see [2]).

In the case of temporal fluctuations in parameters of metabolism the quantity k_i it is convenient to present in the following manner:

$k_i = <k_i>(1 + \delta k_i)$, $<\delta k_i> = 0$,

here the sign <...> denote the average on the ensemble of the all possible realisations of the random processes γ and k_i.

The role of fluctuations in parameters k_i at the small time in the process of accumulation of radionuclides is negligable. The contribution of fluctuations k_i is essential at the time $t \geq \langle k \rangle + \min\{\alpha_{ik}\}$, that is at the times which are comparable with the times of the establishment of stationary state in considered dynamic system. We are found that the relative body burden in compartment of deposition, that is conditioned only by fluctuations of parameters k_i, is:

$$\Delta\langle q_1 \rangle = \sum_k \langle k_k \rangle \left[\frac{\sum_i \langle k_i \rangle \varepsilon_{ik} - \langle k \rangle \varepsilon_{1k}}{\langle k \rangle + \tau_{ik}} \right] \left[\langle k \rangle - \sum_i \sum_k \frac{\langle k_i \rangle \langle k_k \rangle \varepsilon_{ik}}{\langle k \rangle + \tau_{ik}} \right]^{-1} \qquad (i,k = 1,2)$$

This expression is obtained in the case of long-life radionuclides $\lambda_p << \langle k_1 \rangle + \langle k_2 \rangle + min\{\alpha_{ik}\}$. In this expression $\varepsilon_{ik}=<\delta k_i(t)\delta k_k(t)>$, and τ_{ik}^{-1} is correlation radius in cross-correlation function $<\delta k_i(t)\delta k_k(t')>$. As will readily observed, the fluctuations $\delta k_i(t)$ with small τ_{ii} has a dominant role in $\Delta\langle q_1 \rangle$. Fig. 3 gives a presentation about $\Delta\langle q_1 \rangle$ (axis z) as function of fluctuations δk_1 (axis x) and δk_3 (axis y) in the case of radionuclide I^{131} (k_1=0.04, k_3=0.09, k_2=0). Thi example is illustrated that temporal fluctuations in parametrs of metabolism effect the radiation body burdens in depot of

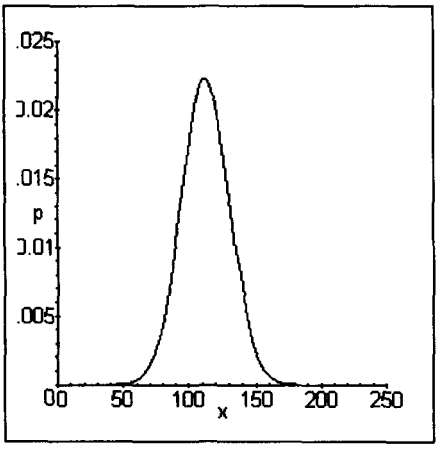

Fig.2.Distribution of body burdens among inhabitants in Olevsk.

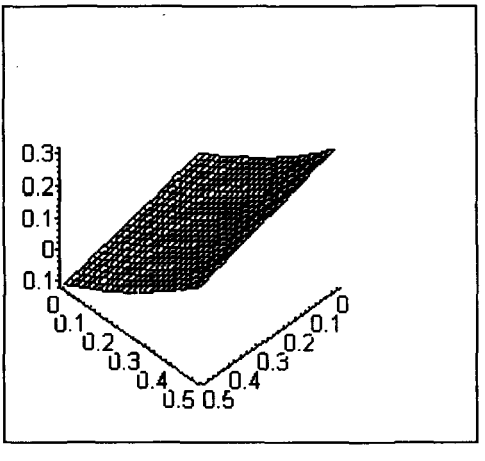

Fig.3. $\Delta\langle q_1 \rangle$ as function of fluctuations δk_1 and δk_3

radionuclides deposition. The change in body burden may be essential in the case of other nuclides (for example: ^{90}Sr, ^{45}Ca, etc.).

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OBSERVATION OF SOME ANOMALIES IN THE ICRP METABOLIC MODEL OF URANIUM

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A simple and sensitive analytical method was developed for determining the concentrations of uranium (^{238}U) in biological and environmental samples. The minimum detection limit (M.D.L) of 0.08ng uranium (0.1 Bq of ^{238}U) in a sample was obtained with this method. Data on daily intake (through diet, drinking water and air), the excretion (through urine) and organ burdens of uranium were obtained for human subjects living in Bombay (India), and were employed to obtain important human metabolic parameters of U such as, gastrointestinal absorption factor(f_1), excretion ratio and biological half-lives of uranium retained in various human organs. The study showed that the f_1 factor for uranium available to humans at dietary level (0.5-5ug.d^{-1}) is 0.016, which is much lower than 0.2 suggested in ICRP 30. The biological half-lives, of uranium stored in kidney and muscles were estimated to be much longer than those proposed by ICRP.

INTRODUCTION

The large variations reported in the uranium concentration of similar human tissues and body fluids for subjects with comparable intakes of uranium, underline the need for the developement of sufficiently sensitive and accurate analytical method for uranium determination (Wrenn et al.1985, Dang 1993). The consistently lower concentrations of uranium in human tissues and body fluids reported by different workers in recent years (Eisenne et al., 1987, Igarashi et al., 1987, Wrenn etal., 1985, Dang et al., 1992, 1993, 1995) in comparison to the values reported in ICRP 30 (1979) also indicated that the ICRP metabolic model for uranium, may need revision.A program was initiated at Bhabha Atomic Research Centre (India) with the main objectives: 1) to develop a sensitive and reliable analytical method (with extremely low reagent blank) for the determination of uranium in biological and environmental samples, 2) employ the method to obtain the daily intake, excretion and organ burdens data on human subjects, 3) to use these data for evolving various important human metabolic parameters of uranium, 4) to compare the metabolic parameters thus obtained with those proposed by ICRP..

MATERIALS AND METHOD

The human tissue (obtained at autopsy), diet and body fluids (urine and blood serum) samples were obtained from population living in natural background area of urban India (Bombay). The samples of air and drinking water were also collected, for uranium analysis. While collecting samples, it was ensured that various samples were obtained from subjects with similar social background, food habits and that all the subjects were living in Bombay City for long duration of time.

The tissue, diet and blood serum samples were homogenised, freeze-dried and powdered before analysis. Urine and drinking water samples were first digested in nitric acid and, then the U present in the samples was co-precipitated with calcium phosphate. The procedure is described in detail by Dang et al (1992). The air samples obtained from various parts of the city were directly analysed along with the filter paper used for air sample collection.

The freeze dried samples and calcium phosphate precipitate were sealed and then irradiated for 24-36h in a nuclear reactor in neutron flux of = 10^{13} n cm^{-2} sec^{-1}. The neutron irradiation converted the uranium present in the samples to neptunium (^{238}U (n,)^{238}U-----^{238}Np). The uranium standard (200 ng of uranium nitrate solution dried on ashless filter paper) was also sealed and taken up for irradiation along with the samples. The irradiated samples containing ^{238}Np were digested in acid and then the ^{238}Np was chemically separated by two step chemical separation, using first DOWEX-1 (an anion exchange resin)and then its co-precipitation with barium sulphate precipitate. ^{237}Np was also used as tracer for the chemical yield determination. The gamma activity (227.5 keV and 278 keV gamma rays) of ^{238}Np was counted using 54 CC Hyperpure Germanium detector coupled to 4096 channel analyser. The uranium standard was also processed in the same way and counted, to compare the activities of sample and standard for the quantification of U. The details of analytical procedure are given elsewhere (Dang et al.,1992 Dang et al., 1993). The reliability of the analytical method was tested using the Standard Reference Materials.

RESULTS AND DISCUSSION

The total daily intake of uranium by adult Indian subjects are shown in Table 1. The daily urinary excretion levels of uranium are also included in the same Table.

Gastro-intestinal absorption factor(f_1)

The uranium entering the blood stream (transfer compartment) after absorption through the gastrointestinal tract, gets deposited in various systemic body organs such as, kidney, skeleton, muscles. Because of the relatively short biological half-lives of U deposited in various organs, as compared to the average age of the subjects (45Y), an equilibrium between the uranium entering and leaving the organs could be assumed. The U leaving the organs is excreted through the kidney into urine. Contribution of air was <2% of total intake. The f_1 factor was estimated to be 0.016 from the daily urinary excretion and intake of uranium.

Excretion Ratio

The excretion ratio is the fraction of uranium excreted daily in urine from the uranium present in the transfer compartment which is assumed to be the blood serum. Table 2 gives the uranium burdens for some of the body organs and total uranium in blood serum. The burdens were calculated from the concentrations of uranium determined in tissues obtained from these organs and average weight of the organs of an adult Indian. Employing the daily urinary excretion of 12.2 nanogram and the 30 ng of U in total serum pool, the excretion ratio was worked out to be 41%. Th excretion ratio is quite close to the clearance fraction of 54% proposed by ICRP. The excretion ratio studied in the present work could provide a guideline for an upper limit which may be used for screening the subjects for any impending kidney malfunction.

Biological half-lives of uranium retained in some organs

The organ burdens calculated on the basis of daily intake of uranium, the gut absorption factor obtained in the present study and the ICRP distribution factors (f_2) for these organs are also shown in Table 2 along with the measured values.

Table 1. The daily intake and urinary excretion (ug d^{-1}) of uranium for an adult Indian population group.

Source of intake	No. of samples analysed	Daily intake Range	Mean ± SD
Duplicate Diet	18	0.24 - 1.3	0.64 ± 0.4
Drinking Water	10	0.05 - 0.39	0.14 ± 0.1
Air	9	0.002 - 0.03	0.01 ± 0.1
Total Intake			0.79 ± 0.2
Daily Urinary Excretion (ng.d^{-1})	28	4.0 - 52.0	12.3 x 2.2

Daily Intake of water 2.6L, and Air 20M^3 Daily Urine vol. 1.3L
Daily Urinary excretion is based on Geometric mean (GM) conc. of U.

The calculated and measured values for skeleton are comparable, but for kidney and muscle, they are 4 and 70 times lower, indicating that the true biological half-lives for uranium retained in kidney and muscle may be much longer. As the concentration of U in muscle is extremely low,the longer biological half-life may not have significant effect on human health the longer biological half-life of uranium deposited in kidney (upto four times longer than that proposed in ICRP metabolic model) however, may warrant the reduction in the annual limit if soluble class (D&W) of uranium..

Table 2. Organ burdens of kidney, skeleton and muscles and blood serum of Indian subjects

Organ	Organ Weight (g)	Measured Uranium Burden (ug) Range	Geometric Mean (GM)	calculated
Kidney	230	0.02 - 0.58	0.13	0.03
Skeleton	7000	0.70 - 11.4	2.18	2.18
Muscle	21000	0.06 - 1.54	1.20	0.03
Blood Serum	2300	0.02 - 0.06	0.03	-

CONCLUSIONS

The f_1 factor 0.016 for uranium incoporated in diet, obtained in present sudy, is much lower than 0.2, as reported in ICRP 30. This study also showed that the biological half-life of uranium for kidney may be upto four times longer than that is proposed in ICRP model. The excretion ratio obtained in the present study may provide a useful guideline for screening the occupational workers handling uranium for any impending kidney damage.

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IMPLEMENTATION OF THE ICRP 66 RESPIRATORY TRACT MODEL : EXAMPLE OF OCCUPATIONAL EXPOSURE TO URANIUM OXIDES FORMED IN A NEW LASER ENRICHMENT PROCESS.

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ABSTRACT

A new uranium enrichment facility using laser isotopic separation generates aerosols consisting of U_{metal} + UO_2 with traces of U_3O_8. Results of lung absorption to blood showed that the U_{metal} + UO_2 transportability was appreciably greater than for other industrial forms of UO_2. Taking into account the new ICRP human respiratory tract model, the data were used as a basis for assessing the dose coefficient, for the dust sampled at the workplace.

INTRODUCTION

A new method for uranium enrichment, developed in France, is based on laser isotopic separation (1-2). In this process uranium metal is vaporised to be selectively photoionised by laser beams. During the removal of condensed uranium from graphite collecting plates, workers may be exposed to different types of uranium aerosols. The aerosols formed have been previously identified as consisting of a mixture of U_{metal} + UO_2 with a highly variable ultrafine component (1-20 % by activity) (1-2).

Only a paucity of data are available on the physico-chemical characteristics and absorption to blood of aerosols formed from uranium metal (1-6).The aim of the biokinetic study with rats described here was to provide specific experimental data for assessing dose coefficients ($Sv.Bq^{-1}$), and due consideration has been given to the new respiratory tract model of ICRP 66 (7) and to the most recent recommendation of ICRP 60 on annual dose limit of 20 mSv for workers (8).

MATERIALS AND METODS

Physico-chemical characteristics of dust : The dust used for the biokinetic study was produced by the SILVA process and collected at Pierrelatte. Physico-chemical analyses carried out by X-Ray diffraction show that trace amount of U_3O_8 were present. Electron Energy Loss Spectrometry (EELS) show that the particles consisted of a natural uranium metal core oxidised at the surface to uranium dioxide.The activity median aerodynamic diameter (AMAD) of the dust measured at the workplace varied between 5 and 10 µm (σ_g=2.5). However the dust contains clusters of particles of about 0.1 µm which represent between 1 and 20 % of the total radioactivity. The density of the dust was 7.2 g. cm^{-3} and the specific surface area 1 $m^2.g^{-1}$.

Sample preparation : To obtain particles of respirable size for the animal study, the dust sample was sedimented in ethanol. The AMAD of the particles was controlled using high resolution particle size analyser (API Aerosizer, Malvern Instrument Ltd., Malvern, Worcs, UK) was 1.61 µm (σ_g=1.29).

Administration of dust : The rats used were females of the HMT strain (NRPB, Chilton). A saline suspension was administered to a group of 35 rats by intratracheal instillation, using a procedure described previously (9) and the initial lung deposit was 86 µg of natural uranium

Tissue analysis : To assess the lung retention and transportability characteristics of uranium, groups of 5 rats were sacrificed at 1, 3, 7, 14, 28, 84 and 168 days after exposure. The lungs, liver, kidneys, carcass (excluding the gastrointestinal tract) and urine were ashed at 500°C and the uranium content determined by the delayed neutron counting technique. These experiments on animal were performed by scientists certified by the French Ministry of Agriculture for carrying out these procedures.

Interpretation of data : Publication 66 of ICRP recommends to determine the specific dose per unit of intake ($Sv.Bq^{-1}$), as a function of the following three main experimental parameters : particle size distribution of dust measured at the workplace, physico-chemical properties and calculation of dissolution fractions and rates obtained from animal experiment. SILVA process is still being a pilot process, so it reasonably seems now time to compare calculated doses with the recommended ICRP 60 annual dose limit , for occupational exposure, and to assess a specific ALI calculation resulting from this study.

The absorption rates into blood obtained from the experimental animal data must be scaled in order to obtain the human data on particle clearance and excretion, using a graphically-interactive parameter-fitting personal computer program (GIGAFIT) (10), and then available in LUDEP (11) application. The dose coefficients of the dust were obtained using the computer program LUDEP which is designed to implement the new respiratory tract model of

ICRP 66. In this program, besides default absorption parameters for type F (fast), M (moderate), and S (slow) materials, specific experimental data can be implemented.

RESULTS

The tissue distribution and excretion data obtained from the animal study, expressed as a percentage of the initial alveolar deposit, are given in Table 1.

Table 1. Tissue distribution and retention of uranium after intratracheal instillation of $U_{metal} + UO_2$.

			% Initial	Lung	Deposit[a]		
Time days	Lungs	Kidneys	Carcass	Total Body[b]	Urine	To blood[c]	Faeces + GIT[d]
1	60.8 ± 2.7	2.8 ± 0.4	5.9 ± 0.4	69.5 ± 2.8	15.2 ± 0.9	23.8 ± 1.1	15.3 ± 2.9
3	47.1 ± 4.6	2.8 ± 0.4	6.1± 1.1	56.0 ± 4.7	20.8 ± 0.7	29.7 ± 1.4	23.2 ± 4.7
7	43.0 ± 2.9	1.0 ± 0.11	4.9 ± 0.4	48.9 ± 2.9	24.8 ± 1.2	30.7 ± 1.3	26.3 ± 3.1
14	41.5 ± 3.8	0.46 ± 0.05	4.9 ± 0.2	46.9 ± 3.8	27.8 ± 1.3	33.2 ± 1.3	25.3 ± 4.0
28	31.3 ± 3.1	0.44 ± 0.06	5.3 ± 0.4	37.0 ± 3.1	30.8 ± 1.3	36.5 ± 1.4	32.2 ± 3.4
84	12.6 ± 1.9	< 0.28[e]	5.6 ± 0.6	18.5 ± 2.0	36.8 ± 1.5	42.7 ± 1.6	44.7 ± 2.5
168	7.5 ± 1.9	< 0.28[e]	4.5 ± 0.3	12.3 ± 1.9	38.2 ± 1.5	43.0 ± 1.5	49.5 ± 2.4

a: Mean ± se, 5 animals per group, ILD = 600 ng ^{235}U (86 µg natural uranium)
b: Sum of uranium in tissues, excluding liver (undetectable)
c: Kidneys + carcass + urine; absorption from gastrointestinal tract (GIT) f_1 was ignored
d: ILD -(total body + urine)
e: Minimum detectable activity; all livers were in this category

They show that the uranium is highly transportable; 24 % and 43 % of the initial alveolar deposit is absorbed into the blood by 1d and 168d respectivly after exposure. Using the computer program GIGAFIT, 27 % of the amount deposited in the alveolar region was rapidly cleared with a rate of 1.01 d^{-1} and the remainder with a rate of 0.0093 d^{-1} . The fraction rapidly absorbed to blood was calculated to be 33.7% with a rate (s_r) of 1.42 d^{-1} and the remainder was absorbed with a slow rate (s_s) of 0.0048 d^{-1}. After absorption, uranium is excreted preferably in urine, eg. 64%, 80% and 90 % after 1 d, 7 d and 168 d respectively. The main site of retention in systemic tissues was the skeleton (carcass); the amounts present in the liver were trivial.

Calculations of dose coefficient (Table 2), using the computer program LUDEP, were based on a mean breathing rate of 1.2 $m^3 .h^{-1}$, a measured particle density of 7.2 $g.cm^{-3}$, and were also made for particles with an AMAD of 5 µm and 0.1 µm. The absorption rates were calculated from the experimental data using the GIGAFIT computer program . For comparison, the dose coefficients were also calculated for aerosols of 5 µm AMAD using the default absorption parameters for type M (10% at 10 min and 90 % at 140 d), and for type S (0.1 % at 10 min and 99.9 % at 7000 d).

Table 2. Dose coefficients for $U_{metal} + UO_2$ ($Sv.Bq^{-1}$) : comparison of the doses calculated with experimental values and with default values.

Compound	Final dose coefficient ($Sv.Bq^{-1}$)
Dust $U_{metal} + UO_2$ [(a)(b)]	$3.34\ 10^{-6}$
Type M	$4.43\ 10^{-6}$
Type S	$12.75\ 10^{-6}$

[(a)] Sum of the activity contribution of each isotope : ^{234}U (0.63), ^{235}U(0.07), ^{238}U (0.3)
[(b)] Aerosol generated by the SILVA process; average AMAD composition is 0.1 µm (20%) - 5 µm (80%)

DISCUSSION

This study was designed to provide an experimental basis for assessing the consequences of human exposure to a $U_{metal} + UO_2$ produced by SILVA process. Emphasis is placed on the comparative behaviour of results obtained on the same dust in a previous study (1-2) and with basic compounds such as uranium dioxide and octoxide.

The biokinetic data reported here are similar to that obtained after inhalation of $U_{metal} + UO_2$ by rats (1-2). The high urinary excretion of uranium, eg. 34 % by 21 d has been confirmed (Table 1). The absorption characteristics of the $U_{metal}+UO_2$ dust contrast sharply with those obtained for two industrial UO_2 preparations in which 3-6 % was transferred to the blood by 168 d (12). This different behaviour of uranium may be due to the rapid dissolution of the ultrafine component of the dust. However, the relatively large amounts of uranium which were absorbed after the early clearance phase, eg. 20 % of the initial lung deposit between 1 d and 168 d, suggest that the UO_2 present in the dust is more transportable than for particles which consist solely of this chemical form. It is also noteworthy that the absorption of uranium was appreciably greater than for U_3O_8 when about 15 % was transferred to the blood by 168 d after exposure in the same species of rat (6,13) After the absorption of uranium by the blood the tissue retention and excretion characteristics of $U_{metal} + UO_2$ (Table1) are similar to those obtained after the administration of very transportable compounds such as U(VI) bicarbonate or nitrate (9).

The results calculated with the experimental s_s and s_r values are very close to those obtained with the default values for a type M component. In this case, the value of the s_s rate has the most drastic influence on the specific dose coefficient. Due to the high contribution of the 0.1 µm particles to the specific dose coefficient, the amount of fine particle in the aerosol is the most important parameter to determine.

The dose coefficient for the aerosol containing 20 % of particles with an AMAD of 0.1 µm and 80 % of particles with an AMAD of 5 µm is calculated to be 3.34 10^{-6} Sv.Bq $^{-1}$. This corresponds to a specific ALI of 5980 Bq or 150 mg of uranium with a given isotopic activity composition (Table 2) , a specific activity of 39770 $Bq.g^{-1}$, and derived from the primary annual limit of 20 mSv. However in the view of the rapid absorption of uranium into the blood and hence potential damage to the kidneys, the daily intake should be limited to 2.5 mg (14). It should be emphasised that the above values may need to be revised in the light of the new ICRP 69 (15) systemic model for uranium.

CONCLUSION

The data presented in this paper show that the biokinetics of a $U_{metal} + UO_2$ dust produced by the SILVA process can differ appreciably from other uranium oxides. The results calculated with the experimental s_s and s_r values are very close to those obtained with the default values for a type M component. This uranium compound shows a highly transportable behaviour, with 24 % and 43 % of the initial alveolar deposit absorbed in the blood by 1 d and 168 d respectivly. Consequently, it is recommended that intakes of this particular dust by workers should be limited by the chemical toxicity of the uranium and assessments of such intakes based on the urine analysis. It should be emphasised that the dust used in this study has a high ultrafine component, about 20 % (due to the pyrophoricity of uranium), and further studies on the biokinetics of dusts with different compositions are needed before more comprehensive guidance on personal monitoring can be given.

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IMPACT OF ICRP PUBLICATION 68

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ABSTRACT

ICRP Publication 61 was a temporary replacement for ICRP Publication 30. It gave ALIs but not the underlying dose conversion factors. ICRP Publication 68 has now been issued to replace Publication 61; it contains the dose conversion factors but not the ALIs, so comparison is impossible without carrying out calculations. This paper presents comparisons between the two publications and calculates the ICRP Publication 68 ALIs for some of the more common radionuclides.

INTRODUCTION

An operational health physicist reviewing a publication of the ICRP feels rather as a Catholic parish priest must feel reviewing a Papal encyclical. Not all the details of the publication are fully understood, some may be disagreed with, but in the end it is a document that will have to be accepted and used.

After the major revision of the ICRP's recommendations in Publication 60 (1) much earlier data, including the many volumes of Publication 30 (2) were no longer valid. The ICRP provided a temporary replacement for Publication 30 in the form of Publication 61 (3). Publication 61 provided ALI values for over 700 radionuclides. These ALIs were, in the normal ICRP manner, rounded to one significant figure. Because of this rounding it is not strictly possible to calculate ALIs for mixtures of radionuclides because the rounding of the input data may result in a very large error in the result of the calculation. In addition, dose calculations in accident conditions need to be made to better than one significant figure. For those needing to carry out such calculations an ICRP Publication with dose conversion factors has been eagerly awaited.

ICRP has now issued Publication 68 (4) to replace Publication 61. It provides dose conversion factors for radionuclides, using the new ICRP lung model (5) with transportability classes of F, M and S. (F = fast, M = medium and S = slow). The inhalation conversion factors are provided for aerosols with particles of an activity median aerodynamic diameter (AMAD) of 1 μm and of 5 μm, which are the default values for public and occupational exposures respectively.

IMPACT OF THE NEW LUNG MODEL

In the past an aerosol with an AMAD of 1 μm was taken as the default and a relatively simple expression was available to calculate an ALI corrected for other values of AMAD. Some organisations were able to demonstrate that typical aerosols in their industry had AMADs of 10 μm or more and justified the use of a site specific ALI. This made a considerable reduction in the calculated inhalation doses. ICRP 68 presents dose conversion factors for 1 and 5 μm AMAD aerosols but there is no longer a simple means to calculate dose conversion factors or ALIs for aerosols of other particle sizes . Contrary to the expectation of some papers presented before the publication of the new lung model, the 5 μm AMAD does not always result in the highest dose conversion factor: sometimes the 1 μ AMAD value is higher.

IMPACT ON "RADIOTOXICITY"

With the publication of ICRP 61 it became clear that previous classifications of radionuclides into groups, usually four, on the basis of their radiotoxicity would need revising. In Australia a new classification was developed (6). The publication of ICRP 68 has rendered this "new" classification out of date. The authors have used the ICRP 68 data to revise the new classification with the results presented in Table 1. The classification is based on the highest ICRP 68 dose conversion factor for each radionuclide irrespective of the AMAD or the transportability class.

Table 1. Changes in "Radiotoxicity" Group if ICRP 68 data are used in place of ICRP 61 data.

	ICRP 61	ICRP 68
Total number of entries.	757	760
Radionuclides in Group 1.	56	58
Radionuclides in Group 2.	212	243
Radionuclides in Group 3.	388	399
Radionuclides in Group 4.	101	52
No. of radionuclides that have changed group.	NA	118
Number of radionuclides that have moved to a lower group. i.e. are more radiotoxic.	NA	101

IMPACT ON THE NATURAL RADIONUCLIDE DECAY SERIES

For the mining and related industries that deal with natural uranium and natural thorium it is necessary to calculate ALIs and DACs for the mixture of radionuclides present in the natural decay series. In most mining situations the decay series is in approximate equilibrium. The authors have calculated ALI_{inh} for the natural series in equilibrium and for the separated metals. The results are presented in Table 2.

Table 2. ALI_{inh} for the Uranium-238 and Thorium-232 series.

	ICRP 30	ICRP 61	ICRP 68
Uranium 238 series ALI			
Uranium series	1200[a]	2000	2100[b]
Uranium metal	600[a]	600	2500
Thorium 232 series ALI			
Thorium series	400	600	700[c]
Thorium metal	70	120	500

a. From reference 7. b. 8000 for type S, 5 μm. c. 4600 for type S, 5 μm.

CHANGES IN ALI FOR SELECTED IMPORTANT RADIONUCLIDES

In ICRP 61 there were seven radionuclides with ALI_{inh} of 100 Bq or less; there are only two in ICRP 68. Table 3 lists those radionuclides that had ALI_{inh} less than 100 Bq in ICRP 61 and gives the ICRP 2 (8), ICRP 30 (2) and ICRP 61 values for comparison. The ICRP2 (8) values are not strictly comparable as the models and factors have changed considerably since 1959. They have been calculated from the ICRP 2 40 hour week DAC and corrected for the reduction in annual dose limit and are included as an indication of the change in recommended limits since 1959. For these seven radionuclides all the ICRP 68 ALI values are greater than the ICRP 61 values, the increase ranging from 150 % to 600 %. Changes in the most restrictive ALI_{inh} for some other radionuclides of interest are given in Table 4.

Table 3. Most Restrictive ALI_{inh} for the Most "Radiotoxic" Radionuclides

	Most restrictive ALI_{inh}			
Radionuclide	ICRP 2	ICRP 30	ICRP 61	ICRP 68
Ac-227	71	20	20	32
Th-229	not listed	30	60	200
Th-232	71	40	90	480
Pa-231	35	60	100	150
U-232	3500	300	100	570
Cm-248	not listed	50	80	140
Cm-250	not listed	not listed	10	25

Table 4. Most Restrictive ALI_{inh} for Selected Radionuclides

	Most restrictive ALI_{inh}			
Radionuclide	ICRP 2	ICRP 30	ICRP 61	ICRP 68
Co-60	1×10^7	1×10^6	4×10^5	6.9×10^5
Sr-90	1×10^4	1×10^5	6×10^4	1.3×10^5
Cs-137	2×10^6	6×10^6	2×10^6	3×10^6
Ra-226	1×10^3	2×10^4	9×10^3	1.3×10^3
Pu-239	3×10^3	200	300	430

CONCLUSIONS

ICRP Publication 68 presents dose conversion factors but leaves the reader to convert these to ALIs and DACs so is less operationally convenient than ICRP30.

Although several conference presentations gave the impression that the new lung model would make an AMAD of 5 μm the most restrictive, this is not the case. For many radionuclides a 1 μm AMAD remains the most restrictive. It is no longer easy to calculate an ALI for a site specific AMAD value.

There has been a relaxation in the ALI_{inh} for the most hazardous radionuclides by a factor of between one and a half and six. There has also been a relaxation in the ALI_{inh} for the natural decay chains. Overall, however, there is a slight reduction in the ALI values.

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IN VITRO SOLUBILITY OF CHERNOBYL NUCLEAR FUEL AEROSOL WITH RESPECT TO COLLECTIVE BEHAVIOR OF ITS RADIONUCLIDES

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INTRODUCTION

The Chernobyl accident was the first one, when big groups of peoples were affected by aerosol containing poor investigated chemical speciation of fission products – matrix-bond radionuclides. Due to uranium-oxide matrix of the spent nuclear fuel, the behavior of the nuclear fuel particle (*NFP*) radionuclides both in the environment and in the body acquires a collective character. Bound by the matrix, the radionuclides do not escape from the fuel particles until the matrix is chemically destroyed. If any nuclear fuel radionuclide tracer was found in the sample from environment or from human barrier organ e.g., lungs or skin, all another NFP radionuclides have to be expected to present in the sample too. Their normalized activities have to be closed to those from Kutkov et al. (1) having regard to the radioactive transformation.

Cuddihy et al. (2) give the results of a study on the radionuclide leaching rate from the particles found in late May 1986 on a wooden container which was about $80\,km$ from Chernobyl at the time of the accident. Approximately 90% of activity were slow to leave the particles, with a leaching half-time of about 160 days and leaching constant of about $0.004\,d^{-1}$. This was characteristic of all the fuel particle radionuclides including ^{131}I, ^{134}Cs and ^{137}Cs commonly have the relatively soluble chemical compounds. Such an effect of U substrate on leaching of radionuclides has been reported earlier by Dua et al. (3) and from our viewpoint it is the result of binding the fission products by the matrix of uranium oxides.

MATERIAL AND METHOD

The strength of a radionuclide-matrix bond in the nuclear fuel particles was studied using dialysis of an aerosol sample in Ringer's solution. Since all commonly occurring Cs compounds are soluble, 100% of the ^{137}Cs activity could be expected to pass into solution in a short time. However, Chernobyl on-site samples collected in June 1986 contained about 40% of aerosol Cs in a relatively insoluble form, probably, as part of fuel particles, the remainder being soluble (4). In 1990-1991, the "Shelter" (special entombment of damaged Chernobyl reactor) samples showed that at least 30% of Cs were still in a nontransportable form, as part of nuclear fuel particles and did not pass in 2 days into aqueous phase through a membrane with a pore diameter of about $0.5\,\mu m$.

A special investigation of longtime dialysis kinetics for "Shelter" samples was carried out in 1992. The test procedure resembled the one described by Eidson and Griffith (5). Smears from vertical surfaces with an area of up to $100\,cm^2$ were collected on ash-free aerosol filters in premises of the "Shelter". All the samples spread to contain nuclear fuel

radionuclide tracers (^{144}Ce, $^{154,155}Eu$). The activities of gamma-emitting fission products normalized to that of ^{144}Ce in the smears showed no statistically significant difference from those in the representative NFP at the time of sample measurement.

Disks of $35\,mm$ in diameter made out of filter sections chosen for the heaviest surface contamination were fixed between two membrane filters MFA-MA-12 (Russia) with a pore diameter of about $0.5\,\mu m$, fixed between closely assembled Teflon rings and placed for dialysis into thermostatic vessels containing $150\,ml$ of Ringer's solution at room temperature.

The leaching of radionuclides from samples into aqueous phase (dialyzate) was investigated on 2, 7, 21, 35, 49, 63, 77, 92, 106 by 122 days after onset of dialysis. Every time assembly extracted from a vessel, it was washed up by 10 ml of distilled water and examined by $\gamma-$spectrometry without disassembling. Every spent portion of dialyzate was acidulated by 5 ml concentrated nitric acid and examined with $\gamma-$spectrometry in parallel with sample examined. Then the total $\alpha-$activity in dialyzate was determined. After that procedure the contaminated solution was changed for fresh one and dialysis continued.

Till the ending of experiment the assemblies were disassembled, and the filters with membranes were used for determination of total $\alpha-$activity, remaining in the sample.

RESULTS

The data obtained on the levels of ^{137}Cs and total alpha emitters were used to determine a constant for the sample–solution transfer of the radionuclides. The transportability of over 70% of ^{137}Cs was found to be weak in the sample, with a leaching constant of $0.002 \pm 0.001\,d^{-1}$. The remaining Cs fraction was 20 times as transportable. For the total of alpha emitters, the dialysis constant was $0.005 \pm 0.002\,d^{-1}$. The difference between the leaching constants for ^{137}Cs and total alpha emitters is not statistically significant ($p > 0.05$). This confirms a supposition that these radionuclides have similar chemical speciation, being matrix bound. Using the summarized data on total alpha and Cs leaching, the constant for radionuclide leaching from nuclear fuel particles is estimated at $0.003 \pm 0.002\,d^{-1}$ and closed to $0.004\,d^{-1}$ from Cuddihy et al. (2).

CONCLUSIONS

It is evident that an *in vitro* experiment cannot simulate the processes of interaction between aerosol particles and intrapulmonary medium; the findings, however, evidence in favor of a hypothesis that there is an equally strong bond between the matrix of the nuclear fuel and fission products of different nature. The results are in good agreement with earlier data on the binding of fission products by U matrix from Eidson and Griffith (5) and Dua et al. (3). This led us to develop a special dosimetry model to describe the behavior of the nuclear fuel particle radionuclides in the human respiratory system (6).

It stands to reason that the NFPs are the crystals produced by fragmentation of fuel pellets. Their lattice forms a fuel matrix and strongly holds fission products and transuranium radionuclides. By analogy with uranium oxides assigned by ICRP to inhalation class Y by their transportability in the respiratory system (7), all radionuclides entering the respiratory system as a part of fuel particles should be assigned to the same inhalation class.

On penetrating the body, the matrix–bound radionuclides of the nuclear fuel particles exhibit collective behavior but in the respiratory system, the gastrointestinal tract and skin, until the bonds between them are broken due to chemical destruction of the nuclear fuel matrix that takes place in these barrier organs. Chemical destruction of aerosol particles in these organs is a fundamental process resulting in transfer of radionuclides to body fluids where they become independent.

Therefore, a model for the behavior of matrix-bound radionuclides in the body can be proposed as follows.

1. Due to the matrix, the behavior of fuel particle radionuclides in the body acquires a collective character. Matrix-bound radionuclides penetrate to "body fluids" only after a chemical destruction of the fuel matrix.

2. The matrix-bound radionuclides acquire biokinetic properties of uranium oxides, normally unusual for them, so that they should be assigned to nontransportable compounds of inhalation class *Y*. When fuel particles enter the gastrointestinal tract, the fractional absorption of matrix-bound radionuclides does not exceed that of uranium oxides.

3. Distribution and retention of radionuclides, penetrating into "body fluids" after NFP destruction has individual peculiarities and is determined by biokinetic properties of the related elements only.

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RECONSTRUCTION OF THE SIZE OF NUCLEAR FUEL PARTICLE AEROSOL BY THE INVESTIGATION OF A RADIONUCLIDE BEHAVIOR IN BODY OF THE CHERNOBYL ACCIDENT WITNESSES

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INTRODUCTION

As a result of the Chernobyl NPP (*ChNPP*) accident aerosol particles of dispersed nuclear fuel were released to the atmosphere. Inhalation of those aerosol became the source of internal exposure for witnesses of the Chernobyl accident.

To assess correctly internal doses from a mixture of radionuclides present in air in the form of aerosol particles one mast assign each radionuclide to a certain inhalation class by its chemical speciation in aerosol and define the airborne characteristics (the activity median aerodynamic diameter, *AMAD* and the standard geometric deviation, β_g) of that particular aerosol. Moreover, information on any particular radionuclide is useless for other components since, in such a mixture, the radionuclides are generally independent and may belong to different particles.

On the contrary, all nuclear fuel particle (*NFP*) radionuclides belong to the same particle, being matrix-bound (1). The collective behavior of the matrix-bound radionuclides in the environment and in the human barrier organs makes it possible to spread to the aerosol of NFP any estimates of AMAD and β_g obtained for any particular NFP radionuclide. This is principal feature of NFP aerosol as distinguished from a mere mixture of aerosol particles carry different radionuclides.

The information on the aerodynamic size of ChNPP in-site aerosol of nuclear fuel particles for the first postaccident days is absent. The only way to obtain it is to use the indirect methods based on the investigation into behavior of NFP radionuclides in body of persons who inhaled that aerosol in the time of the accident. With this aim two groups of ChNPP staff and firemen that witnessed the accident were investigated in 1987-1992 and one group of Gomel inhabitants was investigated in 1991-1993 (2).

MATERIAL AND RESULTS

Two groups of ChNPP staff and firemen that witnessed the accident were investigated. Analysis of ^{137}Cs kinetic in body (Method I) and analysis of ^{239}Pu distribution in body (Method II) were used for evaluation the AMAD of inhaled aerosol.

Out of the workers involved in operations on 26-27 April 1986 and measured with a semiconductor whole-body counter, the first examined group was chosen. It consisted of 264 nuclear fuel particle carriers examined at least three times from 16 May 1986 to 16 November 1986. The measurements were made by officers of the Radiation Monitoring Department in the town Chernobyl. Out of this group, we chose a few tens of individuals

that even 700-800 days after the accident had the ^{137}Cs content of the body in excess of $100\,nCi$, i.e., the average 1987-1988 value for the ChNPP personnel unemployed in emergency operations in April-May 1986. The results of intravital studies on the ^{137}Cs contamination of members of this group make up data base to be used for validation of NFP size.

In NFP cesium acquire properties of uranium oxides, normally unusual for it (1). Following intake by inhalation with NFP aerosol, biokinetics of ^{137}Cs appears to be affected by the particle size, though commonly occurring Cs compounds belong to ICRP inhalation class D (3), with kinetics in the body being, therefore, independent of aerosol particle size. Relationship between the effective half-time of matrix-bound ^{137}Cs clearance from the body of the Adult Reference Man, $T_{ef}(^{137}Cs)$, and the AMAD of nuclear fuel particles was found using the computer code *"R-MAN"* (4). As the AMAD of an aerosol increases from 1.0 to $50.0\,\mu m$ (with a constant β_g assumed to be 3.0), the value of $T_{ef}(^{137}Cs)$ drops from 600-500 to 130 days. Increased dimensions of aerosol particles change the contribution of different processes of the NFP clearance from the respiratory system to the general process of the matrix-bound ^{137}Cs retention in the body. When an aerosol with an AMAD of about $1\,\mu m$ is inhaled, the bulk of particles is deposited in the pulmonary region and the radionuclides enter body fluids through pulmonary communications where radioactive material of NFP is retained for long time, due to particle destruction. The larger particles, the higher the probability of their deposition in the nasal passage and the trachea and bronchial tree; some portion of the radionuclides penetrates body fluids almost without delay, while the bulk transits through the gastrointestinal tract and is removed from the body. In this case, a long-retained activity fraction in the lung is small and the ^{137}Cs elimination half-time appears to be close to 110 days, i.e. the half-time of ^{137}Cs removal following intake by ingestion (3). The elimination half-time of ^{137}Cs in the first group of workers ranged from 230 to 600 days, in full accordance with model predictions. For further analysis, 15 workers were chosen in this group since their legends were most likely to suggest a single intake of NFP aerosol by inhalation within the first postaccident days. For every person examined, the ^{137}Cs content of the body was normalized. The value at the time of intake was taken to be unity, using extrapolation of later individual measurements to the accident date. The normalized data are adequately described by an exponential curve with a half-time of 330 days. This value is three times as high as that recommended by the ICRP (3) to describe the Cs elimination kinetics for Adult Reference Man. According to our model, a half-time of 330 days corresponds to the ^{137}Cs behavior in the body following a single inhalation of nuclear fuel particles aerosol with an AMAD of $16 \pm 2\,\mu m$.

The distribution of the radioactive material in the body also depends on the chemical composition and the particle size of inhaled aerosols. The ratio, $R(Pu)$, of the plutonium content of the lung to that of the remaining organs is a criterion used to assess the AMAD of the relevant aerosol.

The Chernobyl workers and firemen that died of acute radiation sickness in 1986 form the second control group of examined accident witnesses. The results of postmortem studies on the ^{239}Pu distribution in the organs of members of this group make up another data base to be used for validation of NFP size. According to Popov et al. (5), the value of $R(Pu)$ for the workers that died of acute radiation sickness within 90 days after the accident ranged from 0.07 to 10.0. Using that data, we took a mean $R(Pu)$ value of 1.6 for the dead workers and, with the help of the "R-MAN" code, found expected AMAD of

nuclear fuel particles. The estimates proved to be in good agreement with those obtained for the intravital ^{137}Cs behavior in the witnesses. The mean AMAD of fuel particles estimated by $R(Pu)$ is $12 \pm 2\,\mu m$.

CONCLUSIONS

Method II was used for reconstruction of the AMAD of nuclear fuel particles aerosol inhaled by inhabitants of Gomel district, who were affected by the accident (2). That estimates are in a good agreement with the previous one. The results of the reconstruction of the size of NFP aerosol (mean AMAD $\pm$ Standard Deviation of Mean) are as follows:

AMAD, μm	Case specification
16±2	15 alive accident witnesses of ChNPP staff, examined with semiconductor body counter in 1986-87 (Method I)
12±2	23 accident witnesses of ChNPP staff, dead in 90 postaccident days and examined by autopsy (Method II)
12±3	21 accident witnesses of 125 Gomel district inhabitants, dead in 1990-91 and examined by autopsy (Method II) (2)

The obtained estimates for AMAD of nuclear fuel particles are in a good agreement with results of Rudhard et al. (6) on investigation the size distribution of NFP obtained from soil samples collected in South Germany in May, 1986. Autoradiography of that samples showed that they were contaminated with NFPs with median linear diameter of $4.5\,\mu m$, that corresponds to AMAD of 15 μm (for NFP density of 10 g/cm^3). Those values are closed to the dimension of uranium dioxides grains in the fuel pellets of Chernobyl type reactor.

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BIOLOGICAL BEHAVIOUR OF ^{237}Np AFTER INHALATION OF NpO_2 IN THE RAT: PRELIMINARY RESULTS.

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INTRODUCTION

Numerous studies have been reported on the biological behaviour of different masses of ^{237}Np or ^{239}Np administered under various chemical forms and valency states by intravenous or intramuscular injections and by ingestion. In contrast with Pu, some important Np speciation informations are still lacking as well as an appropriate treatment for Np decorporation (1). To our knowledge, only three main studies have concerned the behaviour of Np after inhalation and were confined to soluble forms such as Np-oxalate of unknown valency state and Np (V) nitrate (2, 3, 4). This paper presents preliminary results obtained in rats on the behaviour of Np after inhalation of $^{237}NpO_2$ for which no experimental data is available. This physico-chemical form might be potentially used in the nuclear industry.

MATERIAL AND METHODS

Female Sprague-Dawley rats were used. Two groups of 30 animals were exposed at 3 months of age to $^{237}NpO_2$ aerosols (actual diameter 0.40 μm, σg 2.02) as previously described (5). Four to five animals per group were killed under pentobarbital anaesthezia 7, 21, 48 and 92 days after exposure. The lungs, the liver, the kidneys and the femurs were taken off for α counting by liquid scintillation. This was performed after heat mineralization, solubilization in nitric acid and specific extraction of Np reduced into Np(IV) within the organic scintillation phasis.

At this time, some animals are still alive for further studies up to 300 days after NpO_2 exposure.

RESULTS

The Np lung burden, measured 7 days after exposure were 110 Bq (n=5, sd=10) and 200 Bq (n=5, sd=15) in the 2 animal groups respectively.

Figure 1 shows the evolution of the Np lung burden as a function of time following exposure. The observed curve fits to a single exponential function of time with a half-life at about 68 days.

Figure 2 shows the relative amount of Np retained in the skeleton, the liver and the kidneys as a function of time following exposure expressed as the fraction of the Np lung burden measured on day 7. Maximal retention at about 1 % was observed in the skeleton which remained nearly constant from day 7 to day 92 after exposure. An about two-fold decrease of Np liver retention was observed between days 7 and 21 but this retention remained nearly at a constant value from day 21 to day 92 which corresponded to about 20 times less than the skeleton retention value. The Np kidney retention seemed to gradually decrease by a factor of 2 from 0.003-0.002 % to 0.002-0.001 % from day 7 and to day 92 respectively.

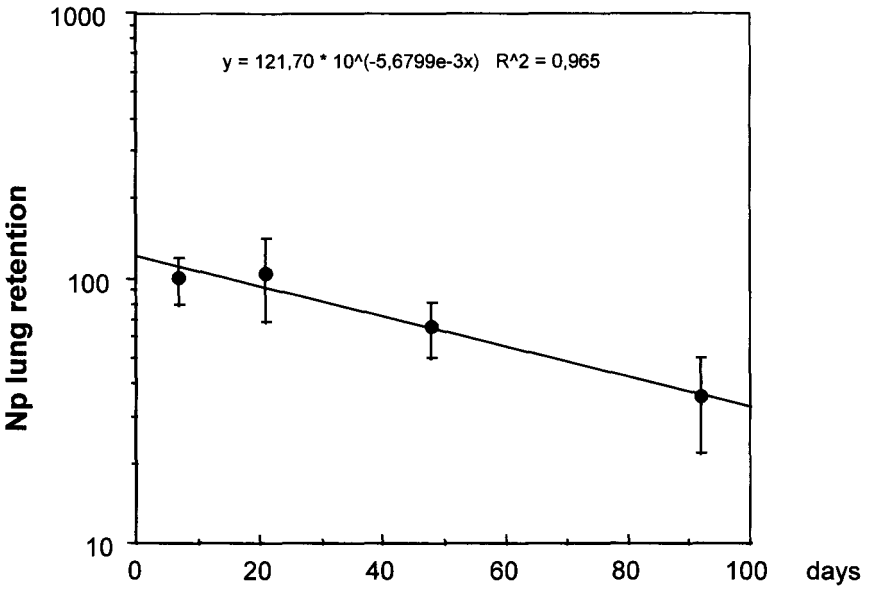

Figure 1: Np lung retention expressed as percent of the retention measured on day 7. Mean values ± sd.

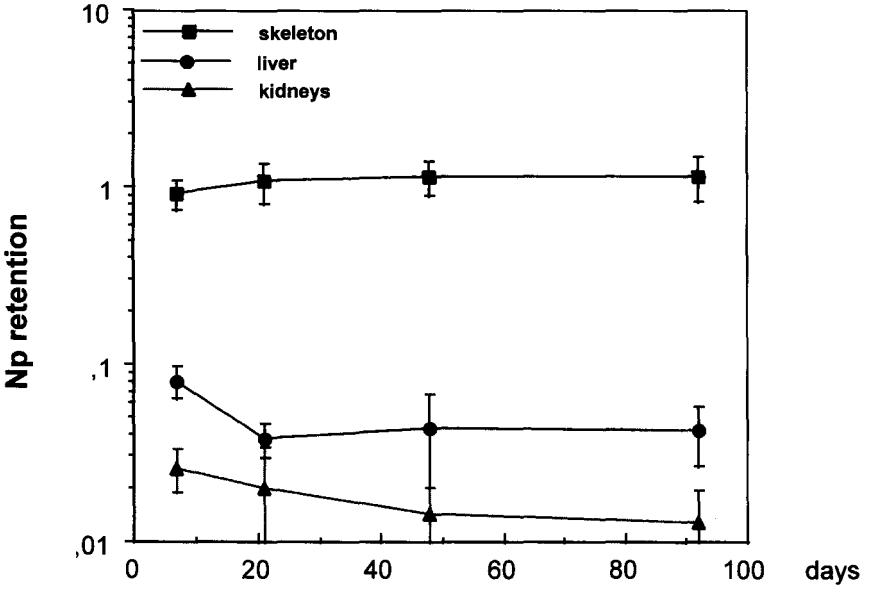

Figure 2: Np retention in different organs expressed as percent of the lung retention measured on day 7. Mean value ± sd.

DISCUSSION

Translocation of Np from inhaled NpO_2 to the other parts of the organism and Np lung clearance appear much slower than those reported for the different Np soluble forms (2, 3, 4). Our results can be directly compared to those obtained in our Laboratory after inhalation of PuO_2 or (UO_2,PuO_2) mixed oxyde containing 20 % Pu which had used the same inhalation device, the same sex, age and strain of rats and a similar α emitter counting method (6).

The half-life of Np retention in the lungs (68 days) was nearly the same as that reported for Pu after exposure to PuO_2 or (UO_2,PuO_2) mixed oxyde (66 days). These Pu exposures were performed at initial lung deposits which are not known to alter lung clearance parameters, especially alveolar macrophage motility involved in the removal of particles by cell migration via the broncho-tracheal tree on the epithelial surface. Thus, our preliminary results suggest that, for the low initial lung deposits studied, between 100 and 200 Bq, $^{237}NpO_2$ is non-toxic for alveolar macrophages and do not alter early lung clearance parameters. Further studies are in progress to characterize these lung clearance parameters as a function of the initial Np lung deposit in order to visualize a potential chemical toxicity of this actinide as compared with Pu.

The fraction of the Np initial lung burden retained in the skeleton was between those reported for Pu after inhalation of PuO_2 and (UO_2,PuO_2) mixed oxyde but close to the Pu value for PuO_2. Moreover, the retention ratio for the skeleton versus the liver appeared very similar at 15 and 20 for Pu and Np respectively. Our results suggest that Np from inhaled NpO_2 might be slightly more translocated to the extrapulmonary compartments than Pu from inhaled PuO_2. Nevertheless, the actinide distributions in the extrapulmonary compartments studied appeared similar for Pu and Np.

In conclusion, these preliminary results suggest that the biological behaviours of Np and Pu after inhalation of NpO_2 or PuO_2 are similar. Further studies are in progress to complete this work and to explain why, in these experimental conditions, Np and Pu have a similar behaviour.

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THE CHEMICAL FORM OF NEPTUNIUM IN THE RAT LIVER CELLS

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INTRODUCTION

The chemical association of the transuranic elements in the liver was investigated more than 20 years ago. The numerous studies realised until now demonstrated the preferential binding of plutonium, americium and californium to lysosomes and to the cytosolic ferritin (1-3). By contrast, few studies were realised with neptunium and no clear association of this radionuclide with any subcellular structure was pointed out. Moreover, the results published until now are in apparent contradiction since some authors reported preferential binding of this radionuclide to lysosomal structures (4-5), while others claimed that the nuclei seemed to be the main binding site (6-7). The reasons involved for these dissimilarities are not yet defined and it could be interesting to test the influence of the initial chemical form of the radionuclide on its intracellular distribution. Moreover, an identification of the biological ligands of neptunium, and particularly their renewal frequency, may enable an estimation of the turnover of neptunium in the cell and should, therefore, give information on internal dosimetry. These informations should give data which could be used for specific decorporation studies. The work reported here aims to identify the biological ligands involved in the binding of neptunium as a function of the chemical form injected.

MATERIALS AND METHODS

The experiments were performed with eight 11-12-week-old, male Sprague-Dawley rats (IFFA Credo, France, 220 ± 25 g) which had free access to food and water throughout the study. These experiments on animals were performed by scientists certified by the French Ministry of Agriculture for carrying out these procedures.

The ^{239}Np was obtained from the Service de Chimie, C.E.A. Bruyères le Châtel (France) and was prepared according to a previously published method (8). Prior to administration, the solutions of Np in 0.1M citric acid and Np in 0.1M nitric acid were diluted in order to inject 17 pg ^{239}Np.kg^{-1}.

Two groups of four rats each were contaminated by intravenous injection of one of the solutions (^{239}Np-citrate or ^{239}Np-nitrate) into the saphenous vein under light anaesthesia with pentobarbital (SANOFI, France).

All the animals were sacrificed 24 hours after contamination by exsanguination under anaesthesia and the subcellular fractions of the rat livers were prepared using a previously published method (7). Briefly, the liver was perfused and then homogenized in ice-cold homogenization medium (20mM Tris-HCl, 0.25M sucrose, pH 7.4) using a Potter-Elvehjem homogenizer (clearance 0.3-0.5 mm, 280 r.p.m.). The homogenate was then subjected to ultracentrifugation at successive accelerations of 1,000g (for 10 min), 10,000g (for 20 min), 40,000g (for 30 min) and 100,000g (for 45 min). In this way, 4 pellet samples and a supernatant liquid were obtained composed of cell nuclei + unbroken cells, mitochondria + lysosomes, membranes, microsomes and cytosol, respectively. The protein content of each fraction was determined by the method of Lowry *et al.* (9). The specific activities of the marker enzymes acid phosphatase (10), cytochrome oxidase (11), 5'-nucleotidase (12) and glucose-6-phosphatase (13) were assayed spectrophotometrically.

For each animal, 5 ml of the cytosol was subjected to gel permeation chromatography. Sephacryl HS-300 was used in a column 85 cm long x 2.6 cm in diameter, surrounded by a water jacket at about 4°C. The column was operated at a flow rate of approximately 40 ml.h^{-1} and was equilibrated with 0.2 M NaCl, 20 mM Tris-HCl, pH 7.4. After elution 3.5 ml fractions were collected and analysed for absorbance at 280 nm and for neptunium content. The radioactivity of each fraction collected was determined by gamma-ray analysis.

RESULTS AND DISCUSSION

Table I lists the subcellular deposition of neptunium in the rat liver, according to the chemical form administered. This shows that nearly-half of the ^{239}Np was distributed in the soluble fraction whereas the nuclear and mitochondria + lysosomal fractions accumulated 20-27% and 16-18%, respectively, of the neptunium. The initial chemical form (citrate or nitrate) of the radionuclide has little or no effect on its intracellular distribution. The main effects were observed for the quantitative distribution of neptunium in the liver, since the organelles were three more times contaminated (in terms of Bq/mg Prot.) after injection of Np-citrate than after Np-nitrate (Table I). Nevertheless, when the results are expressed in term of Relative Specific Activities (RSA), the

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This work was funded by the Nuclear protection and Safety Institute (IPSN, France)

differences between Np-citrate and Np-nitrate were not statistically significant. This is somewhat surprising since all the previous studies realised until now pointed out the importance of both the chemical form and the valency state of the radionuclide on its biodistribution (14). One explanation for the results reported here could be that the element neptunium is distributed from blood to liver bound to one protein, i.e. in a single chemical form. Blood analysis by means of ultrafiltration and liquid chromatography showed that less than 10 % of the blood Np was ultrafiltrable and that nearly all the radionuclide was bound to transferrin (15), whatever the initial chemical form injected. This fact implies that a reduction of the element from the pentavalent form to the tetravalent state occurs *in vivo* after systemic administration, but the mechanisms involved are not yet explained.

		Nuclear	Mito/Lyso	Membrane	Microsomes	Cytosol
^{239}Np- citrate	Np (%)	20.5 ± 3.0	18.5 ± 6.2	8.0 ± 1.7	3.5 ± 0.9	49.6 ± 7.4
	Bq/ mg Prot.	94 ± 57	230 ± 141	258 ± 170	212 ± 149	374 ± 238
	R.SA. [(b)]	0.44 ± 0.07	1.18 ± 0.36	1.26 ± 0.35	0.97 ± 0.26	1.79 ± 0.31
^{239}Np- nitrate	Np (%)	26.6 ± 5.0[(e)]	16.0 ± 2.2	4.5 ± 1.7[(d)]	2.1 ± 1.2[(e)]	50.8 ± 1.8
	Bq/ mg Prot.	29 ± 11[(d)]	81 ± 34[(e)]	68 ± 31[(d)]	45 ± 46[(d)]	118 ± 42[(e)]
	R.SA. [(b)]	0.48 ± 0.05	1.33 ± 0.12	1.1 ± 0.18	0.77 ± 0.41	1.97 ± 0.13
	Ac. Phosphat.[(c)]	0.68 ± 0.17	2.42 ± 0.54	2.11 ± 1.39	1.19 ± 0.58	0.75 ± 0.21
	Cyt. Oxydase[(c)]	0.68 ± 0.12	3.48 ± 0.86	2.30 ± 0.81	2.23 ± 1.34	0.08 ± 0.06
	5' Nucleotid.[(c)]	1.26 ± 0.25	1.13 ± 0.51	3.38 ± 1.92	2.12 ± 1.62	0.07 ± 0.03
	G-6-Phosphat.[(c)]	0.86 ± 0.19	1.64 ± 0.81	4.31 ± 2.54	3.39 ± 1.47	0.29 ± 0.26

Table I. The distribution of total cellular neptunium and marker enzymes in the fractions isolated (mean ± S.D.)[(a)]

(a) : four animals per group
(b) : RSA : Np (%) / Prot (%)
(c) : Mean for 8 determinations
(d) : significantly different from ^{239}Np-citrate. t test. p = 0.05
(e) : significantly different from ^{239}Np-citrate. t test. p = 0.1

The intracellular distribution of neptunium reported in table I is somewhat different from that described by several authors, who claimed that the lysosomal structures within the liver cells were the main storage site for all the transuranic elements (1-3). The cytosolic accumulation of neptunium noticed in our study was higher from what was observed elsewhere and an attempt was made to determine the cytosolic ligands involved in the binding of the element. Fig. 1. represents the distribution of neptunium amongst proteins of the soluble fraction after gel permeation chromatography. The measurement of absorbance showed the elution of 3 main classes of proteins with MWs of 450 kd, 130 kd and 40 kd, respectively. The recovery of neptunium in the eluent was 90-92% and, 24 hours after contamination, neptunium eluted mainly in two peaks, whatever the initial chemical form administered. The first peak coincided with the elution of ferritin (15), whereas the second eluted in a region of low spectrophotometric absorbance, corresponding to compound of MW 200 kd. This last compound probably corresponds to the same protein than that described in previous studies and which was responsible for the early binding of plutonium and protactinium (16-17). The identity of this protein remains unknown and Schuppler et al. (17) postulated that this compound might be related to Calmodulin. Further investigations on this protein are needed before conclusions can be drawn.

The reasons for the preferential binding of Np in cytosol instead of lysosomal or nuclear structures remain undefined. Nevertheless, the main experiments realised until now with neptunium were done with high masses of ^{237}Np and it can be postulated that the storage of this radionuclide in the heavier structures of the cells (nuclei and lysosomes) was influenced by the mass injected. This kind of results has been previously noticed by Schuler and Taylor (18) who showed that the mass of the injected Pu in primary cultures of rat hepatocytes influenced its intracellular distribution, the nuclei being the main deposit site when high mass of Pu were used. By contrast, that hypothesis was inconsistent with the work of Seidel et al. (5) who described a lysosomal accumulation of neptunium after administration of low masses of ^{239}Np.

CONCLUSIONS

After systemic administration of neptunium-239 in a citrate or nitrate form, the radionuclide distributed in all the subcellular structures of the rat liver. Nevertheless, the early distribution occurred mainly in cytosol, where two proteins with MWs 450 kd and 200 kd, respectively were involved. The first protein was identified as cytosolic ferritin, but the identity of the second remains unknown.

The initial chemical form of the radionuclide had no significant effect on its intracellular distribution. This fact probably implied that neptunium was reduced into one single chemical form when administered in blood, and is consistent with the assumption that the circulating Np was bound to blood transferrin.

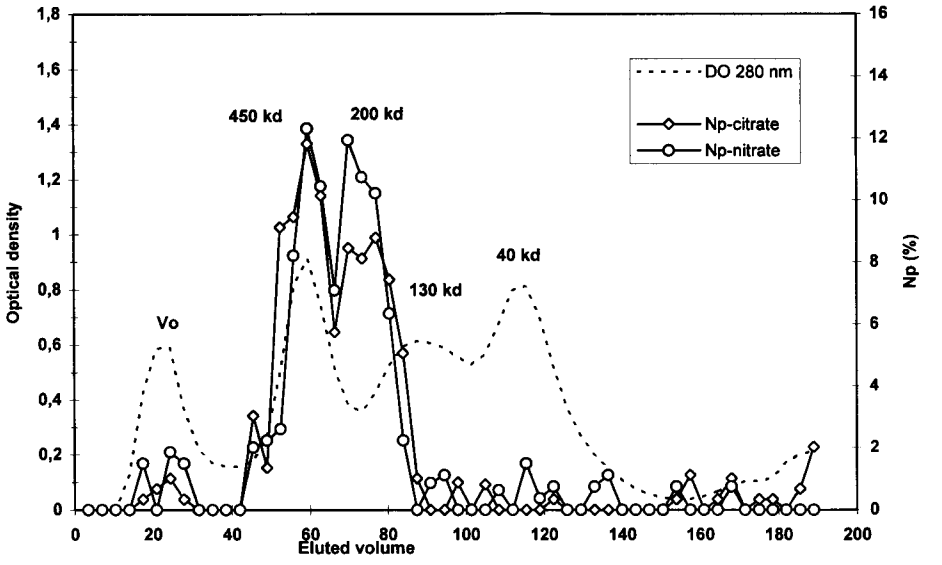

Fig. 1. The elution profile of ^{239}Np-citrate and ^{239}Np-nitrate after chromatography on Sephacryl HS-300 of cytosol of rat liver cells.

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THE MEASUREMENT OF ^{241}Am IN THE BODY WITH A DOUBLE LOW ENERGY GERMANIUM DETECTOR SYSTEM

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ABSTRACT

A system based on two low energy germanium detectors has been calibrated for the investigation of internal contamination with ^{241}Am. The two detectors have a total active area of 5775 mm^2 and a resolution of 600 eV at the energy of 59.5 keV. The measurement system was calibrated with the realistic torso phantom of Livermore which was obtained from the IAEA. The detection efficiency was determined in real measurement conditions with the two detectors above the right lung. Assuming a standard tissue thickness of 25 mm, a detection efficiency of 0.083 cpm/Bq was found. For a counting time of 50 minutes this corresponds to a detection limit of 6 Bq of ^{241}Am. Applying the new ICRP lung model and assuming a typical Pu(MOX) mixture with 1.2 % (w) ^{241}Am, an inhalation corresponding to an internal dose of 20 mSv is measurable 30 days after the intake for a class M compound. In case the americium of the Pu(MOX) mixture can be classified as slow (S), a single intake corresponding to a dose of 20 mSv is detectable over a period of about six months. The subject specific background is evaluated from a background prediction region above the peak region. This system allows a fast response in case of accident. The detectors are also calibrated for the measurement of americium and plutonium in the liver, for uranium and americium in the tracheo-bronchial lymph nodes and for plutonium in the lungs.

INTRODUCTION

The in-vivo measurement of lung contamination with actinides has always been a difficult task because the low energy X-rays emitted by most of the nuclides are strongly absorbed by the tissues. Since two years, a system using two low energy HPGe detectors has been used in the SCK•CEN for this type of measurements. This system greatly improves the capabilities for the quantitative assessment of actinides in the lungs, the liver and the tracheobronchial lymph nodes. The two crystals having a total active area of 5775 mm^2 are placed over the right lung and positioned with an angle between them of about 30 degrees. The technique is based on the quantification of ^{241}Am associated with plutonium in MOX fuel particles. The counting room is made of steel plates 200 mm thick lined with thin plates of Pb (3.2 mm), Sn (1. mm) and Fe (0.5 mm).

CALIBRATION

The efficiency of the counting system has been determined with the aid of a torso phantom of the Livermore type (lent by IAEA). The calibration is made for the assessment of ^{241}Am and of pure ^{239}Pu in lungs, in liver and in tracheo-bronchial lymph nodes. Different overlayers were used for the simulation of the different morphological types. These measurements show an influence of the tissue thickness, tissue composition and distance torso-detector on the total efficiency of the system. An increase of 20 mm in the distance torso-detector results in a decrease of the efficiency of about 20%. The efficiency of our system with two detectors for a man with a tissue thickness of 25 mm is 0.083 cpm/Bq. For a counting time of 50 minutes this corresponds to a detection limit of 6 Bq of ^{241}Am. The influence of the tissue thickness on the counting rate is shown in figure 1.

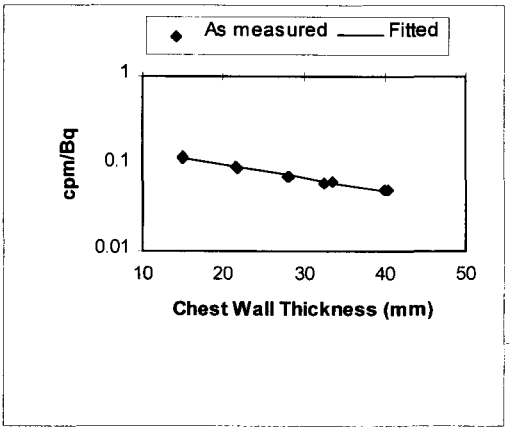

Figure 1. Decrease in counting rate with increasing chest wall thickness of a dual germanium system for the assessment of ^{241}Am in the lungs. The full line is obtained by a least square fitting including all observed values.

ANALYSIS PROCEDURE

A dispersion of 0.2 keV/channel is used for the counting. The examined zone between 59.2 and 60.0 keV, is compared to the zone between 61.4 and 64.2 keV used for the background calculation. The zone chosen for the background assessment is normally free of photopeaks in case of contamination by another radionuclide so that false negative results are minimized. Uranium, in case of contamination with mixed oxide fuel particles, because of the relative proportion of each element, will produce a negligible contribution in the 63 keV region as compared with ^{241}Am.

APPLICATION

A common bioassay tool for Pu(MOX) workers is chest counting for the ^{241}Am progeny of ^{241}Pu. An estimation of the intake requires working assumptions about the isotopic composition of the inhaled mixture and the lung's clearance dynamics. Many factors have to be considered:

- Particle size of the material: From a recent survey of the literature (1) an AMAD of 5 µm and a median geometric standard deviation of 2.56 are realistic default values for occupational exposure when the particle size distribution is unknown.
- Chemical form of the source material: The ICRP-30 and ICRP-68 designate ^{241}Am as class W or class M material. However, a minor contaminant of a matrix may exhibit the behaviour characteristics of the matrix. It could be assumed that the ^{241}Am is of class Y or S such as the plutonium in the Pu(MOX) matrix.
- Metabolic retention and dosimetric model: The new ICRP-66 lung model and ICRP-67 systemic model for Pu and Am provide a basis for the retention and dose assessments. For the density, ICRP-68 uses a default value of 3. In this work, a more realistic density of 10 is used, keeping all the other parameters of ICRP-68 (e.g. breathing rate of 1.2 m^3/h, resting for 31.3 % of the time and light work for 61.7 %, size factor of 1.5).
- Isotopic composition: In this work, a standard isotopic composition (in weight) is assumed: 1.2% ^{238}Pu, 61.1% ^{239}Pu, 23.5 % ^{240}Pu, 8.8% ^{241}Pu, 4.2% ^{242}Pu and 1.2 % ^{241}Am. The doses due to all isotopes of the mixture vary with the solubility class when 1 Bq ^{241}Am is inhaled:

	1 µm (mSv/Bq(^{241}Am))	5µm (mSv/Bq(^{241}Am))
Class M	0.23	0.13
Class S	0.20	0.11

Defining Q(20) as the measured activity in the lung that gives an effective dose of 20 mSv when

we assume the above-mentioned isotopic composition of the mixture:

Q(20)(Bq) = (20 mSv/Dose coefficient (mSv/Bq(^{241}Am)))•Fraction remaining in the lung.

We found:

	^{241}Am class S (Bq)	^{241}Am class S (Bq)	^{241}Am class M (Bq)	^{241}Am class M (Bq)
Time (d)	1μm	5 μm	1 μm	5 μm
0.5	16.4	12.9	12.8	9.8
1	16.0	12.5	12.4	9.5
2	15.6	12.2	12.1	9.2
3	15.5	12.1	11.9	9.1
4	15.3	11.9	11.8	8.9
5	15.2	11.8	11.6	8.8
6	15.1	11.7	11.5	8.6
7	14.9	11.6	11.3	8.5
15	14.0	10.7	10.2	7.6
30	12.6	9.4	8.5	6.2
45	11.6	8.5	7.3	5.2
90	9.8	7.1	4.9	3.5
180	8.3	6.0	2.7	1.9
360	6.9	5.0	0.9	0.6

This table shows that, for class S and an AMAD of 5 μm americium, an effective dose of 20 mSv could be assessed up to 180 days after an acute intake (inhalation), when the detection limit is 6 Bq. The situation is less favourable for a class M compound but better for an older Pu(MOX) mixture.

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CHELATING AGENT, DTPA, CAN NOT REMOVE EFFECTIVELY INHALED-PLUTONIUM OXIDE IN RATS

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INTRODUCTION

Chelation therapy is a useful method to reduce the cancer risk in the patients contaminated with transuranic elements such as plutonium and americium. Chelating agents, calcium and zinc salt-diethylenetriaminepentaacetic acid (Ca-DTPA and Zn-DTPA) are useful drugs to remove plutonium and americium effectively from the body with almost no toxicity. Although these chelating agents combine with radionuclides of ionic chemical forms such as nitrate and citrate, the effect is not expected for the insoluble form, i.e. oxide. If patients will be contaminated with plutonium oxide through inhalation, it is doubtful whether each DTPA therapy is useful or not. The present study is to clarify whether these DTPAs are useful to remove plutonium oxide from the body or not.

MATERIALS AND METHODS

Twenty female Wistar rats, 3 months of age, inhaled the particles of plutonium oxide generated by the radioactive aerosol exposure system for rodents developed in our institute (1). The inhaled dose was determined by a whole body counter for small animals (2). They were divided into four groups of five each; Group 1(G1): DTPA administration was initiated on 1st day after inhalation, Group 2 (G2): on 7th day, Group 3 (G3): on 14th day and was not administered in the no-treatment group. Chelation therapy was carried out according to the following schedule. Ca-DTPA was injected intraperitoneally with a daily dose of 150 μmol/kg for the first 5 days and subsequently Zn-DTPA was administered orally at the same dose as that of injection in drinking water up to 30 days. The 24-h urine and feces were collected for 1 day after the first Ca-DTPA injection (Table 1). The lung retention in each group was measured using a whole body counter during the experiment period (Table 1).

Table 1 Experimental design

Group	Inhalation 1	7	14	21	30	38	44days
No treatment group	WBC #	WBC	WBC	WBC			WBC+
Group 1	WBC *, #	WBC			WBC+ *		
Group 2		WBC *,#	WBC			WBC+ *	
Group 3		WBC	WBC *,#	WBC			WBC+ *

WBC: whole body count, *: chelation therapy , #: collection of urine and feces, +: sacrifice

The rats in each experimental group were killed after receiving chelation therapy for 30 days and the no treatment group was time-matched to G3. The blood was collected and then the organs such as lung, trachea, liver, kidney and femur were removed. The plutonium concentration in the blood, organs and excreta was measured by a liquid scintillation spectrometry after a wet ashing treatment.

RESULTS

The initial lung burdens in G1-3 were not different from that in the no treatment group. There were no significant differences in the lung retention of plutonium with the group (Fig.1).

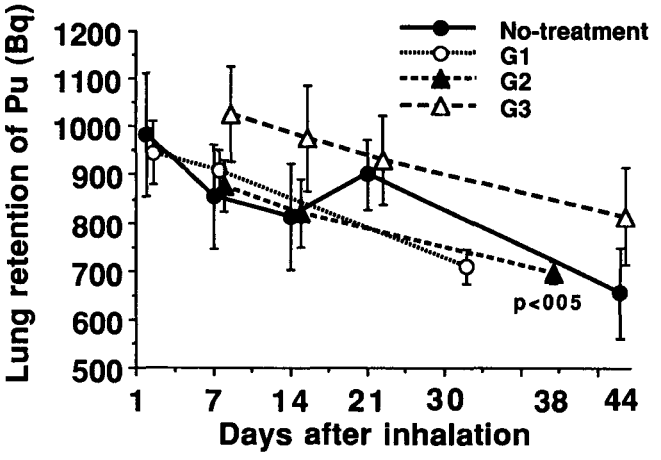

Fig. 1 Lung retention by a whole body counter in each group

The plutonium contents in each organ as well as blood, serum, spleen and ovary were not different between the no-treatment group and each group (Table 2). Also, the excreted plutonium in the urine and feces after first Ca-DTPA injection was not different between the no-treatment group and each treatment group.

Table 2 Plutonium contents (total Bq) in various organs

Organ/Group	No-treatment Group	Group 1	Group 2	Group 3
Lung	772±105	872±50	929±44	1060±144
Trachea	1.20±0.07	1.40±0.25	1.42±0.17	1.12±0.23
Liver	14.1±0.6	17.1±1.0*	15.6±2.5	13.0±1.2
Kidney	2.28±0.25	1.92±0.20	2.04±0.07	1.76±0.12
Femur	0.83±0.02	0.82±0.03	0.88±0.03	0.75±0.04

* significantly different from no-treatment group ($p<0.05$)

DISCUSSION

As an effective method of chelation therapy, Ca-DTPA injection for the first few days after contamination and subsequently the oral administration of Zn-DTPA is generally appreciated. The dose of 150μmol/kg of DTPAs administered here is more than the recommended human dose of 30 μmol/kg/day, but was effective to remove plutonium nitrate with almost no toxicity for rats in our previous study (3). The particles of plutonium oxide deposited in the lung, although insoluble, may change to be soluble, because the activity is determined in the serum, various organs and excreta. Therefore, the chelation therapy for contamination with plutonium oxide of the insoluble form, although slight and gradually,

may be expected to the effectiveness even if the initiation is delayed but the period is long. Although DTPA administration should be initiated immediately after contamination with the soluble form of plutonium to obtain high effectiveness.

We obtained no evidence that chelation therapy using Ca-DTPA and Zn-DTPA can remove plutonium effectively from the body in spite of initiation of chelation therapy and high dose administration for 30 days. Therefore, other treatments such as lung lavage are necessary to decrease the cancer risk with internally contaminated plutonium

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EFFECTS OF NEW CHELATING AGENT, CBMIDA AND ITS ANALOGUES, ON REMOVAL OF PLUTONIUM IN RATS

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INTRODUCTION

Chelating agents are important for removal of radionuclides for reduction of the risk of radionuclide-induced cancer. DTPA (diethylenetriaminepentaacetic acid) is, at present, the most effective chelating agent to remove plutonium and other actinides from the human body. However, the development of new chelating agents that are more effective with lower toxicity than DTPA is required for chelation therapy. We developed CBMIDA [catechol-3,6, bis(methyleiminodiacetic acid)] that can remove plutonium especially from bone, more effectively than Ca- and Zn-DTPA (1) and the side effects are lower or similar to those of Zn-DTPA in animals (2). Since CBMIDA was later found to have an untoward action to accumulate Pu in the kidneys, some analogues of CBMIDA were developed. The present study was performed to clarify the effect of CBMIDA and four kinds of analogues on removal of plutonium in rats.

MATERIALS AND METHODS

Thirty male Wistar rats, 3 months of age, were injected intravenously with plutonium (5.6 x 10^3 Bq of ^{239}Pu in 0.12ml of 0.008M sodium citrate solution, pH adjusted to 7.2) after bring anesthetized with a combination of ketamine hydrochloride and xylazine. The plutonium-injected rats were divided into six groups, each consisting of five animals: those receiving

(1) CBMIDA,
(2) N-(4-carboxypheylcarbamoylmenthyl)-N-phosphonomethyl glycine (C-I),
(3) 2-hydroxy-3-(N-phosphonomethyl-N-carboxymethylaminomethyl)-5-carboxy-N-phosphonomethyl-N-carboxymethylbenzenzylamine(C-II),
(4) N-carboxymethyl-N-(carboxypropylcarbamoyl-methyl)-2,3-dihydroxy-5-carbomethoxybenzylamine (C-III),
(5) 2,3-dihydroxy-N,N'-di carboxymethyl-N,N'-di-(o-carboxyphenyl-carbamoylmethyl)-1,4-benzendimethan amine (C-IV) , and
(6) physiological saline.

Each chelating agent was dissolved in distilled water and adjusted to pH 7.2 with sodium bicarbonate to make the solution 0.2 ml against the rat mean body weight of 350 g. Each rat in each group was daily given intraperitoneal injections with a dose of 100 μmol/kg CBMIDA and its analogues for 2 weeks, beginning at 1 hour after plutonium injection on the first day of treatment.

Animals were killed 2 weeks after the plutonium injection. The plasma, bone(femur), liver, kidney, spleen and testis were removed and the plutonium concentration was measured by an alpha liquid scintillation counter.

RESULTS

Table 1 shows the plutonium contents (Bq/g or Bq/ml) of the organs and plasma. There were significant decreases in the bone, liver and testis in the CBMIDA group, compared to those in the control group. There were significant decreases in the bone, liver and spleen in the C-2 group, and the bone and liver in the C-3 group. There was a significant decrease in the bone, and liver but increase in the kidneys in the C-4 group.

There were no significant differences in the body weight of rats before and after the experiment in any group. During the experimental period, no unusual clinical findings were observed. No untoward findings at autopsy were observed.

Table 1 Plutonium content (Bq/g or ml) in the organs and plasma

Group	Femur	Liver	Kidney	Spleen	Testis	Plasma
Control	159.9±15.5	23.0±3.0	15.4±1.3	34.1±7.4	7.8±0.5	4.0±0.1
CBMIDA	71.1±16.6*2	9.5±2.6*1	25.2±5.2	23.1±7.9	4.0±0.6*2	4.6±0.2
C-1	151.2±10.8*1	23.5±1.2	13.2±0.9	23.1±2.8	8.2±1.0	4.0±0.1
C-2	95.5±14.1*1	11.3±1.4*1	12.3±0.9	17.0±2.5*1	6.4±0.6	4.9±0.2
C-3	98.5±14.6*1	11.6±2.0*1	14.4±1.8	18.5±2.5	6.5±1.0	4.6±0.2
C-4	58.4±11.4*2	15.1±2.3	83.0±8.7*3	20.3±3.1	7.9±0.6	4.4±0.2

Values are means ± standard error. Significant difference from control:*1($p<0.05$), *2($p<0.01$).

DISCUSSION

We have demonstrated that CBMIDA can remove plutonium particularly from the bone and liver, which are the target organs, more effectively than Ca-DTPA or Zn-DTPA, either by the injection or oral administration (1, 3), suggesting that the chelating action of CBMIDA is stronger than that of DTPAs. In this experiment, the plutonium content in the bone was also reduced significantly in the CBMIDA group. Because CBMIDA increased the osteoid volume and thickness, its effect to reduce plutonium deposition in bones is attributed to the inhibition of the mineralization of the bone (4). On the other hand, the plutonium content in the kidneys increased, not significantly from that in the control group, and that in the C-4 group increased in the kidneys. These findings indicate that CBMIDA and C-4 can not be used for chelation therapy.

In the C-2 and C-3 groups, the plutonium content was decreased significantly in the bone and liver, and tended to be decreased in the kidneys, spleen and testes. These compounds improved the untoward action of CBMIDA in which the plutonium content increased in the kidneys. No unusual findings were produced by CBMIDA and its analogue groups even at the dose of 100 μmol/kg. All analogues of CBMIDA had the same toxicity as CBMIDA or lower toxicity, but further examination will be necessary on the toxicity, because a dosage of 100 μmol/kg is more than the recommended dose, for example, the daily dose of DTPA for humans is 30 μmol/kg.

In conclusion, the injection of compounds of C-2 and C-3 of CBMIDA analogues effectively removed plutonium from the body. Therefore they should be examined as drugs for chelation therapy in persons contaminated with plutonium.

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MICRODOSIMETRY OF HOT PARTICLES IN LUNG INTERSTITIUM

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The radiobiological significance of inhaled local α-emitting particles ("hot" particles) was the central point for number of researches. We evaluated the relative carcinogenic effectiveness of "hot" particles with the help of microdosimetry. According to microdosimetry concept the dose-dependent microdistribution of deposited energy in irradiated tissue, $f_D(z)$ is the physical base for prediction of stochastic tissue response on irradiation of different quality. The method of moments was used to approximate $f_D(z)$ in case of α-irradiation of lung interstitium (*LI*) (1). Two sets of sources were investigated. The first one is the set of randomly distributed stationary local emitters ("hot" particles). The second one is the set of α-emitters spread uniformly in LI (the case of uniform irradiation of the tissue). The penetration of α-particles through LI was examined with computer model of 3-d stochastic structure of lung interstitium (2). For the purposes of this model one describe the lung alveoli as a spherical shell with fluctuated inner diameter and thickness of wall.

The term of dispersion of $f_D(z)$, which is affected by spatial correlation of points of α-decay was found (3). For the set of local emitters that term depends on their average activity, energy of α-particles and spatial structure of irradiated tissue. According to our results the spatial microdistribution of α-emitters in irradiated LI affects the dose-response relationship for radiation induced cancer at low doses. In some instances at the same organ dose high local concentration of α-radioactive material in "hot" particles found greater effective carcenogenically than α-emitters spread uniformly. That fact conflicts with judgement of ICRP Recommendations 1990 (4) about relative hazard of the nonuniform irradiation.

The quantitative definition of notion "hot particle", based on microdosimetry approach is proposed.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE PuMA - An improved computer code for assessing plutonium exposure

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MAJOR SCIENTIFIC TOPIC NUMBER 3.1 (see page 7)

ABSTRACT (See instructions overleaf)

Experience gained in developing and using the PLUTO[1] program for assessing plutonium exposure from urinalysis data has been applied to produce an entirely new code PuMA:- Plutonium Maximum-likelihood Assessment.

PuMA allows the assessment of complex plutonium exposure histories from urine sample data sets of variable reliability and containing large numbers of samples recorded as being below the minimum detectable activity. During it's development the PLUTO program was moved from a mainframe to a PC environment and this left it with an outdated interface. PuMA is being developed for the PC and this has permitted the development of a new user-friendly interface. The increased processing power now commonly available on PCs has enabled a more sophisticated implementation of the maximum-likelihood fitting technique to be utilised in the new program. A modular approach to program design will make it easy to implement new biokinetic models and allow future adaption to assess exposure to other nuclides.

(1) Riddell,A.E., Britcher,A.R., *PLUTO - A Software Package Using the 'Maximum Likelihood Method' to Fit Plutonium in Urine Data to an Excretion Function.* Rad. Prot. Dosim. **53**, 199-201 (1994)

BIOKINETIC MODELLING OF SKELETAL LABELS FROM PU-239

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INTRODUCTION

Biokinetic models of the radionuclide distribution in the human body are not only useful for estimating intakes from excretion analysis or calculating committed effective doses. They also may be used for modelling and predicting intra- organ distributions in such cases, where a non- uniform deposition is a characteristic feature of an organ's anatomy and physiology, and such a non- uniformity essentially determines the radiation detriment. This is the case for the deposition of the actinides in the skeleton.

In the following a biokinetic model of the distribution of ^{239}Pu in the human skeleton after systemic intake is discussed. With regard to the non- skeletal parts the model is identical to the one proposed recently by the International Commission of Radiological Protection (1). The skeletal compartments were rearranged to achieve agreement with basic autoradiographic data obtained from animal experiments and human injection cases.

MODEL STRUCTURE

The skeletal compartments of the model are shown in Fig. 1. According to our present knowledge, actinides are bone surface- seekers and only deposit in bone volume by way of new bone formation at remodelling sites. The concentration of these volume labels is determined by the radionuclide concentration in blood at the time of mineralization of the organic matrix. Therefore the compartments representing volume labels are directly connected to the blood (transfer) compartment. Of the radioactivity in blood 43% enters the skeleton with partitioning among the skeletal compartments as shown in Fig.1. This partitioning of the flow from blood to skeleton can be calculated using the concept of affinity ratios. One is the cortical/trabecular affinity ratio (q_{ct}), which specifies the ratio of the fractional flow per unit bone surface to trabecular and cortical bone compartments. The other is the resting/forming affinity ratio (q_{rf}), which specifies the ratio of the fractional flow per unit bone surface to forming (new bone) or resting bone surfaces. Both ratios can be determined autoradiographically and have been chosen as 1/3.

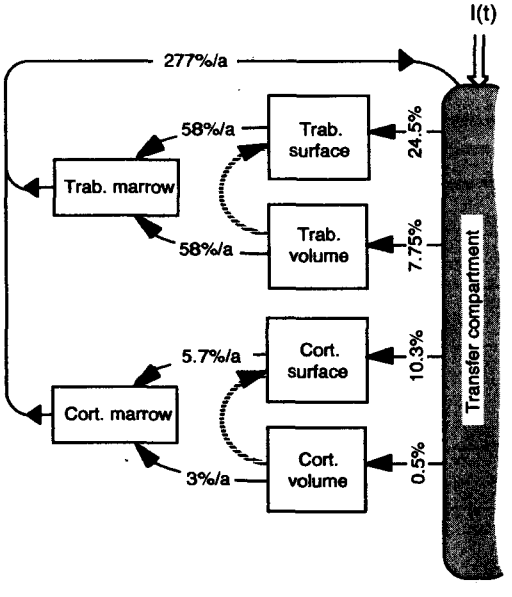

Figure 1. Skeletal compartments of biokinetic model.

Although the turnover rate of the trabecular bone volume is only 26%/year, it can be shown that surface turnover rates are higher (58%/year). Trabecular volume labels are resorbed at the surface turnover rate, cortical volume labels are resorbed according to the volume turnover rate (Fig.1). The model may take into account "internal recirculation", i.e. the direct re-deposition of radioactivity resorbed from volume labels onto neighbouring surfaces (dashed arrows in Fig.1) without entering the blood compartment. However, as specific data from experiments are lacking, this feature presently is not being used.

The question of how much of a surface- seeker remains on bone surfaces after extended burden times is of importance with regard to the relative toxicities of surface- and volume- seeking radionuclides. Volume- seekers such as ^{226}Ra are generally less toxic, because mineralized tissue shields much more of their radiation from reaching endosteal target cells, than is the case for surface- seekers. One can show that an upper limit for the fraction of the trabecular activity (F_{vol}) deposited in volume labels is given by:

$$F_{vol} < \frac{1}{1 - q_{rf} + \frac{q_{rf} MWT\, S_{vt}}{F_{vt}\, \sigma_R}} \qquad (1)$$

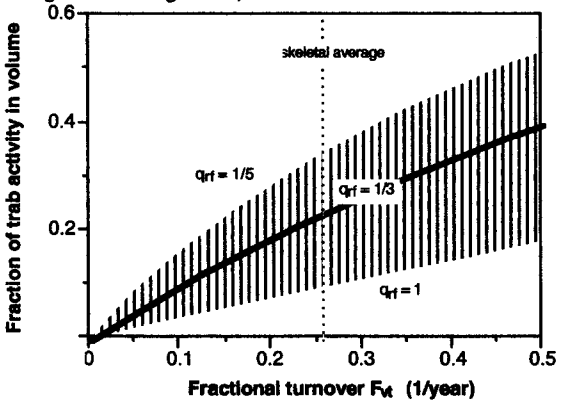

Figure 2. Fraction of total trabecular radioactivity of ^{239}Pu which is deposited in bone volume.

The mean wall thickness (MWT) and the surface/volume ratio (S_{vt}) are two morphological parameters of bone structure. F_{vt} is the (volume-) turnover rate of trabecular bone. F_{vol} decreases with increasing resting/forming affinity ratio (q_{rf}). Fig.2 shows that even for a trabecular turnover as high as 50%/year and a small ratio q_{rf} = 1/5 not more than 50% of the trabecular activity can be in bone volume. For typical values of the human skeleton only about 20% of the ^{239}Pu activity should be found in volume labels. A relationship similar to Eq. 1 holds for cortical bone. The upper limit for F_{vol} in cortical bone slowly increases from about 5% immediately after intake to 15% at 50 years.

LABEL CONCENTRATIONS

In Fig. 3 the average concentrations of ^{239}Pu in trabecular surface labels are plotted for single or continuous systemic intake of 1 kBq. The continuous intake is assumed to occur at a constant rate up to the time read off from the graph. The intake mode has a significant effect on label concentrations with the single intake resulting in lower concentrations than the continuous intake. According to model predictions, the influence of the intake mode on cortical surface labels should be less pronounced. The calculations show that cortical surface concentrations exceed trabecular concentrations after two or five years for single or continuous intake, respectively. Also indicated in Fig. 3 are autoradiographic measurements (2) of trabecular surface concentrations in the axial skeleton of a human injection case (3) who died 17 month after injection. Fig. 3 shows the concentration averaged over all existing surfaces. As formation and resorption sites do not have a surface label, average concentrations over sites where a surface label really exists are obtained by multiplying values of Fig. 3 with the factor 1.11. Cortical surface concentrations reach a maximum of about 4 mBq cm^{-2} (for 1 kBq intake) at 10 or 20 years after intake, for single or continuous intake, respectively.

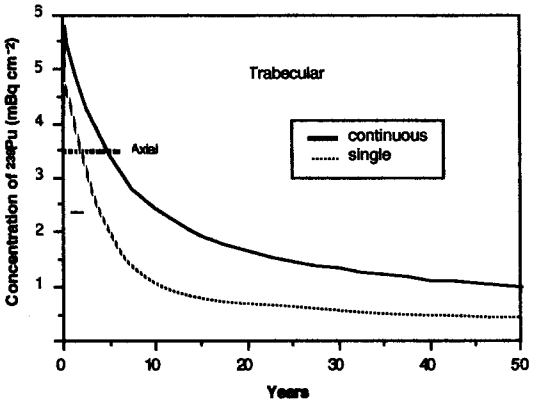

Figure 3. Concentration of ^{239}Pu on trabecular bone surfaces after single or continuous systemic intake of 1kBq.

Assuming random remodelling of bone structure, the fraction of bone surfaces carrying an adjacent volume label (f_{vol}) can be calculated according to

$$f_{vol}(t) = 1 - (1 - \sigma_F F_s)\, e^{-F_s t} \quad (2)$$

where σ_F is the bone formation interval and F_s the surface turnover rate. Eq. 2 may be used to estimate that 99% of all trabecular or cortical surfaces will have an adjacent volume label after 7.8 years or 80 years, respectively. As Fig. 4 shows, cortical volume labels are not influenced very much by the uptake mode. Concentrations of ^{239}Pu in trabecular volume labels are always significantly higher than in cortical labels. The dashed horizontal line in Fig.4 indicates measured concentrations in burial sites of the axial skeleton from the aforementioned human injection case (2).

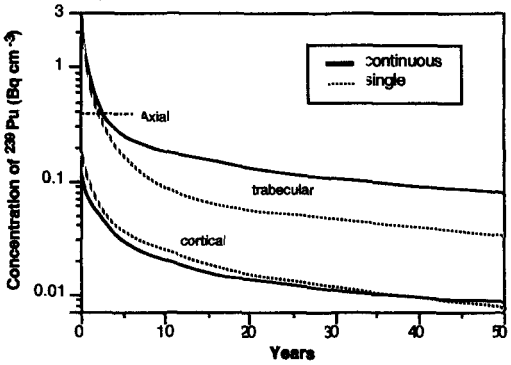

Figure 4. Concentration of ^{239}Pu in bone volume labels after single or continuous systemic intake of 1 kBq.

Because of the rapid clearance of the marrow compartments (Fig.1) they are in quasi- equilibrium with the corresponding volume and surface compartments. This means a nearly constant percentage of 17% of the total trabecular and 2% of the cortical activity is found in the trabecular and cortical marrow, respectively. The dosimetric importance of marrow deposits for the irradiation of the endosteum is small compared to bone volume deposits. Trabecular marrow concentrations are about eight time and cortical marrow concentrations more than two times lower than the respective volume concentrations.

In addition to calculating skeletal averages of surface and volume label concentrations, this model also is able to predict concentrations in formations sites for any chosen time after intake, it quantitatively explains characteristic concentration profiles in formation sites ("buried hot lines") observed in autoradiographs of contaminated human and animal bones, and it describes the kinetic behaviour of labels at individual skeletal sites undergoing the remodelling cycles of resorption, formation and quiescence. One of its major advantages may be its ability to provide estimates for the variability of label concentrations throughout the skeleton.

The main purpose of the model presented here is to provide the basis of a refined dosimetric model for estimating radiation doses to the endosteum and red bone marrow, which are considered to be the principal target tissues of the skeleton.

CONCLUSIONS

- Most of the ^{239}Pu radioactivity in surface labels remains or will be re-deposited on bone surfaces even after extended burden times. There is no complete transformation into volume labels, and thus the radiation risk from actinide depositions remains elevated compared to ^{226}Ra and other bone volume- seekers.
- Intake modes may significantly affect the level of label concentrations and their change in time.
- The model presented is able to describe qualitatively and quantitatively the major features of the microdistribution of actinides as observed in autoradiographic studies of human and animal bones.

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DOES INITIAL LUNG BURDEN AFFECT THE ALVEOLAR LUNG CLEARANCE OF INHALED $^{239}PuO_2$?

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INTRODUCTION

The objective of this study is to establish alveolar lung clearance curves in rats for inhaled $^{239}PuO_2$ of various initial lung burdens (ILB), which are essential for lung dosimetry in the study to define the shape of the dose-response curve.

EXPERIMENTAL

Aerosol generation and inhalation

Stock solution of $^{239}Pu(NO_3)_4$ was treated by the same chemical procedures as described by Raabe et al.(1), which resulted in colloidal $Pu(OH)_4$ suspended in HCl solution. This colloid was nebulized and the resultant droplets were passed successively through a heated tube at 300℃ to dry the droplets and then on to a high temperature furnace heated to 1150℃ to oxidize the dried particles(2). The size of the PuO_2 particles was between 0.4 and 0.5 micron in AMAD and the geometric standard deviation was about 2.

Young adult female Wistar rats were used. The inhalation was conducted using a multiport nose-only exposure chamber without anesthesia. Immediately after the exposure the skin around the nose of the rat was wiped with a sheet of wet thin cloths to reduce surface contamination.

In vivo counting of Pu in rat lung

To follow the lung retention of plutonium, *in vivo* counting of low energy L X-rays (average energy 17 keV) with thin NaI(Tl) scintillation detectors was applied at various intervals up to 1 year after the inhalation (see **Figure 1**). The details were described elsewhere(3,4).

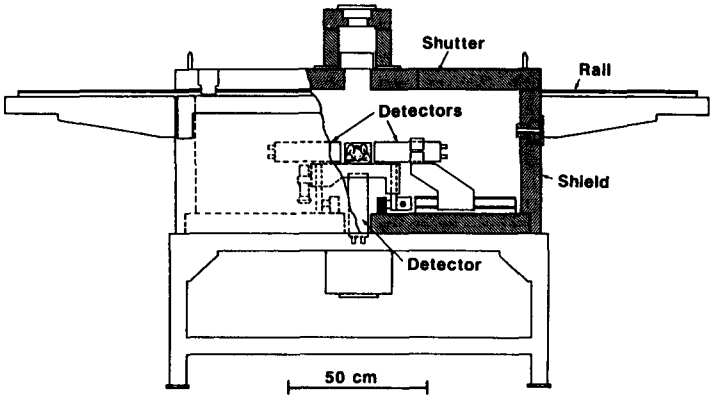

Figure 1. Schematic diagram of the whole-body counter.

The counting system was calibrated by measuring counting efficiencies for rats of various body weights that had been sacrificed for different experimental purposes and then plotted the measured counting efficiencies against the body weight (see **Figure 2**).

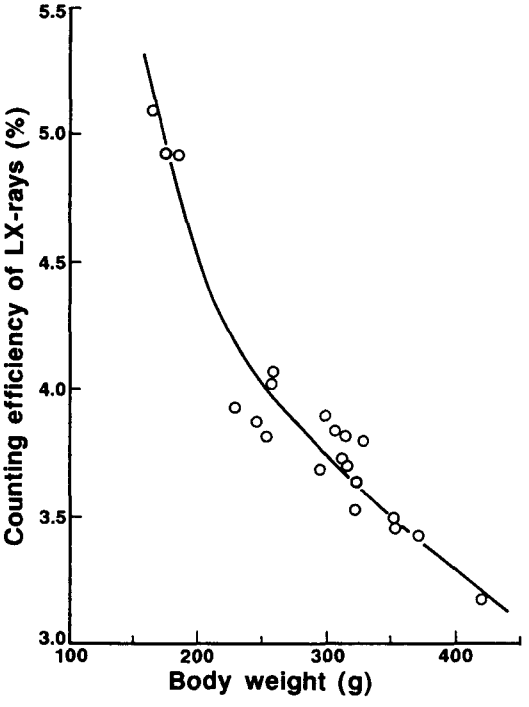

Figure 2. Counting efficiency of L X-rays for the rats of various body weights.

RESULTS AND DISCUSSION

Lung retention of Pu followed in the rats inhaled with various ILB is shown in **Figure 3**; the ILB were 0.28 kBq, 0.72 kBq, 1.37 kBq and 2.4 kBq. The master retention function obtained using five rats of their ILB between 1.99 kBq and 2.96 kBq was:

$$Y(t) = 0.766\exp(-0.0131t) + 0.234\exp(-0.000873t)$$

where $Y(t)$ is the relative Pu content in lungs at time t days after exposure(5), which is shown in the figure as curves. The present result shows that the alveolar lung clearance of $^{239}PuO_2$ was independent of ILB between 0.28 kBq and 2.4 kBq, which does not agree to the recent data of Pacific Northwest Laboratories (PNL) which claimed the alveolar lung clearance inversely proportional to ILB of 0.4-3.9 kBq (6).

The physicochemical characteristics of the aerosols used in these two laboratories were not the same. Considering the present methodology, our aerosol generation system is designed to produce PuO_2 particles that are spherical and of small sizes, compared with those of PNL. Therefore we only give an answer that there are PuO_2 particles whose ILB does not affect the alveolar lung clearance of inhaled PuO_2.

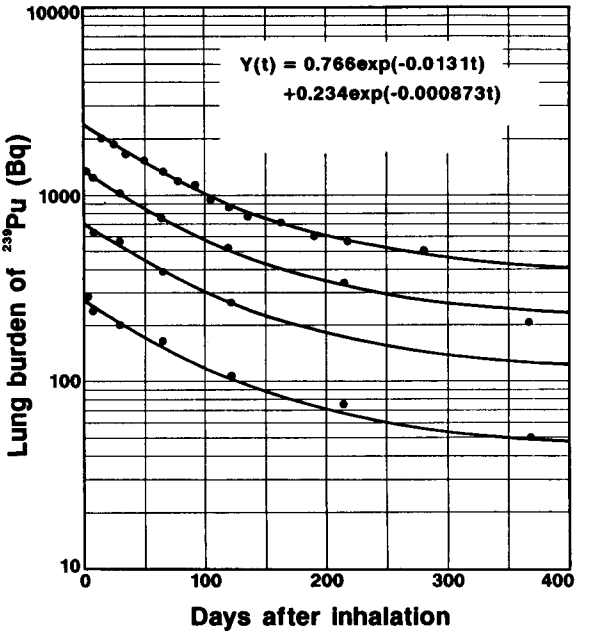

Figure 3. Lung retention of Pu of rats following inhalation of aerosols of PuO_2

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Cs^{137} TRANSFER FROM MOTHER TO EMBRYOS, IN THE FIRST THREE YEARS AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

The kinetics of the transfer of radionuclides from mother to embryo is still a matter to be solved. After the Chernobyl accident, we had the possibility to study the transfer of Cs^{137} from mother to embryo, in the case of a continuous and variable Cs^{137} intake of the mother. The study was carried on for a period of three years after the accident. Our group performed also measurements of transfer from mother to embryo, in the case of a continuous, prolonged, but rather constant intake. The results of this study will be presented in future papers.

MATERIALS AND METHOD

During April 1986 - September 1989 we have determined Cs^{137} content in 96 human embryos, aged between 5 and 11 weeks at the moment of prelevation, as well as Cs^{137} dietary intake and urinary excretion of the mother, while they were in hospital for the abortion. All the subject mothers were living in Bucharest, and were aged between 17 and 44 years.

In order to do this, embryos of different ages were prelevated from the hospitals, as well as dietary intake and urine samples from their mothers, within the entire period of study.

The Cs^{137} content of the embryos and of the dietary and biological samples was performed by radiochemical separation [1], followed by beta counting with a low-level, high-efficiency counter.

RESULTS AND DISCUSSION

Within the period of study, the average Cs^{137} content in embryos increased from 97.3 mBq/g tissue in 1986, to 137.9 mBq/g tissue in 1987, then it decreased to 10.3 mBq/g tissue in 1988. In 1989, the content in embryos was very small; in many cases, it was bellow the minimum detectable activity (MDA). Those variations are presented in Fig. 1, where the peak Cs^{137} content in embryos can be observed, in March 1987.

Since April 1986 until December 1986, 44 embryos of different ages were prelevated monthly, as well as dietary intake and urine samples from their mothers, and we measured the Cs^{137} content. Among the 44 embryos studied, 16 were 7 weeks old, 11 were 6 weeks old, and the other covered the other ages from 5 to 11 weeks.

Table 1: Cs^{137} content in embryos, with respect to the average value

Cs^{137} content in embryos (mBq/g tissue)	Age of the embryos (in weeks)					
	6	7	8	9	10	11
> 97.3	45.45%	56.25%	60%	-	66.66%	100%
< 97.3	54.55%	43.75%	40%	100%	33%	-

The average value of the Cs^{137} content in the embryos analysed in 1986 was 97.3 mBq/g tissue (for all the ages). Considering this value as reference value for that period, we have separated the Cs^{137} content in embryos in two groups: "high Cs^{137} content", when it exceeds the average value, and "low Cs^{137} content" when it is lower than the average. The percentage distribution of the Cs^{137} content in embryos, considering this reference value and the age of the embryos, is presented in Table 1.

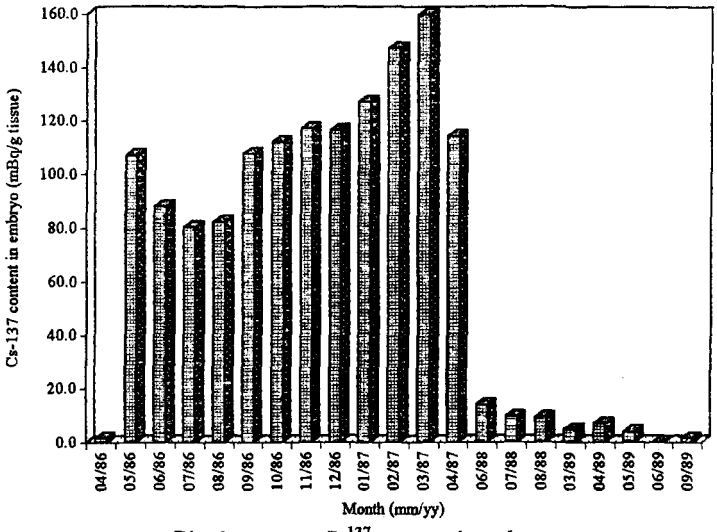

Fig. 1: Average Cs^{137} content in embryos

It appears that, the older are the embryos, the higher the number of cases with increased Cs^{137} content (with one exception: 9 weeks embryos). This indicates that Cs^{137} content tends to increase as the age of the embryo increases.

Cs^{137} intake of the mothers took values between 61.5 and 127.2 Bq/day, in 1986, with an average value of 90.16 Bq/day. The percentage distribution of embryos with Cs^{137} content higher than 97.3 mBq/g tissue, with respect to Cs^{137} intake of the mother, is as presented in Table 2. The table bellow seems to point out that there is a step value for the mother intake (103.5 Bq/day) above which the percentage of embryos with "high Cs^{137} content" is directly dependent of the quantity of Cs^{137} ingested by the mother no matter how old is the embryo.

Table 2: Cs^{137} content in embryos with respect to Cs^{137} of the mother

Cs^{137} content in embryos	Cs^{137} intake of the mother (Bq/day)							
(mBq/g tissue)	40.3	61.5	70.4	93.8	103.5	105.6	119.0	127.2
> 97.3	20%	75%	-	-	80%	66.6%	80%	85.7%
< 97.3	80%	25%	100%	100%	20%	33.4%	20%	14.3%

Cs^{137} content in embryos prelevated during January 1987 - April 1987 was the highest among all the considered periods.

The 16 embryos studied in 1987 had ages between 6 and 10 weeks, the number of embryos of a certain age being variable (from one embryo aged 10 weeks, to 6 embryos aged 7 weeks) imposed an overall discussion of the values for all the embryos. In 1987, the percentage distribution of embryos with Cs^{137} content higher than the annual average was the following:

Table 3: Cs^{137} content in embryos, in 1987

Cs^{137} content in embryos (mBq/g tissue)	Age of the embryo (weeks)				
	6	7	8	9	10
> 137.9	25%	83.3%	50%	100%	100%
< 137.9	75%	16.7%	50%	-	-

It can be seen that, while only 25% of the 6 weeks embryos have Cs^{137} content higher than the average, all the 9 weeks and 10 weeks embryos have higher Cs^{137} content than the average. Cs^{137} intake of the mothers during January 1987 - April 1987 ranged between 92.6 and 134 Bq/day. From the results obtained it seems that the content in embryos increases as the intake of the mother increases.

Cs^{137} content of the 17 embryos studied in 1988 was lower than in the precedent years, and ranged between 6.7 and 15.8 mBq/g tissue, with an annual average of 10.3 mBq/g tissue. Performing the same analysis as before, we have obtained the following data for the percentage distributions of "high Cs^{137} content" with respect to the age of embryos:

Table 4: Cs^{137} content in embryos, in 1988

Cs^{137} content in embryos (mBq/g tissue)	Age of the embryo (weeks)					
	5	6	7	8	9	10
> 10.3	50%	50%	66.66%	66.66%	33.3%	100%
< 10.3	50%	50%	33.3%	33.3%	66.6%	-

In this case too, the Cs^{137} content seems to be higher in older embryos than in younger embryos.

Cs^{137} intake of the mother during January 1988 - August 1988 took values between 6.5 Bq/day and 11.5 Bq/day. In this case, the data were not sufficient to allow us to make any correlation between the mother intake and the content in embryo.

In 1989 Cs^{137} content in embryos was extremely low, 32% of the embryos having Cs^{137} content bellow MDA, and the average Cs^{137} content in embryos was 3.35 mBq/g tissue (for all the embryos analysed), and 6.31 mBq/g tissue (for the embryos with Cs^{137} content above MDA). Because of the low Cs^{137} content in embryos, any attempt of correlating the Cs^{137} content with the age of the embryos or with the intake of the mother is very difficult.

CONCLUSIONS

The data discussed above seem to poin to two conclusions:

- Cs^{137} content in embryos increases as the age of the embryo increases
- there is a step value for the Cs^{137} intake of the mother, above which the content of the embryo increases with the intake of the mother

The present data did not allowed us to corelate the Cs^{137} content of the embryo with the age of the mother.

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EFFECTIVE DOSES COMMITTED IN BUCHAREST AREA DUE TO Cs^{137} AND Sr^{90} INTAKE, 9 YEARS AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

On the 26th of April, 1986, a major accident occurred at the fourth reactor of the Chernobyl nuclear power plant, in Ukraine. The meteorological conditions lead to important fallout over our country. The most important radionuclides carried by the radioactive plume over Romania were I^{131}, Cs^{134}, Cs^{137} and Sr^{90}. After the decay of I^{131}, and of the other short-lived radionuclides, Cs^{137} and Sr^{90} remained the most important contaminants on the Romanian territory. The principal route of intake for these two radionuclides is considered to be the ingestion of contaminated foods. Therefore, we have measured Cs^{137} and Sr^{90} content in dietary intake for a group of adult subjects. The data for dietary intake were used to determine the effective doses committed annually, with the dose factors recommended by ICRP 67 [1].

MATERIALS AND METHOD

All the samples were prelevated since April 1986 until April 1995, during nine years after the Chernobyl accident, in the Bucharest area.

The population group selected for this study was a group of adult men, aged between 25 and 40 years, without any health problems, hard labourers. The food samples were prelevated from the cafeteria of the factory, where their were all taking their meals. Usually, the food was prelevated in the second part of every month.

For all samples, Cs^{137} and Sr^{90} content was determined by radiochemically separated, using standardised methods (HASL-300), followed by beta counting with a low-level high-efficiency counter.

RESULTS AND DISCUSSION

During the nine years of our study, the quantity of ingested Cs^{137} varied a lot from one year to another. As expected, the Cs^{137} intake had the highest value in the first year after the accident. The peak of the Cs^{137} intake was reached in May 1986, when the average daily ingested quantity was 408.5 Bq. Beyond this date, on the background of a continuous decrease of Cs^{137} intake, other two peaks occurred, for reasons presented in previous papers [2]. The experimental data are very well fitted by the following "dietary intake function":

$$I(t) := 1.84 \cdot 10^{-2} \cdot t^{4.31} \cdot \exp(-0.16t) + 184 \cdot \exp\left[-\left(\frac{t - 220.52}{186.56}\right)^2\right] + 23.27 \cdot \exp\left[-\left(\frac{t - 683.4}{193.5}\right)^2\right]$$

with t - the time elapsed after the accident, in days.

As it can be seen in figure 2, there is a strong variation of Sr^{90} intake during April 1986 - March 1995. To begin with May 1986, Sr^{90} dietary intake increased continuously, until it reached 1.485

Bq/day in October 1986 (Fig. 2). Strontium intake remains at high values until April 1987, then it decreases continuously.

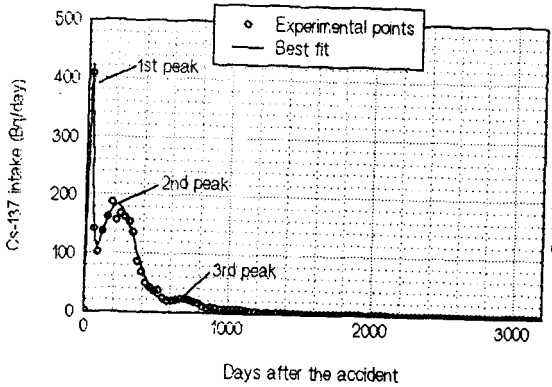

Fig.1: Cs^{137} dietary intake, during nine years after the accident

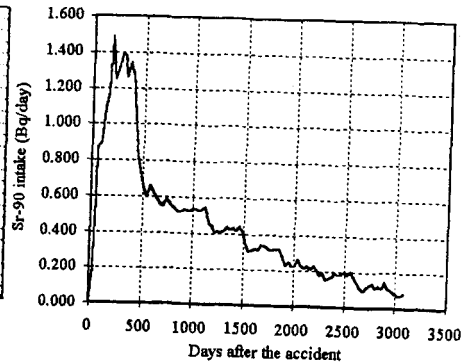

Fig. 2: Sr^{90} dietary intake, during nine years after the accident

The effective doses committed annually due to internally deposited Cs^{137} are presented in Fig. 3, while the effective doses committed due to Sr^{90} dietary intake are presented in Fig. 4 (to red marrow, bone surface, and their sum).

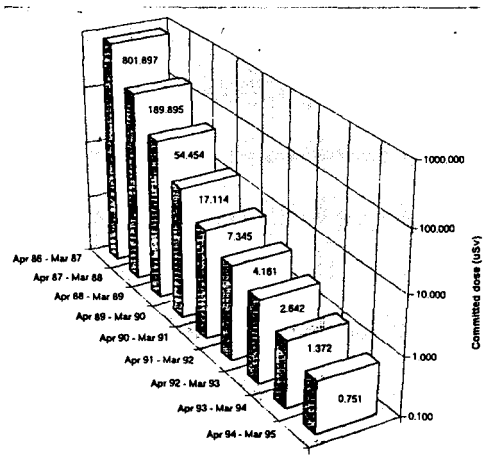

Fig. 3: Effective doses committed due to Cs^{137} intake

As it can be seen, in the first year after the accident, Cs^{137} intake lead to an effective dose commitment of 801.9 μSv. This value decreased quite fast, as the amount of caesium intake decreased, and reached 7.3 μSv five years after the accident, then 0.75 μSv in the ninth year after the accident. A comparison of the doses committed due to Cs^{137} and Sr^{90} intake is presented in Fig. 5.

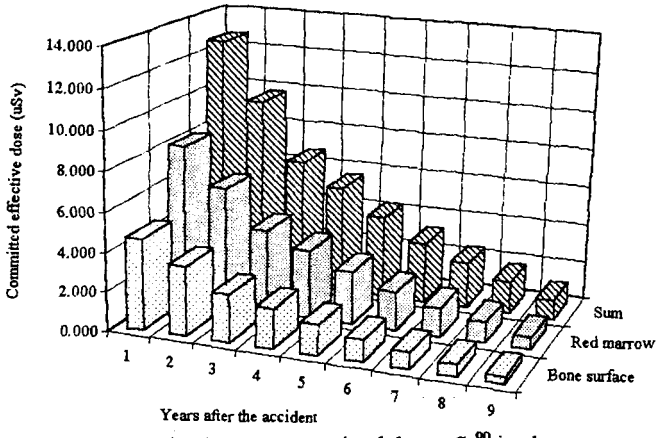

Fig. 4: Doses committed due to Sr^{90} intake

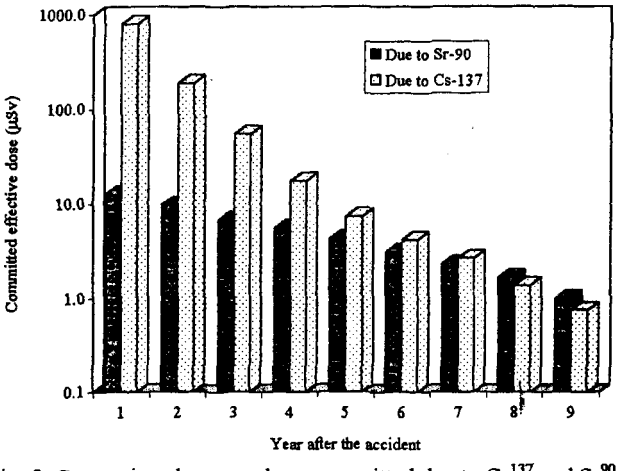

Fig. 5: Comparison between doses committed due to Cs^{137} and Sr^{90} intake

CONCLUSIONS

The effective dose committed by the adults in then first year after the accident, due to Cs^{137} dietary intake was 801.9 μSv while the dose committed due to Sr^{90} intake was 12.69 μSv (bone surface and red marrow). The doses committed annually are continuously decreasing, so that in the eight year after the accident the dose due to Sr^{90} is higher than the dose due to Cs^{137}. The effective dose committed by the adults in the nine years elapsed from the accident were 1079.6 μSv due to Cs^{137}, while due to Sr^{90} the doses were 16.978 μSv (to the bone surface) and 29.814 μSv (to the red marrow).

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IRPA9
1996 International Congress on Radiation Protection
April 14- 19, 1996
Vienna, Austria

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PAPER TITLE _ Cs^{137} AND Sr^{90} DIETARY INTAKE AND URINARY EXCRETION FOR CHILDREN, AFTER THE CHERNOBYL ACCIDENT

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ABSTRACT

A major problem after the Chernobyl accident was to asses the effects of the irradiation for different age groups, especially children. Our group has measured Cs^{137} and Sr^{90} dietary intake and urinary excretion for children having different ages (between 4 and 12 years), during different time intervals. From the intake data, a trend of the annually committed effective doses was deduced. The paper presents the dose values for different age groups, and their evolution in time. A comparison with the doses committed by adults living in the same conditions as the children is presented.

ASSESSMENTS OF INTERNAL DOSES BY INGESTION OF RADIOACTIVE FOODSTUFFS IN BANGLADESH

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ABSTRACT

The internal radiation dose to a man from the consumption of foodstuffs was estimated on the basis of the measured radioactivities in the foodstuffs in Bangladesh. The total annual internal effective dose equivalent was found to be 454.56 μSv. The dose from intake of radionuclides by foodstuffs (ingestion dose) in general is so low that no harmful effects will occur directly.

INTRODUCTION

Exposure of man to ionising radiation can cause harmful effects. The extent of these effects is dependent on the dose. The possible exposure pathways to individuals include direct radiation from an airborne radioactive plume, inhalation of radioactive gases and aerosols, external radiation from the contaminated ground, and internal dose from ingestion of contaminated food. Food intakes is one of the important pathways for long term health considerations. There may be a variety of radionuclides of artificial as well as natural origins which normally get entry into human body through foodchain and other pathways and deposited in the critical organ(s) causing internal irradiation (1,2). Artificial radioactivity comes mainly from nuclear weapon testings, major nuclear power accidents and radioactive wastes. Among the naturally occurring radionuclides, ^{40}K and the products of the ^{232}Th and ^{238}U decay series are the most important ones. To assess the radionuclide intake by man thourgh food consumption, it is necessary to measure the radioactivity in different foodstuffs. This study was undertaken to determine the radioactivity levels in foodstuffs in order to estimate the effective dose equivalent from dietary intakes of radionuclide-contaminated foodstuffs in Bangladesh. In this study, ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs radionuclides have been considered.

RADIOACTIVITY IN FOODSTUFFS

Food samples representing the major dietary habits of local people were collected from the local markets. This included cereal, vegetable, milk, fresh fish, fresh meat, fresh fruit, etc. All samples were cleaned, processed and dried at room temperature (3,4). About 1 kg dry sample was used in a Merinelli beaker, which is then sealed for more than 30 days to allow for a radium-radon equilibrium to be reached before further measurements (5). The gamma spectrum of each samples was determined with a HPGe detector connected to a 4096 channel analyzer. The effective volume of the detector was 72.4 cm^3 and its energy resolution was 2.1 keV from 1.33 MeV gamma line of ^{60}Co. Samples and standard were counted in identical containers. Details of the counting and efficiency calibration have been given in earlier reports (3-5). The radioactivity was determined from the counting rate under respective photopeak areas and the efficiency of the detector for respective energies. The photopeaks at 609 keV (^{214}Bi due to ^{238}U), 583 keV (^{208}Tl due to ^{232}Th), 1460 keV (^{40}K), and 662 keV (^{137}Cs) were used for the determination of radioactivity in foodstuffs.

RADIONUCLIDE INTAKE AND DOSE ESTIMATES

The intake of radionuclides with food is dependent on the concentration of radionuclides in the various foodstuffs and on the food consumption. It is obvious that food consumption depends on many factors, some of which concern the individual while others are group related. Information on the range and amounts of foods consumed regularly by individuals is required. Types of food consumed are related, of course, to the specific geographical, as well as the cultural, economic, social and even political, conditions within and amongst countries (6).

The risk associated with an intake of radionuclides in the body is proportional to the total dose delivered by the radionuclides while staying in the various organs. In general it is assumed that stochastic effects occur linearly with dose and usually the effective dose equivalent H_e is used to define this risk. So H_e is a parameter for the biological effect. Intake to effective dose equivalent conversion factors are needed in order to convert the intake into dose on ingestion of radionuclides into the body. The intake to dose conversion factors (50-year period) cited in the ICRP publication no.51 for the members of the public (adults) were used. The factors used for estimation of doses are (1,7,8): 6.2×10^{-9} Sv/Bq for ^{238}U; 7.4×10^{-7} Sv/Bq for ^{232}Th; 5.0×10^{-9} Sv/Bq for ^{40}K; and 1.2×10^{-8} Sv/Bq for ^{137}Cs. Radioactivity levels in foodstuffs were used to estimate internal effective doses. Internal doses were estimated from the foodstuffs by multiplying the average concentrations of ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs by the yearly food intake and ingestion dose coefficients.

RESULTS AND DISCUSSIONS

The content of ^{238}U, ^{232}Th, ^{40}K, and ^{137}Cs radioactivity in foodstuffs varies from 0.16 to 1.28 Bq/kg, 0.22 to 1.12 Bq/kg, 95.22 to 220.54 Bq/kg, and 1.19 to 3.89 Bq/kg, respectively. The average concentration of ^{238}U, ^{232}Th, ^{40}K, and ^{137}Cs in foodstuffs, grouped according to their food type, are listed in Table 1.

Table 1. Average radioactivity in foodstuffs in Bangladesh.

Name of foodstuffs	Activity (Bq/kg)			
	^{238}U	^{232}Th	^{40}K	^{137}Cs
Cereal	1.28	1.12	150.27	2.25
Roots & tubers	0.32	0.22	120.51	3.89
Vegetables	0.16	0.36	180.07	3.49
Fish	0.26	0.45	109.11	1.82
Meat	0.37	0.29	95.22	4.65
Fruit	1.11	0.95	188.56	1.57
Milk	1.19	0.88	220.54	1.19
Total	4.69	4.27	1064.28	18.86

Based on the food consumption rates (6) and radionuclide concentration given in Table 1, the annual intakes of ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs are estimated. The total annual intakes of ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs are estimated to be 337.13 Bq, 295.23 Bq, 45240.45 Bq and 686.87

Bq, respectively. Table 2 gives the daily intake of various food items and the daily intake of various radionuclides through the above foodstuffs.The intake rates for Bangladeshi foodstuffs were taken from the food consumption statistics data given in the report (6). Internal doses were estimated from the foodstuffs by multiplying the average concentrations of ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs by the yearly food intake and ingestion dose coefficients. The estimated internal doses are also included in the last column of Table 2. A person receives about 226.24 μSv/y from ^{40}K, 2.09 μSv/y from ^{238}U, 218.01 μSv/y from ^{232}Th and 8.22 μSv/y from ^{137}Cs. The total ingestion dose from the consumption of radioactive foodstuffs due to ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs is estimated to be approximately 454.56 μSv/y.

Table 2. Average annual intakes of radionuclides in the foodstuffs and estimated annual effective dose.

Type of food	Consumption (g/day)	Annual intake of radionuclide (Bq)				Estimated annual effective dose (μSv/y)
		^{238}U	^{232}Th	^{40}K	^{137}Cs	
Cereal	631.7	295.12	258.24	34647.75	518.78	183.20
Roots & tuber	44.8	5.23	3.60	1970.33	63.60	10.67
Vegetables	26.9	1.57	3.54	1768.28	34.27	9.28
Fish	20.3	1.93	3.33	808.51	13.49	4.24
Meat	10.8	1.46	1.14	375.17	18.32	2.11
Fruit	39.6	16.04	13.73	2724.69	22.69	14.09
Milk	36.7	15.94	11.79	2955.23	15.95	15.14
Total	810.8	337.29	295.27	45249.96	687.10	454.56

It should be pointed out that these individual dose estimates are based on measured dietary concentrations of ^{238}U, ^{232}Th, ^{40}K and ^{137}Cs radionuclides, taking into account the assumed dietary habits of the general population in Bangladesh. The contributions of other radionuclides to the individual effective dose equivalent can be considered negligible since their concentrations in foodstuffs were found to be less than the lower detectable activities. The dose from intake of radionuclides by foodstuffs (ingestion dose) in general is so low compared to natural external radiation (2000 μSv/y) that no harmful effects will occur directly.

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EVALUATION OF ACTUAL LEVEL OF ^{90}Sr AND ^{137}Cs IN URINE AS INDICATOR FOR INTERNAL CONTAMINATION OF HUMAN BODY IN THE CASE OF NUCLEAR ACCIDENT

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ABSTRACT

In this work it was determined the urinary contamination with ^{90}Sr and ^{137}Cs as indicator of the exposure of population from Bechet zone to the risk of nuclear accident, created by C.N.E. Koslodui-Bulgaria. Two lots of persons were selected, the first from Bechet zone, situated within 30 km area from Koslodui, and the second from a central zone, Melinesti, situated far at 200-300km from Koslodui. Every lot was formed by 35 healthy persons, males and females, and was structured in this way: of 7 to 9 years old, of 13 to 14 years old, of 17 to 18 years old and after 25 years old. The results obtained and the conclusions that can be drawn about the distribution of these isotopes in urinary daily excretion function by age, sex or distance from C.N.E. Koslodui are presented below.

INTRODUCTION

The twenty years fonction of Nuclear Power Station Koslodui - Bulgaria, situated within 30 km from Bechet - Romania zone, determines a permanent risk of nuclear accident, for the population who lives there (1).

That is why in april 1994, our laboratory was recolted total daily urines samples, from two selected lots of population, one from Bechet zone and the second from the Melinesti control zone, situated far from Koslodui and were determined by radiochemical analysis (2),(3), the daily urinary level of ^{90}Sr and ^{137}Cs, as indicator for internal contamination of human body (4),(5).

RESULTS AND DISCUSSION

It's known, the human internal contamination is a result of environmental contamination, that is why we must specifiy the Chernobyl major nuclear accident (april 1986) who afected our country, Melinesti zone higher and Bechet zone lower.

In this circumstances, the human body contamination and also the urinary contamination with ^{90}Sr and ^{137}Cs must have a same direction in case Nuclear Power Station Koslodui - Bulgaria, was not poluante and Bechet zone was not affected.

Indeed, the results of the determinations presented in table 1 + 4, proves the same report exested after Chernobyl, the urinary contamination with Sr-90 and Cs-137 in the lot of population from Melinesti zone, was higher comparatively with the lot of population from Bechet zone.

So, the ponderate mean value for Sr-90 was in Melinesti population 0,123 Bq/day for female and 0,084 Bq/day for male, comparatively with Bechet population 0,071 Bq/day for female and 0,075 Bq/day for male.

For Cs-137, the ponderate mean value was, in Melinesti population 0,505 Bq/day for female and 0,364 Bq/day for male, comparartively with Bechet population; 0,291 Bq/day for female and 0,207 Bq/day for male.

Also, at all the group of age, the urinary levels of Sr-90 and Cs-137 was higher at Melinesti zone comparatively with Bechet zone.

Table 1: Urinary daily excretion level of ^{90}Sr from the lot of Bechet.

Group of age (years)	^{90}Sr - urinary excretion (Bq/day)					
	Female			Male		
	Subjects number	Mean	Variation interval	Subjects number	Mean	Variation interval
7 - 9	2	**0,070**	0,043-0,090	4	**0,044**	0,036-0,046
13 - 14	6	**0,050**	0,010-0,120	2	**0,070**	0,019-0,123
17 - 18	5	**0,070**	0,031-0,120	5	**0,110**	0,047-0170
total teenagers 7 - 18	13	**0,061***	0,010-0.120	11	**0,078***	0,019-0,170
adults 31 - 66	4	**0,102**	0,048 -0,200	5	**0,087**	0,045-0,130
Total lot 7 - 66	17	**0,071***	0,010-0,200	16	**0,075***	0,042-0,170

Table 2: Urinary daily excretion level of ^{90}Sr from the lot of Melinesti.

Group of age (years)	^{90}Sr - urinary excretion (Bq/day)					
	Female			Male		
	Subjects number	Mean	Variation interval	Subjects number	Mean	Variation interval
7 - 9	1	**0,190**	-	3	**0,095**	0,035-0,160
13 - 14	3	**0,078**	0,060-0,093	4	**0,073**	0,029-0,097
17 - 18	4	**0,169**	0,116-0,290	4	**0,081**	0,049-0,136
total teenagers 7 - 18	8	**0,138***	0,060-0,290	11	**0,082***	0,029-0,160
adults 31 - 56	2	**0,065**	0,051-0,078	2	**0,095**	0,090-0,095
total lot 7 - 56	10	**0,123***	0,051-0,290	13	**0,084***	0,029-0,160

Table 3: Urinary daily excretion level of ^{137}Cs from the lot of Bechet.

Group of age (years)	^{137}Cs - urinary excretion (Bq/day)					
	Female			Male		
	Subjects number	Mean	Variation interval	Subjects number	Mean	Variation interval
7 - 9	2	**0,330**	0,014-0,510	4	**0,196**	0,049-0,130
13 - 14	6	**0,310**	0,104-0,530	2	**0,255**	0,170-0,340
17 - 18	5	**0,365**	0,075-0,690	5	**0,224**	0,044-0,360
total teenagers 7 - 18	13	**0,334***	0,075-0,690	11	**0,219***	0,044-0,360
adults 31 - 56	5	**0,224**	0,090-0,540	5	**0,180**	0,090-0,261
total lot 7 - 56	18	**0,291***	0,075-0,690	16	**0,207***	0,044-0,360

*Note: The value is ponderate mean.

Table 4: Urinary daily excretion level of ^{137}Cs from the lot of Melinesti.

Group of age (years)	^{137}Cs - urinary excretion (Bq/day)					
	Female			Male		
	Subjects number	Mean	Variation interval	Subjects number	Mean	Variation interval
7 - 9	2	**0,830**	0,650-1,01	4	**0,377**	0,097-1,120
13 - 14	4	**0,265**	0,140-0,450	4	**0,293**	0,140-0,400
17 - 18	4	**0,684**	0,175-1,800	4	**0,448**	0,140-0,960
total teenagers 7 - 18	10	**0,546***	0,140-1,800	12	**0,352***	0,097-1,120
adults 31 - 56	3	**0,370**	0,150-0,630	3	**0,410**	0,240-0,520
total lot 7 - 56	13	**0,505***	0,140-1,800	15	**0,364***	0,097-1,120

* Note: The value is ponderate mean.

CONCLUSIONS

The conclusion of the work is, the fonction of C.N.E. Koslodui - Bulgaria, was not modified significant the human body contamination with ^{90}Sr and ^{137}Cs, the urinary level of these isotopes, proved it.

But, also we consider it is necessary to establish a permanent supervise program, determine the urinary contamination with Sr-90 and Cs-137 in Bechet population, special in young people and in children, as indicator for internal contamination of human body, parallel to food and drinking water program.

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POOLING TECHNIQUES FOR BIOASSAY SCREENING[1]

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ABSTRACT

Pooling techniques commonly are used to increase the throughput of samples used for screening purposes. While advantages of such techniques are increased analytical efficiency and cost savings, the sensitivity of measurements decreases because it is inversely proportional to the number of samples in the pools. Consequently, uncertainties in estimates of dose and risk which are based on the results of pooled samples increase as the number of samples in the pools increases in all applications. However, sensitivities may not be seriously degraded, for example, in urinanalysis, if the samples in the pools are of known time duration, or if the fraction of some attribute of the grab urine samples to that in a 24-hour composite is known (e.g., mass, specific gravity, creatinine, or volume, per 24-h interval). This paper presents square and cube pooling schemes that greatly increase throughput and can considerably reduce analytical costs (on a sample basis). The benefit-cost ratios for 5x5 square and 5x5x5 cube pooling schemes are 2.5 and 8.3, respectively. Three-dimensional and higher arrayed pooling schemes would result in even greater economies; however, significant improvements in analytical sensitivity are required to achieve these advantages. These are various other considerations for designing a pooling scheme, where the number of dimensions and of samples in the optimum array are influenced by: 1) the minimal detectable amount (MDA) of the analytical processes, 2) the screening dose-rate requirements, 3) the maximum masses or volumes of the composite samples that can be analyzed, 4) the information already available from results of composite analysis, and 5) the ability of an analytical system to guard against both false negative and false positive results. Many of these are beyond the scope of this paper but are being evaluated.

INTRODUCTION

Often a large number of samples must be analyzed to screen them for any possible high values. If the costs of analysis are high, it may be cost-effective to combine (pool) several samples and analyze the pool, or a fraction of it, to decide if specific analyses of the individual samples in the pool are justified. In 1987, a fission track analysis (FTA) method was developed at Brookhaven National Laboratory (BNL) (1) and used for plutonium urinalysis for the Marshallese (2,3). Although sample analysis using the FTA method is expensive, because of the FTA's ultra-low level of detection sensitivity, pooling methods given in this paper can be used to increase the throughput of samples, saving both time and money.

POOLING SCHEMES AND TECHNIQUES

In the same available time, square and cube pooling schemes allow a significantly larger throughput of samples than making individual analyses, and hence can considerably reduce analytical costs (on a per sample basis). For a square matrix composite method, a total of N^2 (N is an integer) individual samples can be pooled in batches of 2N single composite samples; the composite samples are pooled from the corresponding rows and columns, respectively. Once all composite samples have been analyzed, no further testing is needed for determining individual dose levels. In a square composite pooling scheme, all rows' and columns' test samples can check and verify for each other. This thesis applies also to a multidimensional pooling scheme. Therefore, the additional advantage of using a square matrix or multidimensional pooling scheme over a simple composite pooling method is reducing the probability for false positive and false negative results.

In a square pooling matrix, the probability of two composite samples, from an intersecting row and an intersecting column, being false positive is p^2. There is a 10% probability of false positives in the existing BNL FTA system established for plutonium dose assessments; therefore, the chance of generating positive results in both a column and row composite urine is reduced to 1% (0.1^2). Similarly, if spike (e.g., calibration) samples are used in a pooling scheme, then the probability of false negative results also can be improved because positive results in each intersecting composite must be identified and be traceable for all spiked samples. Overall, the final

[1]This work was supported by the U.S. Department of Energy under contract No. DE-AC02-76CH00016.

positive results are only accepted as positive, for example, using a square matrix method, when both the row and column composite results are positive simultaneously, and both results also are within a factor of two of each other. The criterion of 2 can be changed according to the precision required of the measurements and the mass balance of plutonium activity in the samples. Therefore, all "non-coincident" row and column or array composite results are to be treated as false positive outcomes and should be eliminated for dose calculations. These include the following three cases: 1) only one out of the ten test samples is positive, 2) only multiple row (or columns) composite test samples are positive, 3) all imbalanced (greater than outside preset limits) plutonium activity in all positive coincident rows and column results.

The square matrix composite pooling approach can be generalized further to three and higher dimensions. Significant improvements in analytical sensitivity would be required for four-dimensional and higher arrays. Table 1 illustrates several options including square, cube, and four (quad) dimension arrays. For example, a 3x3x3 array would allow 27 individual urine samples to be linked using 9 composites, with portions of 9 samples in each composite. Each sample is uniquely identified at the intersection of three composites. Further, the benefit-cost ratios in a 5x5 square matrix and a 5x5x5 cube schemes are 25/10=2.5 and 125/15=8.3, respectively. Three-dimensional arrayed pooling schemes and higher ones would result in even greater economies, as the last column of Table 1 shows. However, significant improvements in analytical sensitivity are required to achieve these advantages. The probability of accepting false positives is reduced as p^n, where n is the dimension of the pooling array. Although we do not expect improvements in the probabilities of avoiding false negatives for screening the maximum individual outcomes, this is unimportant for the FTA system.

Table 1. Multidimensional Pooling Options

Array		No. of Samples	Samples per Composite	Volume of Each Urine Sample	Composites Analyzed	Benefit-Cost Ratio
Dimensions	**Size**					
Square	5x5	25	5	1/5	10	25/10=2.5
Square	10x10	100	10	1/10	20	100/20=5.0
Cube	3x3x3	27	9	1/9	9	27/9=3.0
Cube	4x4x4	64	16	1/16	12	64/12=5.3
Cube	5x5x5	125	25	1/25	15	125/15=8.3
Quad	4x4x4x4	256	64	1/64	16	256/16=16

The number of dimensions and samples in the optimum array is influenced by: 1) the MDA of analytical processes, 2) the maximum masses or volumes of the composite sample that can be managed in the analytical process, 3) the information available from results of previous composite analyses, and 4) the criteria that the analytical system must meet to guard against both false negative and false positive results. Many of these, and other, considerations are beyond the scope of this paper, and are being evaluated for application in future pooling methodologies.

CONCLUSIONS

The main advantage for pooling urine samples (grab or 24-h) for bioassays is increased analytical efficiency and cost savings. The sample's screening power is inversely proportional to the duration of collection of a grab sample, and may be derived from a recommended guideline or a specific protection standard. Using a larger fraction of the collected individual sample lowers the screening goal. In cases where it is necessary to ensure that the dose rate of the maximally exposed individual in the pool is below some fixed value, pooling samples can be a viable option. It must be recognized that interpretations of the results of short-term samples are based on assumptions which require further verification (e.g., a constant rate of diurnal excretion; the ability of approximating the duration of grab samples using measurements of specific gravity or creatinine). There also may be no opportunity for analytical verification because small grab-sample volumes can be quickly depleted in the preliminary stages of pooling.

In general, for designing a pooling scheme, the optimum screening plan will depend on the following parameters: 1) the total number of samples to be analyzed, 2) the sensitivity (MDA) of the analytical method to be used, 3) the cost per sample to be analyzed, and 4) desired screening goals or acceptable sensitivity limit. We are making further studies to optimize this multidimensional approach.

ACKNOWLEDGMENT

The authors wish to thank Maria Beckman for her help in preparing this manuscript and Avril Woodhead for her editorial assistance.

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ASSESSMENT OF THORIUM EXCRETION IN URINE BY MEANS OF ICP-MS

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INTRODUCTION

Thorium is a primordial element with only radioactive isotopes. It is quite ubiquitous in the earth's crust and small amounts are present in foodstuffs and also in the human body. Whereas ingestion is the main pathway of uptake in non-exposed subjects, incorporation due to occupational exposure occurs most probably by inhalation. A number of industrial applications of thorium are known e.g. in heat resistant materials, welding rods, gas mantles, Mg/Th alloys, thermistors, catalysts etc. Therefore, adequate monitoring of exposed workers is required. Due to its physical characteristics as alpha-emitter and its biokinetic behaviour following incorporation, ^{232}Th is considered as one of the radionuclides with the highest radiotoxicity. Among the methods available for monitoring of the internal thorium content, whole body counting or urine analysis by alpha spectrometry proved to be inadequate to comply with the required detection limits. Interpretation of results of faecal analysis is aggravated by the variation of thorium excretion from natural intake. Up to now, basic data on dietary thorium content and in particular its biokinetics in humans is rather limited. E.g. for the intestinal uptake, the ICRP has adopted an f_1-value of $2 \cdot 10^{-4}$ in its publications N°30 (1) and N°54 (2) and of $5 \cdot 10^{-4}$ in its publications N°56 (3) and N°69 (4). But combining data of Beyer (5) and Riedel (6), for Germany an f_1-value of at least $4 \cdot 10^{-3}$ has to be assumed even under neglect of any endogenous faecal excretion or body accumulation of thorium in adults.

ICP-MS provides a tool for rapid measurement of very low concentrations of thorium in biological fluids. Therefore, in this study the feasibility and sensitivity of ICP-MS for the assessment of renal thorium excretion was investigated. Variation of urinary thorium content was studied in a group of non-exposed persons as well as its day to day variation in a particular subject.

SUBJECTS

A total of 18 healthy volunteers (7 males: mean age 46 ± 11 years (MV ± SD), range 30 to 57 years; 11 females: mean age 42 ± 23 years, range 17 to 84 years) were included in the study. None of them had a history of previous occupational exposure to thorium. The subjects were asked to collect urine for 24 hours under normal habits in polyethylene bottles. After addition of 50 ml/l conc. HNO_3 the urine was stored at 4°C until measurement. Additionally, in one of the subjects complete urine was collected for 6 consecutive days.

PROCEDURE

For the measurements an ICP mass spectrometer ELAN 5000 (Perkin Elmer Sciex) coupled to an AS90 sample changer (Perkin Elmer) was employed. Unprocessed urine was

pumped at a rate of about 0.9 ml/min to a Ryton Chamber. In all measurements a conventional GemTip cross flow nebulizer was used. The plasma flow was set to 15 l/min, the nebulizer flow to 0.95 l/min and the auxiliary flow to 0.8 l/min. Rhenium (^{103}Rh) was taken as internal standard. Standard solutions for thorium and rhenium were obtained from SPEX Ind./USA. Only ultrapure water (Milli-Q, Millipore) and nitric acid distilled by subboiling were required for the analysis. For the determination of thorium concentrations in the urine samples the method of standard additions was employed. Under these conditions a detection limit (3 SD of background signal) of 4 μBq/l urine was achieved for ^{232}Th.

RESULTS

The mean urine volumes obtained for males and females of 1.6 ± 0.5 l and 1.3 ± 0.7 l respectively were in the expected range. For all subjects investigated the mean daily ^{232}Th excretion is 25 ± 14 μBq/day, for males it is 33 ± 14 μBq/day (range 17 to 54 μBq/day) and for females it is 21 ± 9 μBq/day (range 9 to 42 μBq/day). The difference between males and females is statistically not significant. Moreover, the results obtained so far show no clear dependence of ^{232}Th excretion on age (Fig.1). The day to day variation of renal thorium excretion in one subject during 6 consecutive days is shown in Fig.2. The values obtained range from 21 to 52 μBq/day with a mean of 35 μB/day.

DISCUSSION

The interpretation of data obtained for excretion analysis of workers occupationally exposed to primordial radionuclides requires basic knowledge on the natural content and its variation of these radionuclides in faeces and urine. Due to the low f_1-value almost all thorium ingested with food is excreted in faeces i.e. faecal monitoring of occupational thorium exposure is limited by the fluctuations of Th intake from natural sources. Because of the biokinetic behaviour of Th the urinary excretion seems to be more closely related to the Th body content and may therefore be a better measure of occupational Th incorporation. But up to now, only few data are available on urinary Th content of non-exposed persons. Applying 48 h urine collections and alpha spectrometry with 10,000 minutes measuring time, Riedel (6) could obtain a mean excretion of 30 μBq Th per day in normal urine which is in good agreement with the data found in this study. Concentration of ^{232}Th was measured by Dang (7) in the urine of 11 non-exposed subjects by means of neutron activation analysis. From his data a mean daily excretion of 16 ± 6 μBq Th can be derived if a mean volume of 1.4 l urine per day is assumed. The same technique was applied by Hewson (8) on five subjects without history of occupational thorium exposure. While for two of the subjects the urinary thorium concentration was below the detection limit of 4 μBq/l, the mean daily urinary ^{232}Th content of the other three subjects can be estimated at 29 μBq.

Both methods i.e. alpha spectrometry and neutron activation analysis require careful sample processing. They are time-consuming techniques. In contrast, measurements by means of ICP-MS are carried out within few minutes without sample work-up. Moreover, the method shows a good sensitivity and reproducibility for the assessment of thorium concentration in urine. Therefore, the application of ICP-MS offers an attractive alternative for monitoring of thorium body burdens in occupationally exposed subjects and also in larger groups of the general population.

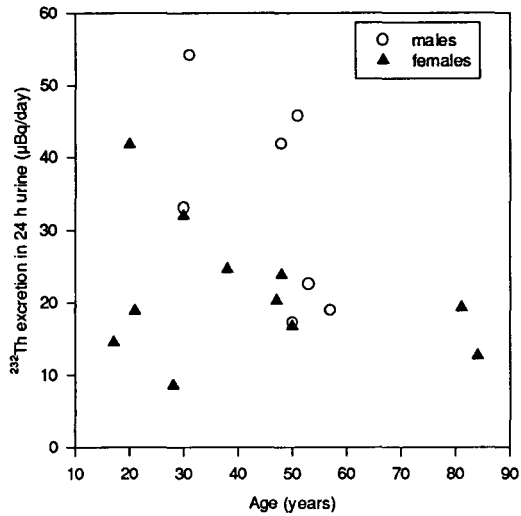

Figure 1. ^{232}Th excretion in 24 h urine in unexposed subjects.

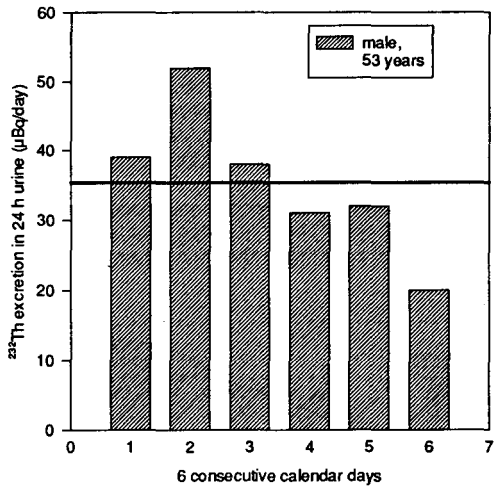

Figure 2. Intra-individual variation of urinary ^{232}Th excretion in a healthy male subject

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INVESTIGATIONS OF SECONDARY ION MASS SPECTROMETRY IN RADIATION PROTECTION DOSIMETRY

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ABSTRACT — Secondary Ion Mass Spectrometry (SIMS) allows a rapid detection of stable or radioactive elements. Based on the erosion of samples by ion bombardment, this technique makes possible the localization and the quantitative assessment of trace elements in biological samples. We have been investigated the feasibility of SIMS to measure long-lived alpha emitters at low concentrations of relevance to the individual monitoring of workers. Special attention has been paid on the preparation of the sample on a thin specimen which was found a critical parameter for this method of analysis. Our preliminary results show that ^{238}U specimen activities below 10^{-4} mBq can rapidly be measured compared with alpha spectrometry.

INTRODUCTION.

For the individual monitoring of workers exposed to the risk of contamination by actinides, conventional methods of analysis are based on the use of nuclear counting techniques, mainly alpha spectrometry. Because of the small range of the alpha particles and their low probability of emission, the use of these methods for the analysis of trace actinides implies a heavy chemical treatment to purify the sample and prepare it on a thin source. Whereas this approach was proved very efficient, a long counting time is generally necessary to reach the low detection limits required. To lower the time of measurement represents a challenge for research to improve routine analysis and especially to provide adequate methods for investigations of accidental situations.

This work was carried out to investigate the feasibility of Secondary Ion Mass Spectrometry (SIMS) as an alternative method for long-lived actinides analysis. This method was developed by R. Castaing and G. Slodzian (1). The application of SIMS is well established in the fields of solid state physics and metallurgy; its application in biomedical research was initially proposed by P. Galle (2). The detection principle is based on the erosion of a sample by ion bombardment which sputters atoms and groups of atoms from its surface. Some of the particles are ejected as charged ones. These secondary ions are collected by an electrical sector and eventually analyzed by a magnetic sector according to their mass charge ratio (Fig.1).

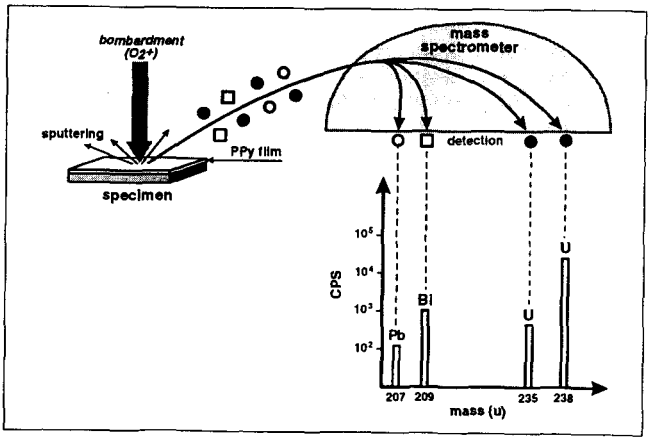

Figure 1 - Schematic representation of SIMS physical principles.

As a major advantage, SIMS is usually coupled to ion microscopy that makes possible to study the localization of a given nuclide in tissue sections. However, as a pure mass spectrometer, it provides also an accurate analytical method. Its mass resolving power, in particular, makes possible accurate determination of the isotopic ratio. In the present work therefore, as a first step to investigate the feasibility of SIMS for bioassay analysis, the isotopic ratio of sources prepared with known amounts of uranium was accurately determined. The performance of SIMS is discussed with respect to alpha spectrometry by comparing the results obtained using both methods for the same specimens.

MATERIALS AND METHODS

Specimen Preparation. Due to the erosion process by ion bombardment, the performance of SIMS critically depends on the properties of the specimen such as chemical composition and thickness. Common problems of mass interference and surface charging effects are respectively related to these two characteristics. Hence, the preparation of the specimens is the first important step for the analysis by SIMS. The use of conventional techniques based on electrodeposition was tested in preliminary experiments and showed inappropriate. In these methods indeed the specimen thickness was too thin for the erosion process to be efficient. Alternative methods needed therefore to be found.

Recent works of M. Carrier et al. (3) and O. Stéphan et al. (4) have demonstrated the viability of the use of functionalized polypyrrole (PPy) films for the preparation of alpha sources. Their method permits the synthesis of PPy films with different thickness to allow sufficient erosion ratio and signal stability. This polymer was selected because of its good incorporation yield for uranium by ion-exchange. For the present work different specimens were produced containing either natural uranium or depleted uranium.

The films was synthesised by electrochemical polymerisation following the same protocol as in reference 3, leading in our conditions to a film thickness of 250 nm. After the electrochemical process, the films, air-dried, were cut in rectangular pieces of 1 cm^2. The films fixed on its support were placed into a solution of 10 ml of oxalate of ammonium acid 10^{-2} M containing known uranium quantities. In this case, the incorporation of uranium is made by anion-exchange with the ammonium groups contained in the PPy.

Alpha Spectrometry. The specimens were previously analysed by alpha spectrometry using silicon surface barrier detector (Eurisys, France). The alpha spectrometry was used to evaluate the incorporation yield of uranium in the PPy films. The specimens measured for the present work contained mass of natural uranium and of depleted uranium in the order of 10^{-8} and 10^{-6} g respectively.

Secondary Ion Mass Spectrometry. A SIMS apparatus (Cameca, France) type IMS 300 was first investigated. This system had a mass resolving power of 300. It was used at the beginning to verify the quality of the specimens and the signal stability for isotopic ratio measurements. However, a system type IMS 4F of a newer generation allowed a mass resolving power better than 1,000. Moreover, the IMS 4F permitted a rapid switching between the masses due to its double-focusing mass spectrometer which allowed a better statistical treatment.

The primary ion beam used oxygen ions O_2^+ with energy of 5 keV to scan a maximum field of 250 μm in diameter. For the measurement of isotopic ratio, a good agreement was observed between measured and theoretical values after up to 5 different scan for a given specimen. A single scan per specimen was therefore assumed sufficient for the present experiments. During such a scan, the mass range of interest was covered several times, the isotopic ratio being obtained as the average over five such cycles. This process provides simultaneously a depth profile of the specimen. This capability was not available using the IMS 300.

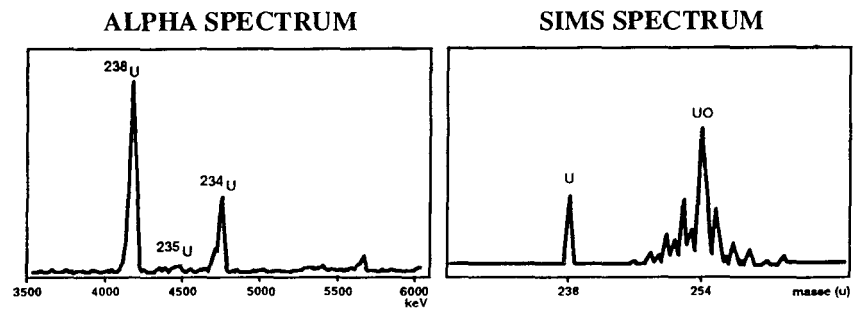

Figure 2. Alpha and mass spectra of the same specimen : depleted uranium with an activity of 17 mBq.

RESULTS

Figure 2 compares the alpha spectrum of a PPy specimen containing 17 mBq of depleted uranium with the mass spectrograph obtained with the IMS 300 for the same specimen. The counting time was 5 days and 1 min for the alpha spectrometry and the SIMS measurement respectively. The standard spectrograph given by the IMS 300 represents the average intensity of the secondary ions. It was sufficient in a first attempt to emphasise the emission of uranium either as monoatomic positive ion ^{238}U (99.6 % in mass) or as a positive ion cluster ^{254}UO. However the peaks of ^{235}U (0.4 %) et ^{234}U (0.0029 %) are not shown on the figure.

Figure 3 shows the depth profile obtained by the IMS 4F for a specimen containing natural uranium (^{238}U=99.28 %, ^{235}U=0.711 %, ^{234}U=0.0059 %) with activity of 0.75 mBq (3 x 10^{-8} g). The IMS 4F is more sensitive due to higher ion collection and transmission efficiencies which permitted the analysis of the three isotopes. Using the switching between the masses of 235 and 238, the average value of the isotopic ratio $^{238}U/^{235}U$ was measured equal to 138.9 in excellent agreement with the theoretical value (139.6). The depth profile shows that this isotopic ratio remains relatively constant with the analysis cycle although the intensity rapidly decreased. Further measurements are currently carried out to assess the overall accuracy of the isotopic ratio measurement depending on the set up of system parameters such as cycle time and primary ion beam energy. As derived from the Figure, the detection of ^{234}U, present in the specimen with the lowest concentration, indicates the low activities which can be analysed with SIMS. To give an order of magnitude, the isotopic ratio ^{234}U / ^{238}U equal to 6 x 10^{-5} lead to a detection limit below 10^{-4} mBq, independent of the isotope.

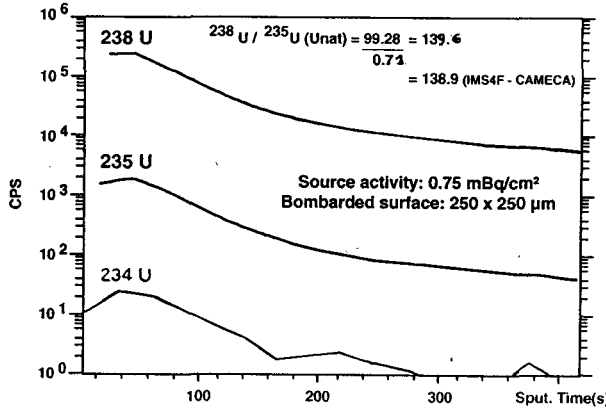

Fig 3 - Depth profile obtained from a specimen containing natural uranium with an activity of 0.75 mBq .

CONCLUSION

This investigation has shown the rapidity of detection using SIMS compared with alpha spectrometry. Whereas further investigations are required to demonstrate the feasibility of this analytical method for accurate determination of specimen activities, the isotopic ratio observed on specimen with known activity and isotopy was found very close to the theoretical value. This indicates the possible application of an internal tracer. Therefore, there is a potential interest for application of the SIMS technique for assessments of internal contamination by analysing long-lived alpha emitters in bioassays. Studies are currently carried out to test the method for analysis of uranium and plutonium in urine samples.

ACKNOWLEDGEMENTS —The authors are grateful to M. Carrier and M. Stéphan (INSTN, France) for the implementation of the PPy technique and are grateful to M. Renard (Cameca, France) for his technical assistance.

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DESIGN AND PERFORMANCE CHARACTERISTICS OF A THYROID COUNTER FOR RADIOIODINE MONITORING

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ABSTRACT

The design and performance characteristics of a thyroid counter using two NaI(Tl) detectors, with 5.08x5.08 cm and 5.08x0.2 cm crystal sizes, respectively for measuring ^{131}I and ^{125}I are described in this paper. Two neck phantoms and a series of standard sources of each radionuclide were applied for efficiency calibration as well as to investigate the effects of thyroid depth in the neck and the source activities on the responses. The results are reported and discussed.

INTRODUCTION

Internal contamination monitoring of persons occupationally exposed to radioiodine, particularly ^{131}I and ^{125}I, is of great concern in radiation protection. The most common method is thyroid counting by applying different techniques either by using a single NaI (Tl) gamma ray detector connected to a scaler or a multichannel analyzer; or by using two NaI (Tl) detectors operating separately or in a coincidence counting mode for ^{125}I (1-6). Due to a need for a thyroid counter system for monitoring of some workers and a high cost to purchase such a system as well as the availability of the components required, a thyroid monitoring system was designed and constructed at the National Radiation Protection Department (NRPD) of the Atomic Energy Organization of Iran (AEOI). In this paper, the design and performance characteristics of this thyroid counter are presented and discussed.

MATERIALS AND METHODS

Two NaI(Tl) detectors of different sizes including a 5.08x5.08 cm detector (I) with 1.6 mm thick Al window (SPA-3, Eberline) and a 5.08x0.2 cm detector (II) with 0.025 mm Al window (PG-2 Eberline), a scaler (MS-2 Eberline), a collimator and its carrier and a chair were used in this study. The detectors were connected to a single channel analyzer and a scaler. The detector I has been mainly used for ^{131}I although its performance for counting ^{125}I photons has also been studied. The detector II has been used only for measuring ^{125}I. The collimator was designed and constructed in our laboratory, as shown in Figure 1, with a wall thickness of about 2.5 cm of lead except at the front side and lined inside with 1 mm of copper sheet. The calibration of the detectors was carried out by a neck phantom made according to ANSI specifications (7). This phantom is a plexiglass cylinder of 12.7 cm diameter and 12.7 cm height whhich has a cylindrical hole for inserting a polyethylene vessel of 30 ml, as shown also in Figure 1, simulating the thyroid gland. The depth of the vessel inside the phantom is 2.2 cm. The distance between the collimated detector and the phantom surface (counting distance) is 5 cm for both detectors. However for the calibration of the detector II for ^{125}I, an extra distance of 10 cm was used by drawing the detector back in the collimator to investigate the effect of counting distance on the detector response. To evaluate the detector efficiency as a function of the amount of activity in the thyroid, for selected counting distances, a series of standard sources of each nuclide with a range of activities from 0.03 to 200 kBq for ^{131}I and from 0.7 to 320 kBq for ^{125}I were used. To evaluate the influence of the thyroid depth in the neck on the counting efficiency, at selected

counting distances, a cylindrical vessel made of plexiglass with dimensions similar to ANSI phantom and filled with water was applied. The depth of thyroid simulator in this water phantom was ranged from 1.9 to 3.4 cm in 5 mm intervals.

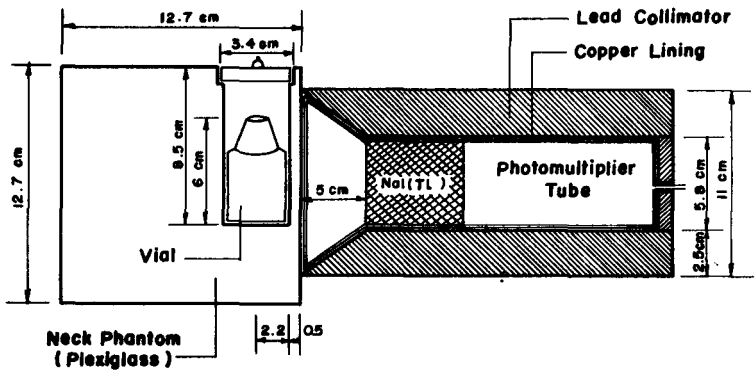

Fig. 1 The schematic diagram of the detector inside the collimator designed and constructed at the NRPD, AEOI.

RESULTS AND DISCUSSION

Figure 2 shows the calibration curves of the detectors for ^{131}I and ^{125}I. Each curve represents the net count rate of the detector as functions of the activity in polyethylene vessel in the neck phantom. The net count rate due to a minimum detectable activity (MDA) for the detector and nuclide of interest is also shown as the first starting point in each curve. The MDA values are calculated based on statistical error of the background count rate of the detectors in counting windows of interest, with a 95% confidence level and 5 minute counting period. As shown in Fig. 2, the calibration curve for each detector and nuclide of interest is linear up to a certain amount of activity (nearly 60 kBq for ^{131}I and 100 kBq for ^{125}I for a 5 cm counting distance). Increasing the counting distance of detector II from 5 to 10 cm, as shown in Fig. 2, resulted in a significant extension of linearity of the calibration curve for ^{125}I from 100 to 320 kBq. The use of stronger sources than those specified in the calibration curves, particularly for 5 cm counting distance, resulted in gradual reduction of counting efficiency of the detectors for both radionuclides. It appears that the reductions are due to dead-time losses in the count rates of the detectors which are not corrected by the scaler. Also as shown in Fig. 2, the MDA value for ^{125}I with detector II due to a higher efficiency and much lower background count rate is about 12 times lower than that of the detector I, for the same counting conditions.

Figure 3 shows the variation of the counting efficiency of the detectors for both radionulides as functions of the depth of the polyethylene vessel in the water phantom, as well as the efficiency values for average depth. Such variations, as shown in Fig. 3, are of more significance for ^{125}I than those for ^{131}I particularly for the 5 cm counting distance (nearly 20% for ^{125}I against 5% for ^{131} I for a 3 mm relative to average depth). In addition, increasing the counting distance from 5 to 10 cm for detector II for ^{125}I results in the reduction of efficiency variation with source depth in the phantom. So, it can be concluded that the use of relatively large counting distances makes the response of detector less dependent on the anatomical features of the thyroid including its depth in the neck as well as the amount of activity in the gland; i.e. the measurement accuracy can be very much improved. Of course the efficiency and therefore measurement sensitivity will be reduced with increasing the distance.

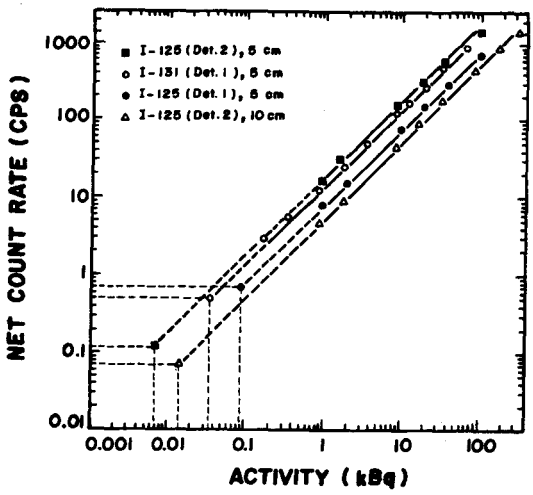

Figure 2. Calibration curves of the detectors for ^{131}I and ^{125}I.

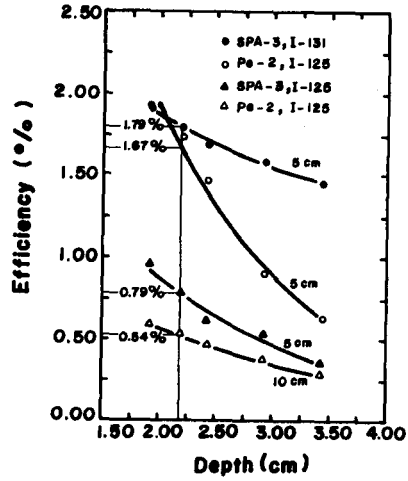

Figure 3. Efficiency of the detectors as functions of vessel depth in the phantom.

CONCLUSION

The above studies show that among the two detectors, detector I and II are respectively appropriate for ^{131}I and for ^{125}I. The use of relatively short counting distances although is appropriate from efficiency point of view, it makes the efficiency of the detectors more dependent on anatomical variations of the thyroid among the individuals particularly for ^{125}I. Therefore, the use of very large counting distances over 10 cm improves the measurement accuracy. Of course for ^{131}I, this problem is not very critical. This monitoring system in general showed to be very practical for our applications and it has been in routine use for near 2 years the results of which will be reported elsewhere.

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DEVELOPMENT OF THYROID MONITORING SYSTEM FOR RADIOIODINE IN THYROID GLAND

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ABSTRACT

A thyroid monitoring system consisting of a high-purity germanium (HPGe) detector and a spectrometry unit has been developed for measurement of thyroid radioiodines which could be released in nuclear accidents.

We report the characteristics of the system and results of the calibration experiment used realistic neck phantom and ORINS type IAEA phantom.

INTRODUCTION

In the early stage of the nuclear accidents, several radioiodines, such as ^{131}I, ^{132}I, ^{133}I, ^{134}I, and ^{135}I could be released together with other radionuclides. The mixture of these iodines varies from dependency on the time after the accident, fuel type, burn-up of fuel and protective measures of a reactor. These radioiodines in the thyroid are predominant cause of the exposure to the public in the early stage, and countermeasures such as sheltering or iodine prophylaxis should be considered. So, the evaluation of thyroid dose is very important for the risk assessment and medical treatment in point, and it is necessary to measure the thyroid burden of each radioiodine for individual dose estimates. The monitoring of iodines in the thyroid are made using lots of portable counters, such as NaI(Tl) thyroid detector (or even GM counters) in hospitals or nuclear institute. These counters should be calibrated correctly for these nuclides, and then the precise monitor for this calibration purpose is needed to be introduced and studied.

EXPERIMENTS

We developed a thyroid monitoring system consisting of a high resolution HPGe detector and a spectrometry unit in portable size as illustrated in Figure 1, and the spectrometry unit makes it possible to transport the system to monitoring places, hospitals and sheltering places. The HPGe detector is surface barrier type with 2 inch diameter and 2cm thickness for low energy photon detection, which is housed in a designed lead collimator of cylindrical shape of the 1 inch thickness. This collimator shields effectively background radiations from the body excluding a thyroid, and it makes the counting response of the detector uniform.

The high resolution detector makes it possible to identify gamma rays from the above nuclides separately and then to estimate the reasonable thyroid doses caused by these isotopes which offer a proper calibration factor the other counters such as GM counter used for actual thyroid iodine measurement of people. The energy resolutions obtained for gamma rays of 122 keV for ^{57}Co, 284 keV and 364 keV for ^{131}I are 0.67 keV, 1.18 keV and 1.25 keV respectively. Figure 2 shows characteristics of the isoresponce curve for different size collimators.

The present study for the calibration of the monitor is made using two phantoms, realistic neck phantom and ORINS type IAEA phantom of adult size. The realistic neck phantom (Photo 1) is made of tissue equivalent substitute for gamma rays and the thyroid model is made of plastic vessel of 20 ml volume for adult size. The obtained detection efficiency for ^{131}I is compared with 3 inch NaI(Tl) detector as shown in Table 1 for two different adult thyroid phantoms, in which 364 keV gamma ray used for evaluation and the distance between detector surfaces and neck surfaces is 5 mm.

CONCLUSION

Nevertheless its relatively low efficiency compared with NaI(Tl) detector, the HPGe detector has an advantage of high resolution which enables to get the proper calibration factor for mixed radioiodine in accident cases, and supports the monitoring of thyroid iodine of the people. Comparing the results for the two phantoms, we found that ORINS phantom is adequate enough for such a monitoring purpose.

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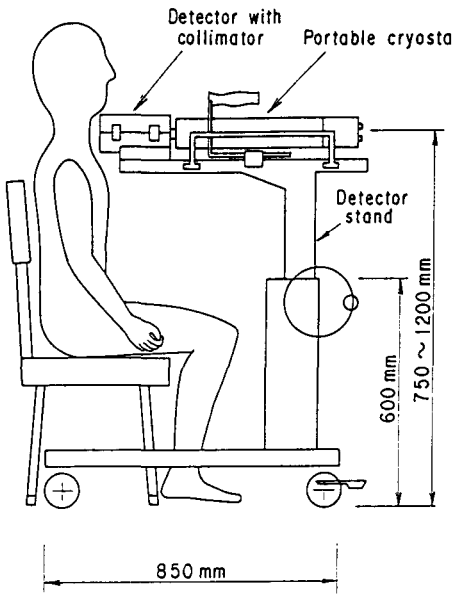

Detector : Hyper-pure low energy Ge detector (CANBERRA ACT-I)
Active area : 20 cm^2 , Thickness : 20mm
Energy resolution : 640 eV for 122 keV γ-ray

Figure 1. Illustration of thyroid monitor in counting position of human thyroid.

Photo 1. Overview of realistic neck phantom.

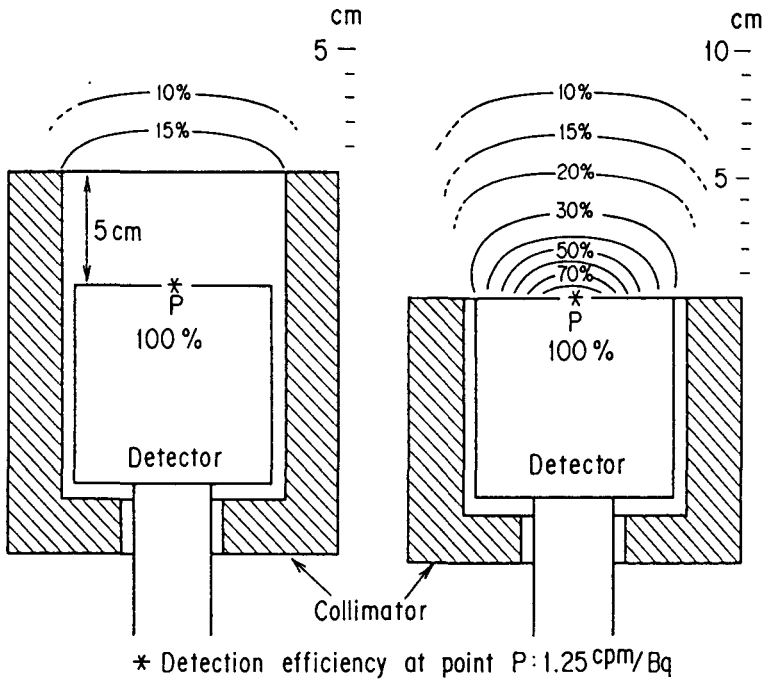

Figure 2. Isoresponse curves for different size collimators. (The relative values are normalized by detection efficiency at center of detector : *P.)

Table 1. Results of calibration experiment of thyroid iodine detectors using neck phantoms

Neck phantoms	detection efficiency(cpm/Bq)	
	2" ϕ HPGE detector	3" ϕ NaI(Tl) detector
Realistic neck phantom	0.15	3.10
ORINS neck phantom	0.16	3.19

EFFECTIVE DOSE FROM INTAKE OF SOME RADIOLOGICALLY SIGNIFICANT RADIONUCLIDES IN INDIAN POPULATION

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4. CCE(R&D), Secundrabad - 500 003, India.

ABSTRACT

ICRP-67 gave the age-dependent dose coefficients to the members of the public from ingestion of most radiologically significant radionuclides taking into account the various anatomical and physiological characteristics of ICRP Reference Man, Child and Infant. In this regard, Indians differ significantly from their ICRP counterpart. In this paper, efforts have been made to estimate effective dose from ingestion of some radionuclides viz. ^{95}Zr, ^{95}Nb, ^{132}I and ^{134}Cs taking into consideration the organ weights of Indian adult and children of various age groups. The effective dose estimates have been found to be marginally higher as compared to the corresponding values for their ICRP conterpart.

INTRODUCTION

Radiation dose to members of the public from ingestion of most radiologically significant radionuclides, that might be released to the environment due to various human activities are given in ICRP - 67 (1). The ingestion dose coefficients for various organs and tissues have been computed taking into account the various anatomical and physiological characteristics of ICRP Reference Man, Child of various age groups and Infant. In this regard, Indian population differs significantly from their ICRP counterpart (2-4). In the present study, efforts have been made to estimate effective dose from ingestion of some radionuclides viz. ^{95}Zr, ^{95}Nb, ^{132}I and ^{134}Cs for Indian adult and childern of various age groups. These effective doses have been compared with the corresponding values as given by ICRP-67.

METHODOLOGY

The weight of five organs, namely, brain, kidney, liver, lungs and spleen was obtained from postmortem records of 3000 accident death cases in Indian population including children of various age groups (3). These organ weights were utilised to estimate dose coefficients in Indian population of various age groups namely 1y, 5y, 10y, 15y and adult by computing the dose transformation factors (DTF) for different organs using the method

suggested by Yamaguchi (5). The dose coefficients to various organs as given in ICRP-67 are multiplied by the corresponding dose transformation factors to give the revised dose estimates for the organs in the Indian population. In the present study, the effective dose has been estimated from ingestion of four radionuclides viz. ^{95}Zr, ^{95}Nb, ^{132}I and ^{134}Cs. The dose transformation factors for the various organs have been estimated for the principal gamma energy emitted by these radionuclides. Thymus is taken as a surrogate to oesophagus. The contribution to the effective dose from thymus as one of the remainder organs is not taken into account as its contribution has been already included while considering contribution from oesophagus. The equivalent dose to colon and gonads is computed in the manner similar to that adopted by ICRP-67 (1).

The effective dose was computed from ingestion of various radionuclides for Indian population utilising the transformation factors as computed for five selected organs mentioned above. For the other organs, if the weight of the organ is less than or equal to 20 g, DTF is taken as 1 and for the others, DTF has been computed considering the organ masses to be proportional to the body weight. For comparison, the relative difference between the E values for ICRP population to that of Indian counterpart has been worked out and expressed as %age change.

Table 1. Effective dose from ingestion of radionuclides

Effective dose(Sv/Bq)	Age at Intake				
	1y	5y	10y	15y	Adult
^{95}Zr					
E_1	5.7E-9	3.1E-9	1.9E-9	1.2E-9	9.6E-10
E_2	6.4E-9	3.7E-9	2.4E-9	1.4E-9	1.1E- 9
^{95}Nb					
E_1	3.2E-9	1.8E-9	1.2E-9	7.4E-10	5.9E-10
E_2	3.5E-9	2.1E-9	1.4E-9	8.4E-10	6.7E-10
^{132}I					
E_1	2.4E-9	1.3E-9	6.2E-10	4.2E-10	2.9E-10
E_2	2.5E-9	1.4E-9	6.7E-10	4.5E-10	3.1E-10
^{134}Cs					
E_1	1.6E-8	1.3E-8	1.4E-8	1.9E-8	1.9E-8
E_2	1.7E-8	1.4E-8	1.5E-8	2.0E-8	2.0E-8

E_1 : Effective dose as given by ICRP-67 (1)
E_2 : Effective dose estimates for Indian population

Table 2. Relative Difference in Effective Dose Estimates for Indians to that of ICRP Population (%

Radionuclide	Age at Intake				
	1y	5y	10y	15y	Adult
^{95}Zr	10.9	19.4	26.3	16.7	14.6
^{95}Nb	9.4	16.7	16.7	13.5	11.9
^{132}I	4.2	7.7	8.1	7.1	6.9
^{134}Cs	6.3	7.7	7.1	5.3	5.3

RESULTS & DISCUSSION

Table 1 gives the effective dose (E_1) as given by ICRP-67 and the estimated value of ^{95}Zr, ^{95}Nb, ^{132}I and ^{134}Cs. Table 2 gives the %age change in effective dose estimates for Indian population as compared to the effective dose values for ICRP population. In case of ^{95}Zr and ^{95}Nb, an increase of 9-26% in effective dose values is found as the maximum contribution comes from colon whose DTF is computed considering its mass proportional to body weight. However, in case of ^{132}I and ^{134}Cs, the increase in E values are less (4-8%). In case of ^{132}, the increase in E values is less as the maximum contribution to comes from thyroid, whose weight in Indians is considered to be the same as that of ICRP counterpart. However, the dose coefficients for ^{134}Cs is nearly same in all organs.The effective dose estimates for Indian adult and childern of various age groups are higher than their ICRP counterpart as the organ masses in Indians are smaller as compared to ICRP.

In conclusion, it may be said that the revised effective dose estimates for Indian adult and child of various age groups were marginally higher as compared to the corresponding values for their ICRP counterpart for the four radionuclides considered.

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IAEA-RCA CO-ORDINATED RESEARCH PROGRAMME: COMPILATION OF ANATOMICAL, PHYSIOLOGICAL AND METABOLIC CHARACTERISTICS FOR A REFERENCE ASIAN MAN

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INTRODUCTION

The world-wide radiation protection concerns that followed the Chernobyl accident created an increased need for basic data to assess internal doses to members of the public in each country. Therefore, the Reference Asian Man Programme was proposed by Japan and initiated as a part of the IAEA Regional Cooperative Agreement Project for Strengthening Radiation Protection Infrastructure to coincide with the revision of Reference Man data by ICRP Committee 2 (1, 2, 3).

The first priority was given to compilation of data on physique, and mass of internal organs where possible, for populations throughout the Asian region. Data on daily consumption of foods and nutrients were also emphasized since they are important components of the ingestion pathway for uptake of radionuclides by humans. Pulmonary function and water balance parameters were considered important in relation to the inhalation pathway and water contamination (4). The programme was also endorsed by ICRP.

MATERIALS AND METHODS

Physical measurements

Data on the body height and weight, sitting height, head and neck circumference, and circumference, width and depth of the chest were compiled or measured in 9 countries. Age ranges assigned were the newborn, 1, 5, 10, 15 and 20-50y (2, 4) which apply to all types of data in the CRP. The source of data was recent national statistics (e. g. school health statistics, national nutrition survey, etc.), relevant scientific papers or specific measureent programme. The sample size ranged from 865 to 1.01×10^6.

Organ mass measurements

Masses of the brain, heart, kidney, liver, lung, pancreas, pituitary gland, spleen, testis, thymus and thyroid gland in the male, and the above and the breast and ovary in the female were studied in 7 countries, 4 of which also compiled data on relative weights of individual organs to the body weight. The m ain partof the potocol was as follows: autopsy should be carried out by qualified medical doctors within 24 h after sudden deaths and with exclusion of data showing any pathological changes which affect normal weights of internal organs. Data were obtained from a medical examiner's office, forensic departments, medical colleges and hospitals. The number of cases ranged from 155 to some more than 20, 000.

Nutritional data

The daily dietary intake of categorized foods (cereals, pulses, potatoes, vegetables, fats and oils, fish and shellfish, meats, milk and milk products, etc.) and principal nutrients (carbohydrate, protein, fat or lipid, energy) were studied in 8 countries using national statistics and special programs of limited scales. Determination of elements for intake study was left to each country due to available resources and manpower.

Physiological parameters

The vital capacity, total lung capacity, minute volume and 8-hr working volume were studied in 4 countries, for different age groups and three activity levels, mostly in hospital personnel and university students. Other data were extracted from routine medical examination files or from executive check-ups. For water balance, measurement was carried out on the 24-hr intake and elimination of water in the urine, sweat, breath and faeces, in healthy volunteer subjects also in 4 countries.

RESULTS AND DISCUSSION

Physical measurements

Some of the compiled data on physical measurement, e. g. body height, weight and sitting heiht are shown in Table 1. Mean and population-weighted mean, range and rounded mid-point for height and weight in the adult male and female were also obtained. Most participants agreed that the range mid-points would

Table 1. National means of body height, weight and sitting height of the adult (20-50y)

		BAN	CHI	IND	JPN	KOR	PKS	PHI	VIE	Midpoint *
Male	Height (cm)	163. 9	169. 2	163. 4	167. 8	166. 8	170. 6	163. 4	163. 8	167. 0
	Weight (kg)	57. 8	58. 3	51. 5	63. 6	63. 8	63. 9	56. 6	51. 8	57. 7
	Sitting ht. (cm)	85. 7	91. 8	85. 8	89. 4	90. 1	-	86. 0	86. 2	88. 8
Female	Height (cm)	154. 9	158. 2	151. 0	155. 0	154. 9	157. 5	151. 3	154. 0	154. 62
	Weight (cm)	49. 9	51. 1	44. 2	52. 3	54. 5	52. 6	49. 2	46. 8	49. 4
	Sitting ht. (cm)	80. 6	86. 2	80. 0	84. 1	83. 9	-	80. 3	81. 2	83. 1

*) Including Indonesia.

represent most reasonable compromise if a single value is to be used in practical aspects. These results on heights and weights were compared with those of revised ICRP Reference Man, 170cm and 73kg for the male and 163cm and 60kg for the female. In revised Tanaka Model, these are 170cm and 60kg, and 160 cm and 51kg in the adult male and female, respectively.

For intra-regional variation, the data might be grouped in the two: (a) China, Japan, Korea and Pakistan, and (b) Bangladesh, India, Philippines and Vietnam. For instance, the average body height is 5cm higher in the former group (168. 6cm) than the latter (163. 6cm). The lower mean height and weight from developing countries are attributable to a large proportion of the populations having various degrees of undernutrition. Secular trends in the body height and weight were noticed for Japan, China, Korea during the last decades. Assuming secular trends of 1-1. 5cm in 10 years for height during the period from 1985 to 2005, possible reference values would be 169-170cm for the adult male, and 157-158cm for the female.

Organ mass measurements

Mean and S. D. s of organ masses in the adult were calculated to compare the results with values of the ICRP Reference Man and Tanaka Model and some of the data are shown in Table 2. Accuracy of measurement can be adversely affected by several factors (presence of blood, remaining extraneous tissues, delay in post-mortem measurement, etc.). The brain mass in the male had a mean of 1361 g with a small variation (RSD 5. 7%) as the testis, 33. 6g (RSD 5. 4%); the liver and lung showed a mean (RSD), 1429g (17. 8 %) and 976g(20. 1%), respectively. Mass of the thyroid gland widely varied from 16. 8 to 35. 5g and from 16. 8 to 36. 0g for the male and female, respectively, which may presumably be associated with variation in natural iodine intakes.

Table 2. Reported means of mass of the brain, liver and thyroid gland in the adult (20-50y), g

		CHI	IND	IDN	JPN	KOR	PHI	VIE	Reference Man	Tanaka Model
Male	Brain	1433	1236	1345	1442	-	1387	1321	1400	1470
	Liver	1357	1135	1156	1599	1864	1472	1418	1800	1600
	Thyroid	27.4	19	16.8	18.8	-	22	35.5	20	19
Female	Brain	1317	1140	1203	1309	-	1321	1284	1200	1320
	Liver	1272	1051	1115	1345	1611	1361	1319	1400	1368
	Thyroid	26.6	-	17.8	16.8	-	30	36	17	17

In the male, ten organs showed comparable or smaller masses than those for ICRP Reference Man while the pancreas and thyroid weights were larger than the Reference Man's. In the female, the mean mass -es ofeight organs were smaller than or comparable to Reference Woman's while larger for the brain, heart, pancreas, thyroid and adrenal gland. Influences of general nutritional conditions on masses of organs were also shown in the systematic study for normal Japanese (5).

Mean fractional masses reported for many organs by Japan, India and China may indicate a little lrger contribution of such organs to the total body weight in Asian populations than in Westerners.

Nutritional data

Daily consumption of cereal varied from 171 to 520g as compared with 207g for Reference Man. That for meat, milk and milk products, was 12-172 and 9-147g, respectively (5 countries) in contrast to 228g and 508g, respectively in Western European and North American countries. Fat or lipid is much less consumed in the diet (15.8-56.2g against 120g in the Western diet). The levels of calcium intake reported were approximately one half of that for Reference Man. Information on intake of 9 other elements was also obtained.

Physiological parameters

Total lung capacity ranged 4.43-5.76 liters (range mid-point 5.10) and 3.28-4.40 (3.84) liters in the male and female, respectively; mid-point of minute volume being 6.6 and 5.1 l/min in 5 countries. These are approx. 85-91% of the corresponding ICRP values. Water intake ranged from 3.31-4.53 (mid-point 3.53) l/d in the male and 2.14-3.60 (2.87) l/d in the female among 4 countries as compared with 3.00 and 2.10 l/d of Reference Man and Woman. Measurement of water balance was made difficult due to multitude of mechanisms for elimination. The particular values and pathways are dependent on ambient and environmental conditions. This is important in the tropical conditions that are prevalent in many Asian countries.

CONCLUSION

A great amount of data on physical measurements was accumulated in this research programme along with essential information on the organ masses in Asian populations was obtained, that are useful for internal dosimetry. Data on consumption offood groups, principal nutrients and some elements was provided, suggesting typical Asian dietary habits. Some information on pulmonary function and water balance was also obtained. The results of the CRP also contributed to revision of ICRP Reference Man.

However, due to limited physiological and metabolic data, intake of elements of importance in radiation protection, representative of each populations, particularly need to be studied further.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Tritium Level in Blood as a Measure to Determine the Dose due to Organically Bound Tritium.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 3.1

ABSTRACT (See instructions overleaf)

A significantly large contribution of the organically bound tritium (OBT) to the total absorbed dose in human body after tritium intake has been emphasized by many authors. In the present research, an animal experiment using mice was performed to find possible correlation between the tritium level in the blood and that averaged for the total body after intake of OBT. A mixture solution of five tritiated amino-acids was administered to male DDY mice and the tritium level in the blood, urine, total body, spleen, liver , brain, kidney and testis was determined after various periods. It was found that there is a strong correlation between the tritium level in the blood and the body-averaged tritium concentration. The accumulated dose that is estimated on the basis of the blood tritium level gave a conservative estimation for the body-average accumulated dose. The dose commitment estimated from the tritium level in the urine gave an estimated dose for 55 days periods which is about 60 percent of the body-averaged dose.

ELECTRON PROBE X-RAY MICROANALYSIS OF INHALED (UO_2, PuO_2) MIXED OXYDES IN ALVEOLAR MACROPHAGES OBSERVED *IN TOTO*

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INTRODUCTION

Microanalysis has been mainly used to characterize chemical composition of uranium compounds retained *in vivo* (1) but, to our knowledge, similar experimental approach on other actinides has not yet been performed. However, some results are available on native (UO_2, PuO_2) mixed oxyde powders containing 18 % Pu which corresponded to a solid solution of $(U, Pu)O_{1,96}$ (2). We have previously shown that chemical composition of mineral particles phagocytosed by alveolar macrophages can be performed on entire cells by using transmission electron microscopy and energy dispersive X-ray microanalysis (EDX) (3). The aim of this work was to study the chemical composition of inhaled particules in alveolar macrophages observed *in toto* after exposure of rats to (UO_2, PuO_2) mixed oxyde aerosols containing 5.3 % Pu.

MATERIAL AND METHODS

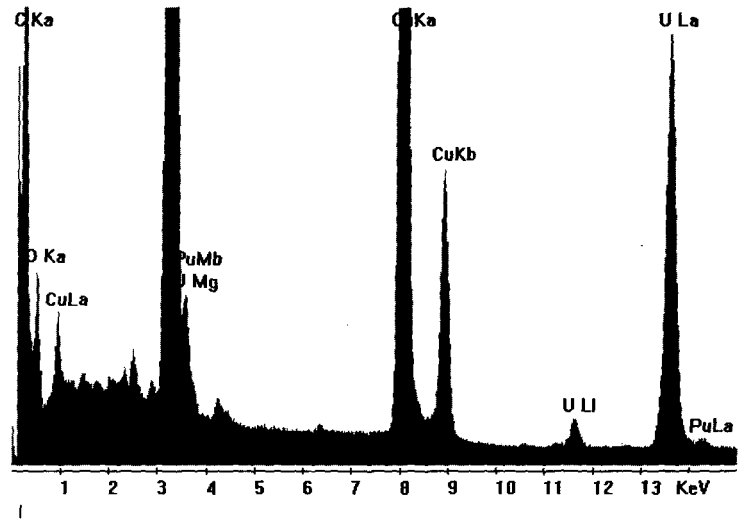
Male Sprague-Dawley rats were exposed at 3 months of age to (UO_2, PuO_2) mixed oxydes aerosols as previously described (4). Animals were killed during the first week after exposure under pentobarbital anesthesia (40 mg/kg). Lungs were lavaged by repeated instillations of 5 ml NaCl 0.9 % until a total volume of 20 ml had been collected. Cells from 0,4 ml of the lavage fluid were put on collodium coated slides using a cytospin. They were fixed during 30 minutes in 70% ethanol and the collodium film was put on 300 mesh copper grids which were coated with carbon.

Entire cells were observed with a CM200 Philips electron microscope at 200 kV with the scanning transmission mode. Microanalysis was performed using EDAX silicon detector with the super Utw window which allowed us to detect any element with atomic number down to that of beryllium. Most of the EDX analysis were performed at a 300.000 magnification using a spot size of 14 nm.

RESULTS

Preliminary EDX analysis were performed on either uranyl hydroxyde particles directly deposited on coated grids or on unexposed alveolar macrophages. The spectra obtained for particles have shown specific rays of U without any Pu-ray whereas, no specific U or Pu-ray was observed in control alveolar macrophages.

After exposure to (UO_2, PuO_2) mixed oxydes, most of the particles had a diameter less than 0.3 µm and were clearly identified at magnifications down to 5000 with high-angle annular dark-field observation. This was due to the discrimination according to the atomic number of the elements using this "dark field" observation mode which provided more contrasted images than usual "bright field" observation mode.



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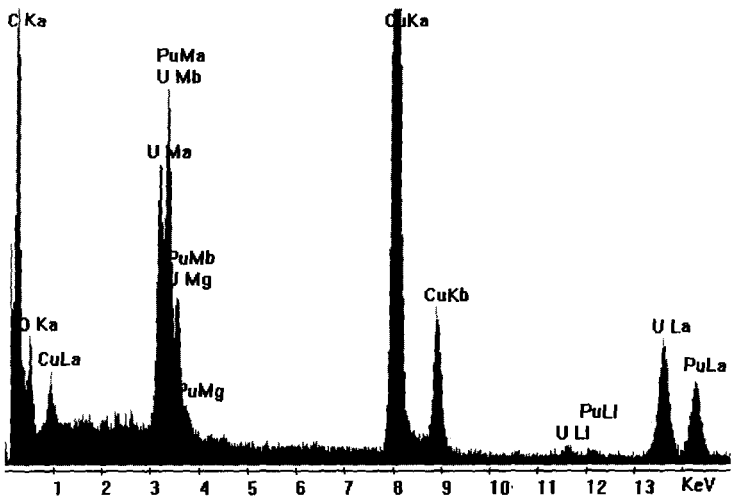

1B

Figure 1: example of EDX spectra obtained on two different (UO_2,PuO_2) particles

Figure 1 shows spectra observed on two different (UO_2, PuO_2) mixed oxyde particles. Only energies up to 15 keV are shown in which specific Mα, Mβ, Mγ and Lα rays of actinides are encountered. In this energy range, specific Kα and Kβ Cu-rays are also observed and correspond to the background due to the chemical composition of the grid. Both actinide specific rays can be observed but the U-Mβ and the Pu-Mα, and the U-Mγ and the Pu-Mβ are superposed. The Lα-ray intensity values of Pu and U allowed to estimate the actual relative concentration of these actinides within the particle. In Figure 1A, the U/U+Pu ratio is about 0.95 which corresponds to the expected composition of the mixed oxyde studied. However, for some particles, a quite different ratio was measured as for example in Figure 1B, with a ratio at about 0.6. This phenomenon could be observed according to the particle analysed and even according to the site analysed within the same particle.

Spectra up to 2 keV, including usual organic elements were not very different if they were performed in the alveolar macrophage on or out of the particle and contained the specific rays for C, O, N, S, P... which were due to the presence of the biological matrix.

DISCUSSION

These preliminary results demonstrate that EDX is a useful tool to characterize the chemical composition of inhaled particles containing actinides especially (UO_2, PuO_2) mixed oxydes. Because the measurements are performed on alveolar macrophage observed *in toto,* quantitative results are easily obtained for experimental purposes in rodents but could also be obtained in primates. Moreover, this analysis is potentialy possible on humans after pulmonary lavage without any technical modification. However, these measurements need very specialized and costly equipment and are very time consuming.

Because particles are within a biological matrix, the experimental procedure used did not allow us to determine the actual chemical composition of the particle including O and other element which could be part of the particle once it has been phagocytosed such as phosphorus for example. Further studies are in progress in order to extract phagocytosed particles for determination of their actual chemical composition

These preliminary results obtained with mixed (UO_2, PuO_2) oxydes i.e. apparent heterogeneous distribution of Pu, are difficult to explain. Other experiments are needed including mapping of actinides and autoradiography which are in progress. Thus, once these complementary methods will be available, our experimental approach might be usefull both to characterize experimentally, the behaviour of any inhaled actinide oxyde or mixed oxyde and to analyse human samples.

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THE AUGER ELECTRON EFFECT IN RADIATION DOSIMETRY - A REVIEW

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ABSTRACT

Radionuclides that emit Auger electrons are widely used in nuclear medicine and biomedical research. The Auger emitting radionuclides give off a cascade of low energy electrons, the total energy of which is deposited within some nanometers; the local dose is therefore very high. If the radionuclide is part of a chemical compund which forms a DNA base analogue or which preferentially enters the cell nucleus, the biological effects of the Auger electrons to the DNA or the nucleus can be as severe as from high-LET alpha particles. Neither the recommendations of the International Commission on Radiological Protection (ICRP) in Publication No. 60, nor the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources provide any guidance on calculating the equivalent dose for these radionuclides. For the radiation weighting factor of electrons (value 1) they recommend excluding the Auger electrons emitted from nuclei bound to DNA. However, recently a Task Group of the American Association of Physicists in Medicine (AAPM) has proposed that the component of dose from the Auger electrons for radionuclides bound to DNA be given a preliminary radiation weighting factor of of 10 for deterministic effects and 20 for stochastic effects. The dose equivalent calculated with these weighting factors must be modulated by experimentally determined subcellular distributions.

INTRODUCTION

Radionuclides that emit a high proportion of Auger electrons are widely used in nuclear medicine (e.g. ^{99m}Tc, ^{123}I, ^{201}Tl) and biomedical research (e.g. ^{51}Cr, ^{125}I). Natural radioactive isotopes exist with Auger electron emissions (e.g. ^{40}K). In nuclear weapon debris exists inter alia the isotope ^{55}Fe and in the nuclear energy cycle the isotope ^{65}Zn. The dosimetry of Auger electrons and other low energy radiations has been discussed in ICRU Report 32 (1). Recent reviews of the Auger electron effect are found in two articles are by Persson (2,3).

ICRP discussed the Auger electron effect in its Publication No. 60 (4). On page 6, paragraph 26 they state: "Auger electrons emitted from nuclei bound to DNA present a special problem because it is not realistic to average the absorbed dose over the whole mass on DNA as would be required by the present definition of equivalent dose. The effects of Auger electrons have to be assessed using the techniques of microdosimetry". In the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (5), it is stated that Auger electrons emitted from nuclei to DNA are excluded from the radiation weighting factor of value 1 for electrons of all energies with the remark that special microdosimetric considerations apply.

MEDICAL APPLICATIONS

There appears to be two important issues for Auger electron emitters in medicine. At the root of both is that characteristic of the Auger electron decay: the highly localized irradiation of the surrounding volume. Considerable exposure is delivered to the part of a cell or a macromolecule which is in the vicinity of the decaying nuclide. This on the one hand brings about a problem in risk estimation for nuclear medicine and on the other hand promises of a selective attack on cancer cells: "molecular surgery".

The following isotopes (normally bonded in a chemical compound) are of special interest in nuclear medicine as Auger electron emitters: ^{51}Cr, ^{55}Fe, ^{67}Ga, ^{75}Se, ^{77}Br, ^{80m}Br, ^{99m}Tc, ^{110}In, ^{111}In, ^{114m}In, ^{123}I, ^{125}I, ^{145}Sm, ^{193m}Pt, ^{195m}Pt, and ^{201}Tl. Some of these radionuclides may also have a role in cancer treatment.

EXPERIMENTS WITH LIVING CELLS

Rao et al. (6) have studied the radiotoxicity of ^{125}I-iododeoxyuridine (IUdR) by the determination of the survival of spermatogonial cells of mice. Narra et al. (7) studied the same issue by investigating the survival of pre-implantation mouse embryos. Iododeoxyuridine is a thymidine analogue and incorporates into the DNA of proliferating cells. ^{125}I incorporated into DNA was as effective as densely ionizing 5.3 MeV α-particles from

^{210}Po in reducing the sperm head population in mice. The embryo survival curves show that the dose at 37% survival is only about 0.15 Gy for ^{125}IUdR, whereas for 662 keV gamma rays from ^{137}Cs, it is 1.75 Gy. These results are consistent with the observations in mouse testis and cultured cells and point to the need for assessing the radiation risk from incorporated Auger electron emitting radionuclides based on their sub-cellular distribution. Also ^{125}I-labelled DNA binding agents other than ^{125}IUdR have been shown to cause severe damage to the DNA molecule, as discussed by Ludwikow et al. (8).

EQUIVALENT DOSE FOR AUGER ELECTRON EMITTERS

Howell et al. (9) stated: "Depending on the subcellular distribution of the radionuclide, the biological effects caused by tissue-incorporated Auger emitters can be as severe as those from high-LET alpha-particles. However, the recently adopted recommendations of the International Commission on Radiological Protection (4) provide no guidance with regard to calculating the equivalent dose for these radionuclides. The present work, using spermatogenesis in mouse testis as the experimental model, shows that the lethality of the prolific Auger emitter ^{125}I is linearly dependent on the fraction of the radioactivity in the organ that is bound to DNA. This suggests that the equivalent dose for Auger emitters may have a similar linear dependence. Accordingly, a formalism for calculating the equivalent dose for Auger emitters is advanced within the ICRP framework".

The equivalent dose in an organ or tissue T is defined as $H_T = w_R \cdot D_{T,R}$, where w_R is the radiation weighing factor, and $D_{T,R}$ is the absorbed dose in the tissue from radiation R. For a mixed radiation field, such as those generated by many radionuclides including Auger emitters,

$$H_T = \sum_R w_R \cdot D_{T,R} . \qquad (1)$$

Howell et al. (9) propose that the equivalent dose specifically for the Auger electrons may be expressed as:

$$H_{T,R(Auger)} = (1 + f_o(w_{R(Auger)} - 1)) \sum_{R(Auger)} D_{T,R} . \qquad (2)$$

where f_o is the fraction of the radioactivity in the organ bound to DNA. This equation limits appropriately at $f_o = 0$ and $f_o = 1$. Although this equation is fundamentally sound, separation of the biological effects of the Auger electrons from those of other radiations emitted by the radionuclide is not possible experimentally because the observed RBE values are for the composite spectrum of emissions. Therefore, it is difficult to assign a value to w_{Auger} that corresponds directly to measured RBE values.

In Report no. 3 of American Association of Physicists in Medicine - AAPM - Nuclear Medicine Task Group no. 6 methods of Auger electron dosimetry at the DNA, cellular, multicellular, and organ level are discussed (10). This Task Group recommends a preliminary value of 10 be used for $w_{R(Auger)}$ in equation (2) to obtain the deterministic equivalent dose H_T for prediction of therapeutic outcome and a value of 20 for stochastic effects. The dose equivalent calculated with these radiation factors must be modulated by experimentally determined subcellular distributions. It should be noted that equation (2) is based on experiments where ^{125}I is covalently bound to DNA in the cell nucleus. When the Auger emitter is localized in the nucleus but not covalently bound to DNA, somewhat lower RBE values may be expected. The equivalent dose from the Auger electrons may then be a factor of 2 lower.

Calculations of equivalent dose for Auger electron emitting radionuclides distribution in humans organs have recently been performed (11). These calculations show an increase in the mean equivalent dose for Auger electron emitters when a significant fraction of the organ activity localizes in the DNA.

CONCLUSIONS

The AAPM Task Group (10) recommends use of radiation weighting factors for cellular and organ dosimetry in conjunction with equivalent dose formalism that takes the subcellular distribution distribution of the Auger emitter into account. Based on the currently available radiobiological data which show that the effects caused by the Auger emitters are similar to those of incorporated alpha emitters, a preliminary radiation weighting factor of 10 is recommended for deterministic effects (i.e., cell survival) and a value of 20 is recommended for stochastic effects (i.e., risk assessment for cancer induction). The dose equivalent calculated with these weighting factors must be modulated by experimentally determined subcellular distributions.

There are good reasons to consider the Auger electron effect not only in medical radiation protection of patients but also in the context of annual limits of intake for workers and the public. It may also be prudent to review the current equivalent dose estimates for radiopharmaceuticals labeled with Auger electrons.

Further experimental and theoretical, radiobiological research in the field is advocated by the author. Inter alia studies of survival rates, chromosome aberrations, micronuclei formations, and cell tranformations, performing animal carcinogenic experiments and mathematical modeling are important tools for a deeper understanding of the subcellular structures and also for the processes involved in the interaction of radiation with biological materia. The scientific basis for radiation protection, especially against the Auger emitting isotopes, will then also improve.

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SIMPLIFIED EXCRETION FUNCTIONS FOR RECENT RECYCLING BIOKINETIC MODELS

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Abstract

Recent trends in biokinetic model development have seen the replacement of a simple, schematic one-way transfer of materials between compartments with more realistic and complex linkages, including "recycling". In its Publications 67 and 69, ICRP proposes the use of such models for Am, Np, Pu, Ba, Ra, Sr, Pb, Fe, Th, U and I. These models have been developed primarily with a view to providing time-dependent organ retention data which may be used as the basis for calculations of effective dose. However, these new models may also be of value in predicting faecal and urinary excretion rates, particularly if models are adapted for non-standard subjects such as children. In this paper simplified functions for organ retention and instantaneous excretion rates are obtained from the above models, and the latter are compared with excretion functions given in ICRP Publication 54[1]. The potential for using future biokinetic models to predict urinary and faecal excretion rates is discussed.

1 Background

The physiologically-based biokinetic models recommended in recent ICRP Publications have the potential for becoming useful tools in bioassay assessments. In most cases, excretion data are fed into the models at a development stage, since such data are relatively abundant. Models therefore carry within them bioassay information obtained from experiment, in addition to physiological information embedded within their formulation. It is possible to adapt the models for non-standard subjects when good physiological data is available. For example, bioassay data is not generally available for children, but copious physiological data does exist, allowing biokinetic models to be adapted for children.

The biokinetic models of ICRP consist of compartments which represent tissue groups of interest, linked by first order kinetics. Calculating retention and excretion from the simple, Bateman[2] type models of ICRP Publication 30[3] is straightforward, as general analytical solutions may be obtained. For models of the type represented by the recent recycling models of ICRP Publications 67[4] and 69[5] however, general solutions are not available and alternative methods are used. In this paper, an exact, eigenvalue method is used, which obtains organ retention functions in the form of the sum of decaying exponentials. These expressions, typically containing as many terms as there are compartments in the models, are reduced to shorter, approximate expressions by a least squares function minimisation procedure[6].

Exact curves of urinary and faecal excretion are obtained from the models of ICRP Publications 67 and 69 for injection of unit quantities of Am, Np, Pu, Ba, Ra, Sr, Pb, I, Fe, Th and U. Simplified excretion function are published elsewhere[7].

2 Results and Discussion

The models may be placed in two main categories; generic actinides, including Am, Np, Pu and Th, and generic alkaline earths, including Ba, Ra, Sr, Pb and U. The models for the remaining elements are treated individually. Plots of the excretion curves show that the functions can depart markedly from those of ICRP Publication 54. These differences are discussed briefly for a few representative elements below.

2.1 Generic actinide models

Urinary excretion rates predicted by the biokinetic models of ICRP Publication 67 for plutonium and americium are on the whole higher than the rates given by ICRP Publication 54, with the difference being particularly marked at times remote from the intake (see figure 1) e.g. for plutonium the difference approaches an order of magnitude at 10,000 days. The Pu data of the earlier report was obtained from experiments with humans by Langham[8] and later animal experiments reported by Durbin[9]. Good recent human volunteer data has become available however[10], and this has been incorporated in the Publication 67 biokinetic model. The rates predicted by the model

are therefore closer to the recent data than those of Publication 54. Faecal excretion rates predicted by the biokinetic models are also characterised by higher rates at times remote from intake (approximately t > 10years), though they do dip noticeably below those of Publication 54 in the range 1 - 100 days. These data, too, are consistent with recent experiments with humans.

2.2 Alkaline Earth models

The elements for which the generic alkaline earth biokinetic model is applied (Sr, Ra, Ba, Pb and U) generally follow the biochemical pathways of calcium in the body. A good deal of human and animal excretion data are available for all of these elements and much of these have been incorporated in the formulation of the models. To take an example, the excretion curves given by the model for strontium are generally similar to those in ICRP Publication 54, but are on the whole smoother (see figure 2). Differences of up to a factor of 3 occur at particular times. Simplified excretion functions given for strontium are so close to the exact functions as to be indistinguishable on a plot.

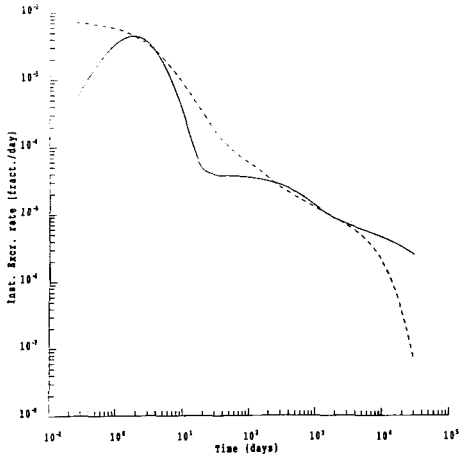

Figure 1 Faecal excretion rate of Pu: ICRP Publication 67 model (solid line) against Publication 54

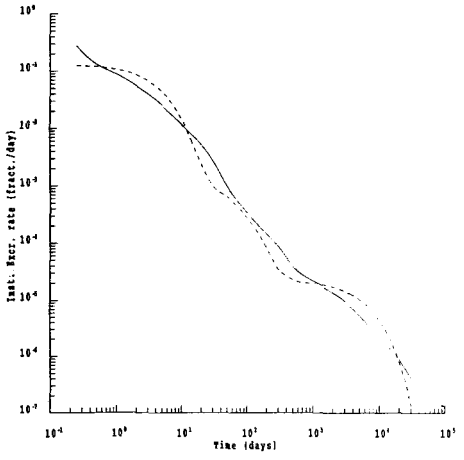

Figure 2 Urinary excretion rate of Sr: ICRP Publication 67 model (solid line) against Publication 54

3 Revised estimates of doses

Upon measuring levels of a radionuclide in urine or faeces at a particular time, a common procedure is to estimate the excretion rates and then use the function/curve for time dependent excretion rates to estimate the size of the original intake of the radionuclide. The latter is then used in conjunction with a dose coefficient to obtain an estimate of the effective dose likely to result from the intake.

From the viewpoint of a user of the excretion functions given in NRPB-M564[7] an important question would seem to be: "Where the new excretion rates differ substantially from the rates of Publication 54, will there also be a substantial differences in doses estimated from bioassay ?". Effective doses are calculated for an observed faecal excretion rate of 200 Bq/day 10 days after an intake of Pu-239 and a urinary excretion rate of 200 Bq/day 30 days after intake of Sr-90 using the excretion curves of both ICRP Publication 54 and the Publication 69 model. The results, given in table 1, demonstrate that the differences in the excretion rates predicted by Publications 54 and 67 are translated to differences in estimations of effective dose. As a general rule, effective dose varies with excretion rate.

Table 1 Effective doses for Pu-239 and Sr-90 following typical bioassay analysis

	Estimated intake Publication 54	Estimated intake Publication 67	Effective dose Publication 30	Effective dose Publication 67
Pu-239	2.22×10^5 Bq	1×10^6 Bq	1.22×10^2 Sv	4.93×10^2 Sv
Sr-90	2×10^5 Bq	6.67×10^4 Bq	17.6×10^{-3} Sv	5.85×10^{-3} Sv

Note that when this procedure for working back to doses is adopted it is necessary to use the appropriate dosimetric model for a particular excretion function. In the example given above, the urinary excretion rates for uranium given in ICRP Publication 54 were obtained from the biokinetic model presented in Publication 30. This same model should therefore be used to estimate the intake and effective dose.

4 Conclusions

Instantaneous excretion rates are obtained for the biokinetic recycling models of ICRP Publications 67 and 69 and are expressed as simple exponential functions. These rates are based on models which have in most cases been adjusted against recent human excretion information and can differ substantially from the rates given in ICRP Publication 54, which were largely obtained from the biokinetic models of Publication 30. It is expected that bioassay analyses utilising the new excretion rate functions will lead to estimations of effective dose which can be substantially different from those that would be obtained from the existing ICRP Publication 54 excretion rates. It is expected that in future more emphasis will be placed on the use of biokinetic models for predicting excretion rates.

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METHODOLOGIC ASPECTS FOR SELECTING MOLECULES APPROPRIATE TO DECORPORATION TREATMENT

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ABSTRACT
Specification conditions are suggested here for guiding the work in order to obtain compounds appropriated to in vivo chelation of uranium. The first step will consist in speciation studies of the contaminant in the biological field to determine its biokinetics and the kind of interactions it makes with the biological components. In the second step, structural and modelisation studies will help to determine a potential chelate. The latest step is to ensure that the complex remains stable at physiological pH and to test the decorporating compound in vivo.

INTRODUCTION

Internal contamination with actinides induces radiological toxicity and the current approach to reduce the radiation dose resulting from accidentally-deposited radionuclides is to use a decorporation therapy in order to accelerate excretion and to reduce their binding to target organs.
At the present time, the therapy proposed after uranium contamination is still an infusion with 250 ml of sodium bicarbonate (1.4 %) (1). However, because of the incomplete effectiveness of bicarbonate treatment, the development of more effective substances remains an important aspect of radioprotection. Despite the interesting results already obtained, the decorporation therapy remains not completely satisfactory for some compounds like uranium or caesium, or uneffective for a compound such as neptunium. For these elements, studies are still in progress in order to find a molecule able to chelate them *in vivo* and to enhance their excretion out of the body. Concerning uranium, a lot of work has been done using different compounds like phosphonates or host molecules, but it was not really successful. Some interesting results have been obtained with 3,4,3-LIHOPO but present limitations for further work are due to the fact that this compound is hard to synthetize.
This work emphasizes uranium decorporation and it suggests a methodology for guiding the search for compounds appropriate for *in vivo* chelation.

SPECIATION STUDIES

The speciation studies will concern more specifically the interactions between the radionuclide and the biological components, at the cellular and blood levels and several studies will be necessary such as the *in vivo* speciation of the radionuclide in the main target organs (plasma, kidney and skeleton), and study of mechanisms which govern the uranium chelation.
On one hand, the determination of the chemical form of the complexes between uranium and blood components involves the development of specific analytic tools. The efficiency of capillary electrophoresis is currently tested to determine the chemical form of uranium at the blood level. On the other hand, the determination of the binding to biological ligands mechanisms in order to substitute a chelating agent will allow to optimise the choice of this agent as well as the therapeutic treatment. This work will mainly be done using cell culture models, in order to identify and localise the accepting sites of the radionuclide. Complementary *in vitro* studies devoted to determine the complexation kinetics will be performed.
The determination of the metabolism and pharmacokinetic of the complex is an important point in order to assess whether or not the chelating treatment do not increase the tissue concentration of the radionuclide in some organs other than the target ones. The injection of the complex will allow to determine its *in vivo* stability and biokinetics.

STRUCTURE OF POTENTIAL LIGANDS:

Chemical requirements
Several theoretical studies have already been conducted in order to propose a strategy allowing to find new interesting ligands for decorporation purpose. For some compounds like plutonium, the ratio of the ion charge versus the ionic radius is a value which has been considered to be important (2). Topologic analysis, taking into account the branching of the compounds (Wiener indice) have shown that this factor can also contribute to the decorporation properties (3).

The analysis of this different strategies allowed to propose some specifications adapted to uranium decorporation. According to the chemical properties of the uranyl ion some chemical requirements are of first importance: the diameter of the cavity which the ligand makes has to be sized between 4.7 and 5.2 Angström to complex the uranyl ion. Concerning the complexation type, uranyl ion forms planar equatorial hexacoordination with the ligating atoms. This means that it can form complexes particularly with one hexadentate ligand or two tridentate ligands. For statistical considerations, one hexadentate ligand is preferable considering that it has more chances to complexes with uranium in the body than two tridentates.
Due to its hard acid character, the uranyl ion forms complexes preferentially with hard bases. For that reason, the best binding groups are those containing oxygen.
The complexation constant between the uranyl ion and the ligand has to be high and the ligand has to be very uranium selective, compared to other metals present as traces in the body.

Biolological requirements
Some biological requirements have to be taken into account: after incorporation in the body, the uranyl ion is mainly excreted through the urinary way. The compound complex formed with the ligand has, for this reason, to be hydrosoluble. Moreover, the size of the compound must be appropriate for an easy elimination of the complex. The complex must remain stable at pH values ranging from 5 to 7; this will avoid the dissociation of the complex at blood pH and in the kidneys, leading to the binding of uranium to the membrane phospholipids of the of tubular cells which results in nephrotoxic effect. Moreover, the pK_a of the complexing groups have to be lower than 5 in order to allow the complexation at physiological pH 5 (including in the kidneys). Obviously, the compound has to be non toxic at the effective concentration.

COMPUTER SIMULATION
In addition to the chemical structure studies, we try to collate our experimental results with theoratical calculations using the HYPERCHEM (4) computer model. It could help to screen the potential ligands for uranium faster than physico-chemical studies do. HYPERCHEM allows to calculate the lowest energy levels between differents ligands and a specific metal. We are currently introducing in the model the specific values for uranium force fields, calculated from uranium carbonate and uranium nitrate.
For uranium carbonate, the total energy of the complex was found to be 610.6 $kcal.mol^{-1}$. This value includes different terms mainly due to electrostatic, van der Waals and covalent interactions. For uranium nitrate, the total energy was found to be 523.5 $kcal.mol^{-1}$. These first results have shown that the force field of the uranyl ion determined with the carbonates remains valid with the nitrates. Further tests are needed before using them to screen complexing compounds.

SYNTHESIS OF TRIPODE COMPOUNDS
All this considerations lead us to choose a family of tripode molecules. The theoretical complex which can be formed with the uranyl ion are shown Fig.1,2,3,4.

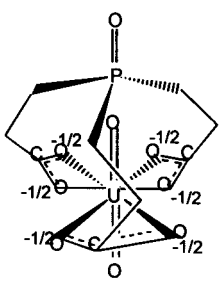

Fig.1: Complex 1/1 with phosphorylated tripode

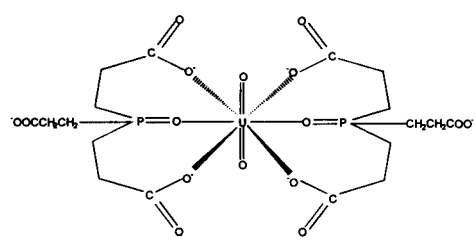

Fig.2: complex 1/2 with phosphorylated tripode

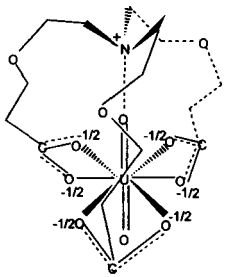

Fig.3 : Complex with nitrogenated tripode; G=COOH

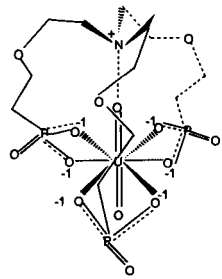

Fig.4: Complex with nitrogenated tripode; G=PO_3H_2

The tris(2-carboxy ethyl)phosphine oxide (R_3PO) has been synthetized (5). The stoechiometry of the complex R_3PO/UO_2^{2+} has been determined by conductimetry. The results confirmed that the formation of the complexes 1/1 and 2/1 is possible as shown in Fig.1 and 2.

The efficiency of this compound for uranium decorporation has been unsuccessfully tested *in vivo*. This lack of efficiency is probably due to the length of the molecule, which could be too short, and to the nature of the complexing groups, which are not completely ionisided at physiological pH.

We are currently synthetizing a new molecule, belonging to the same family, longer than R_3PO and with a pKa lower than 5, in order to allow the ionisation of the 3 groups in the blood and in the kidneys.

CONCLUSION

Methodologic aspects for selecting molecules appropriate for uranium decorporation treatment have been considered in this work. It includes several steps:

(1) speciation of uranium in the target organs: intra- or extracellular localisation, pH, affinity constants with biological ligands, etc., in order to provide basic information for the synthesis of new chelators,

(2) theoretical study of radionuclide and potential ligands structures,

(3) modelling studies consisting in theoretical and computer simulation studies,

(4) syntheses of new chelating compounds

(5) assessment of the physico-chemical properties of uranium/ligand complex: stability constant, pK, stability versus pH, hydrosolubility...

(6) first screening of the compound efficiency based on cell culture model,

(7) animal studies to optimize the removal of uranium.

The same process could be, later on, used for decorporation studies concerning other radionuclides.

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I.R.P.A. 9
VIENNA 14 - 19 April 1996

Poster N° P31-48

INTERPRETING FAECAL ANALYSIS RESULTS FOR MONITORING EXPOSURE TO URANIUM

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I INTRODUCTION

Radiotoxicological monitoring of workers exposed to non-transferable forms of uranium requires six-monthly examinations. These examinations are prescribed according to the kind of product manipulated and to the industrial risk attached to the workplace. The range of examinations that are useful for this kind of monitoring includes whole body counting examinations, urine analyses and in-line faecal sampling:

- ⇨ whole body examinations, which are fundamental to monitoring, provide a lung retention value. However, the detection limit of lung examinations is not low enough for chronic operational monitoring;
- ⇨ urine examinations are extremely sensitive to alpha activity (1mBq per isotope) but the fraction detected in the urine after incorporation by inhalation is very small;
- ⇨ in-line 24-hour faecal sampling allows avoiding any workplace exclusion;

Before 1989, faecal examinations were not prescribed systematically but only in case of incident. In 1989, after being informed of particular modifications in two workrooms, the occupational medical doctors decided to settle systematic faecal monitoring in addition to the existing radiotoxicological examinations (lung retention examinations and urine analyses). The authors intend to present their experience acquired over a six year period in the field of interpreting systematic faecal examinations after chronic inhalation of the different uranium compounds. They also present results of a study carried out to determine normal uranium concentrations in the faeces of a non-exposed population, the uranium content in drinking waters and the consequences on faecal excretion. Establishing the isotopic content of uranium in the faeces makes it possible to determine practical investigation levels for occupational monitoring.

Even if faecal sampling may be critically perceived by the personnel, the authors' experience highlights the value of this kind of analysis which allows to track down the industrial reality of the exposure. Internal dosimetry calculations cannot, however, be carried out, because the physical parameters of the inhaled aerosols are not always known.

II MATERIALS AND METHODS

The sampling of the faeces takes place over a period of 24 hours. It is prescribed without any prior exclusion. The notifications are sent out at the same time as the notifications for the lung retention examinations and the 24-hours urine analyses.

Measurement of uranium in the faeces : analytical method

The samples are reduced to ashes using a heating programme. They are then mineralised by an acid solution. The uranium is extracted and specifically eluted. Once the uranium present has been electrically deposited, alpha spectrometry is carried out.

Measurement of uranium in drinking water : analytical method

The samples come from the drinking fountains of the company restaurant and from bottles of water sold in the self-service cafeteria. The uranium present is extracted and specifically eluted. After electrodeposition, the eluate is analysed by alpha spectrometry. A weight measurement is also carried out on the eluate.

III RESULTS

Table 1: Uranium content in drinking water bottles sold by the company restaurant

Water samples	Uranium weight	Uranium activity	^{234}U	^{235}U	^{238}U	^{234}U / ^{238}U ratio	activity / weight ratio
	$\mu g.l^{-1}$	$mBq.l^{-1}$	$mBq.l^{-1}$	$mBq.l^{-1}$	$mBq.l^{-1}$		$mBq.\mu g^{-1}$
1	2.8	72	35	2	35	1	26
2	2.9	64	36	< 1	28	1.3	22
3	3.7	246	211	5	30	7	67
4	81	3800	2600	46	1154	2.3	47
5	0.3	15	11	< 1	4	2.8	50
6	0.5	12	8	< 1	4	2	24
7	0.1	2	1	< 1	1	1.0	20
8	< 0.1	< 1	< 1	< 1	< 1	om	om
9	4.9	115	54	15	46	1.2	24
10	1.4	49	30	< 1	19	1.6	35

om = out of matter

The results presented in table 1 show the importance of knowing the water intake (and, more generally, the food intake) of the workers being monitored. The variation of the isotopic composition is apparently due to the depth of the water capture source.

Table 2 : Yearly time series of faecal uranium content

Year	1989	1990	1991	1992	1993	1994	1995	general
Faecal content mBq/24h	350	325	942	325	300	218	200	390

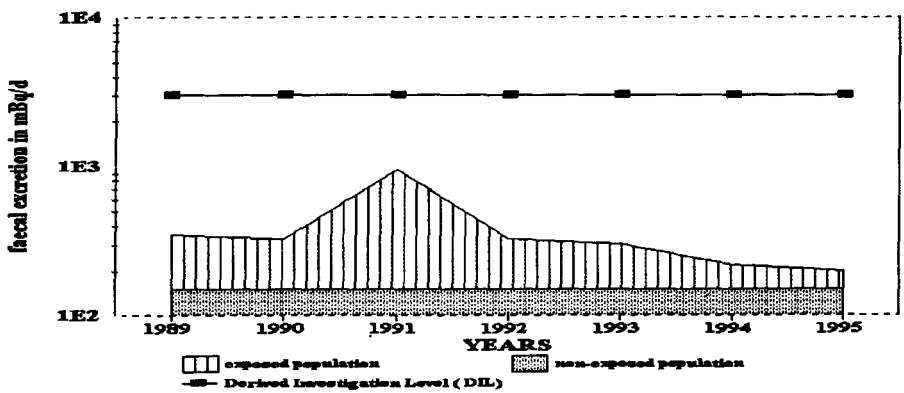

Figure 1 : faecal excretion variation as a function of the year

The table 1 and the figure 1 reveal an increase of total alpha activity in 1991. This evolution was sudden and significant. Since 1992, we observed a distinct decrease of the faecal uranium content. This is due to the information campaigns carried out by occupational medical doctors and the employers who were able to take corrective and preventive radioprotection actions in the workrooms and at individual workplaces.

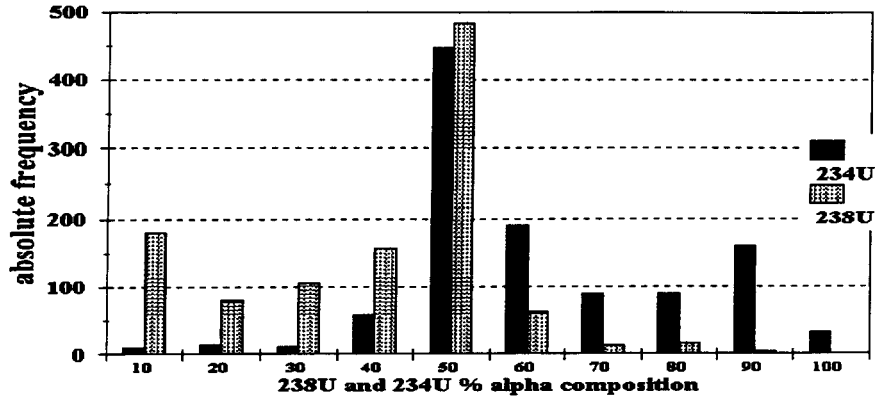

Figure 2 : Histogramms showing the values of faecal isotopic compositions

The figure 2 shows the various isotopic alpha compositions of the faecal samples. The most frequent alpha determinations are around the natural composition of uranium (50% ^{234}U and 50% ^{238}U). All the other isotopic compositions correspond to different workplaces. This informations indicate the origins of the intakes.

IV CONCLUSIONS AND PERSPECTIVES

The analyses prescribed after a contamination incident (urine and faecal measurements and whole body examinations) allow the importance of the incident to be quantified. They also make it possible to estimate the value of the incorporation according to the various International Commission for Radiological Prevention (ICRP) models.
The incorporation mode taken into account in systematic monitoring is chronic uranium inhalation. Given that the results of lung and urine analysis are always negative, the measurements of 24-hour non-exclusive faecal samples make it possible to track the evolution of an exposure to non-transferable compounds of uranium. The study carried out allows to determine normal values in the faeces for a non-exposed population on the one hand and the uranium content in the food supply, in particular in drinking water, on the other. In this way, it allows practical action levels to be established, leading to improved occupational monitoring.

The authors recommend the use of the values given below:

- ⇨ daily food intake value = 150 - 200 mBq
- ⇨ average value calculated for 1300 systematic examinations (over six years) = 390 mBq/24h
- ⇨ value higher than 1 Bq/24h: 24-hour faecal examination check

As well as the information concerning total 24-hour activity, the doctors have information on the isotopic composition of the uranium present in the faeces. This information means that the origin of the exposure can be ascertained so that specific tests may be ordered at the workplaces.
Because the whole body examinations and a 24-hour urine analysis were carried out at the same time as the faecal examinations, this study also makes it possible to satisfactorily quantify the occupational risk attached to a given workplace. However, it does not justify internal dosimetry calculations because the physical parameters of the inhaled aerosols (particularly, the granulometry) are not always known and because the sampling was done without preliminary exclusion.

The authors' experience highlights the usefulness of this type of radiotoxicological monitoring for occupational medical doctors who are able to track the radiological risk of personnel exposed to non-transferable uranium compounds in a manner that reflects industrial reality as closely as possible.

OPTIMIZATION OF MEASURING GEOMETRY OF THE SEIBERSDORF WHOLE-BODY COUNTER

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ABSTRACT

Whole-Body Counters are very useful for measuring incorporated radioactive isotopes that emit gamma rays. One of the main problems in whole body counting is the measuring geometry. The whole - body counter at the Research Center Seibersdorf is working in scan position and has two high purity germanium detectors. One of these detectors is located under the bed. Because of the constant vertical distance from the back of the measured object (human) and this detector, the geometry is the same for all measured humans. The second detector is located above the bed. The aim of the vertical position of this detector should be to keep the same distance from the chest for all humans. This distance is very important for the efficiency of the measuring system. We found that the best distance is 12 cm from the chest. If the chest thickness is known the obtimized distance can be fixed very quickly, because the upper detector can be adjusted vertically by a ruler. The main problem now is that different humans have different body sizes. For quick measurements we tried to calculate the vertical thickness of the chest (when the body lies on the bed) depending on weight and height of the measured person. We measured people at the Research Centre Seibersdorf and found a linear correlation between the chest thickness and the weight per height (kg/dm). With this new optimized geometry the efficiency calibration of the whole body counter was done.

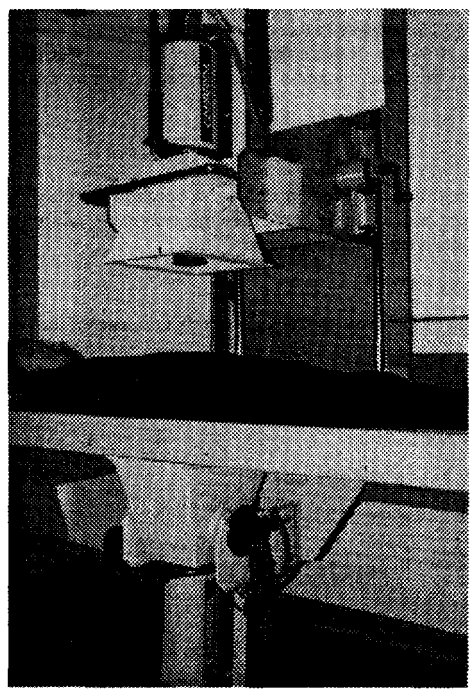

Fig. 1: HPGe-Whole-Body Counter Seibersdorf

INTRODUCTION

The Seibersdorf Whole-Body Counter (Fig. 1) has two coaxial hyper pure Germanium detectors (30% rel. efficiency each) in scan-geometry, i.e. that the person is lying on a bed and both detectors are moving from the head to the legs of the person. Because of the measuring room dimensions the maximum scanning length is 174 cm. One detector is located under the bed (p-type, Al-entrance window). Its distance to the bed is fixed but it can be changed indirectly by adjusting the height of the bed which is possible in 2 cm steps. We set the distance from the Al-entrance window to the upper bed surface to 7 cm. Generally, this distance is the same for all routine measurements. The upper detector (n-type, Be-entrance window) can be adjusted vertically. Its position depends on the chest thickness of the person to be measured. This position of the upper detector effects the efficiency very much. Both detectors are shielded with 2 cm to 4 cm thick lead. This shielding surrounds the whole detector crystal except towards the person. The open area has a dimension of 22 x 30 cm. The walls of the measuring room consist of 30 cm thick low activity concrete, 1 cm thick steel in the walls and ceiling and 7 cm thick steel in the floor. The entrance is built as a labyrinth.

For quantitative measurements with a Whole-Body Counter you need the efficiency (1,2) of the measurement. So it has to be always the aim to improve the efficiency of the system. Because of the $1/r^2$ - law and the attenuation in the body, the geometry of the object to be measured is very important for efficiency reasons. Also in wholebody counting this geometry (distance: body surface - detector) should be the same for all measured persons. But here we have the problem that the bodysizes (i.e. the body thickness) are not the same for different persons. In our case the distance from the lower detector to the upper bed surface (i.e. to the back of the measured person) is always the same. The upper detector can be changed in height. Its position should be arranged so that the pulse rates at this detector are the same as that of the lower detector in a wide range of energies. On the other hand physical damage of the person should be prevented. In our case there was a third point to be looked at. For a later calibration for localisation of pointsources in the body the count rates of the upper and lower detector should be equal when the pointsource is located in the middle of the chest. Because of the different attenuation (bed) and the different detector types the distance from the chest to the upper detector is not the same as from the lower detector to the back of the person. To get this optimized distance of the upper detector five measurements in different heights of the upper detector with the same phantom were done. We used 1 litre bottles with homogeneous distributed ^{166m}Ho solution. With these measurements we found an energy dependent position of the upper detector to get equal count rates. So we decided to use a distance of 12 cm because in this case we get equal count rates for middle energy region (about 800 keV) for both detectors. Now this distance (chest - upper detector) was fixed. For quick measurements the only value unknown now was the chest thickness when the person lies on the bed. That value we had to figure out.

MEASUREMENTS

In routine whole body counting we need to know the height and the weight of the person. We tried to calculate the chest thickness out of these data. We measured 41 adult people at the Research Centre Seibersdorf. The measurements were done as follows:

- people were lying on the bed
- the highest point of the chest was taken for measurements of the body thickness

Statistical data of measured people:

- age between 20 and 65 years
- height between 156 cm and 197 cm
- weight between 52 kg and 112 kg
- 38 male and 3 female

RESULTS

We found that the chest thickness varied between 21 cm and 30 cm. To get a correlation between the chest thickness and the height and weight, we used the weight per height (relative thickness in [kg/dm]) as parameter. The relationship is shown in Fig. 2. All datapoints are meanvalues of the weight per height at a given chest thickness. The error bars are the standard deviation. No error bar means, only one person was available. We derived the linear regression of the mean values. The values of the line variable are shown below.

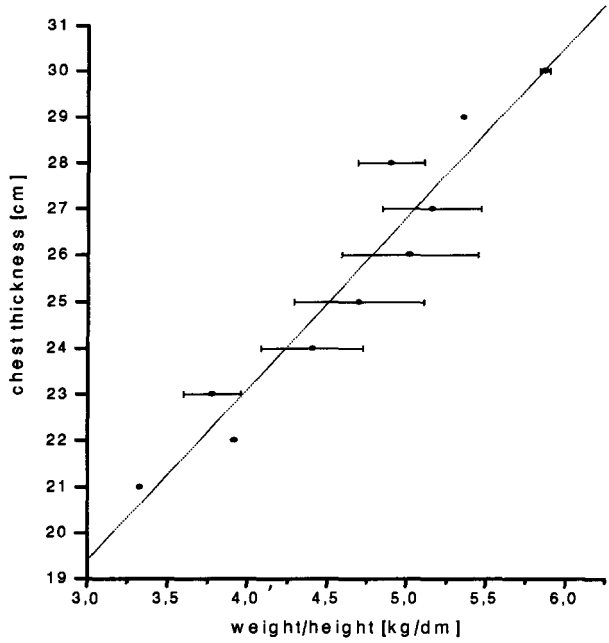

Fig. 2 Relationship between chest thickness and weight per height

Line data:

$$y = a + b \cdot x$$

a=8,2783

b=3,70758

y...chest thickness [cm] when the person lies on the bed

x...weight per height [kg/dm]

CONCLUSION

With this linear regression it is possible to calculate the chest thickness of a person with the personal data weight and height. The accuracy of the derived value is ±2 cm. It follows, that the uncertainty of the efficiency for homogeneous distributed activity in the body in the region of 800 keV is ±15% in worst case. With that uncertainty this method is useful for our routine whole body counting. For higher accuracy it will be necessary to measure the chest thickness of each person before scanning. We found no significant difference between male and female.

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IRPA9
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PAPER TITLE Assay Variations of Environmental Samples caused by Different Constructions of Standards

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MAJOR SCIENTIFIC TOPIC NUMBER 4.1. (see page 7)

ABSTRACT (See instructions overleaf)
When determining the quantity of a particular radionuclide contained in an environmental sample, errors arise due to differences in geometry and specific construction of the standard versus the sample. The difference is particularly acute when the isotope of concern is a low energy beta emitter.

Environmental samples are assayed against commercially available standards of various constructions including anodized, electroplated, polymeric membrane, and filter paper. In order to cover a large range of beta energies C-14, Tc-99, Cs-137, and Sr-90 are investigated. Differences in assay results are discussed in terms of beta spectra and average beta energies. Substantial discrepancies are demonstrated when when improper standards are used for low energy beta assays.

NEUTRON MEASUREMENTS BY IN-SITU GAMMA-RAY SPECTROMETRY USING CADMIUM CONVERTERS

Matjaž Korun, Rafael Martinčič, Bogdan Pucelj

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Abstract. Ambient fast neutron fluxes were measured with a neutron-gamma ray converter consisting of an aqueous solution of cadmium nitrate. The gamma rays were detected by a portable gamma-ray spectrometer. For fission neutrons with an asymptotic energy spectrum fair agreement between the flux density measured with the cadmium converter and converters containing boron was achieved. Also for the natural neutron background agreement between the measured and predicted flux density was established.

INTRODUCTION

The presence of neutrons is reflected in gamma-ray spectra in peaks belonging to neutron capture and inelastic scattering on materials in the vicinity of the sensitive volume of the detector. These peaks are usually present in low-level background measurements and contribute to the background [1]. In in-situ measurements neutron-induced peaks cannot be observed in the spectra because of the high background. However, the sensitivity of the measurements to neutron induced radiation can be enhanced by surrounding the sensitive volume of the detector by a neutron-gamma ray converter. Such converters are most effective at thermal neutron energies because of the high cross sections for neutron reactions. The best materials boron and cadmium [2]. ^{11}B has a high cross section for the n,α reaction. After α particle emission the residual nucleus is formed in its first excited state with a probability of 0.94 and emits deexcitation gamma rays at 477 keV which are Doppler broadened. In case of a cadmium absorber neutrons are captured by the isotope ^{113}Cd. If the converter is mixed with a moderating material, the measurements are also sensitive to fast neutrons. This offers the opportunity to measure thermal and fast neutron fluxes separately.

Ambient neutron fluxes were measured on-site in a nuclear power plant using boron converters [2]. Also the natural cosmic-ray induced neutron background at an altitude of 300 m was measured [3]. However, measurements to determine the natural background with a relative uncertainty of some tens of percent last approximately one day. To improve the sensitivity of the measurements, boron in the converter was exchanged by cadmium. From the point of view of sensitivity, boron exhibits several disadvantages as compared to cadmium:

- Its line is Doppler broadened to about 14 keV;
- The energy of the gamma ray, characteristic of absorption of neutrons in boron, is the same as that belonging to the decay of cosmic-ray produced ^{7}Be and
- The Compton edge of gamma rays belonging to the decay of ^{137}Cs induces an uneven background at the energy of 477 keV.

All these effects contribute to the uncertainty of the peak area and the sensitivity. Therefore, cadmium converters are more suitable for measurements of small neutron fluxes. The line at 558 keV from the capture of neutrons on ^{113}Cd is not interfered by lines from naturally occurring radionuclei and is not Doppler broadened.

METHODS AND MEASUREMENTS

The measurements were performed with a 25% semiconductor detector and a cadmium converter with a volume of approximately 2 litres. The converter thickness was 5 cm at the front of the detector and 4 cm at the side of the detector. The converter was prepared as an aqueous solution of cadmium nitrate. The amount of cadmium dissolved in the converter is about 255 g. The count rate in the 558 keV line is given by:

$$n(558) = b(558)\,[\eta_T(558)\,\Phi_T + \eta_F(558)\,P_F\,\Phi_F] \quad .$$

Here $b(558)$ denotes the probability for the emission of gamma rays of energy 558 keV which is 0.73, $\eta_T(558)$ the probability for the detection of gamma rays with the energy of 558 keV from capture of thermal neutrons, $\eta_F(558)$ the probability for the detection of gamma rays from capture of fast neutrons, P_F the probability for thermalization of fast neutrons and Φ and Φ the fluxes of thermal and fast neutrons through the converter surface. Since the average free path of thermal neutrons in the converter is about 1.7 cm, thermal neutrons entering the converter through the surface are absorbed far from the detector sensitive area. Therefore, the probability for the registration of the corresponding gamma rays in the detector is small. On the other hand, since the diffusion length of fast neutrons during moderation is larger than the thickness of the converter [4], the source of gamma rays from fast neutrons is homogeneous. Neglecting the contribution of thermal neutrons, the flux density of fast neutrons is expressed as:

$$\Phi_F = \frac{n(558)}{b(558)\,\eta(558)\,P_F S} \quad ,$$

where S denotes the cross section of the converter.

The response of the converter-spectrometer system to neutrons was determined from the response of the same spectrometer to neutrons using two borated paraffin converters. The first had the same geometry as the cadmium converter, and for the second the thermalization probability was calculated with the ANISN code for asymptotic fission neutron spectrum [2]. The ratio of count rates in the 477 keV line in the measurement with the two converters is:

$$\frac{n_1(477)}{n_2(477)} = \frac{\eta_1(477)}{\eta_2(477)}\,\frac{P_{F1}}{P_{F2}}\,\frac{S_1}{S_2} \quad .$$

Since $S=S_1$ and assuming $P_F=P_{F1}$, because the converters are of equal dimensions, the flux density can be expressed as:

$$\Phi_F = \frac{n(558)}{b(558)\;\eta_2(447)\,\dfrac{\eta(558)}{\eta_1(477)}\,\dfrac{n_1(477)}{n_2(477)}\,P_{F2}\,S_2} \quad .$$

The efficiency of the second converter $\eta_2(477)$ was calculated from the measurement of an internal soil standard in the same geometry [2] and found to be 1.70 (1 ± 0.03). The efficiencies $\eta(558)$ and $\eta_1(477)$ refer to the same geometry. Their ratio was determined from the measurement of another internal standard in the geometry of the cadmium converter to be 0.88 (1 ± 0.02). This value was used for the calculation of the neutron flux density, and therefore the differences in the self-absorption of the cadmium converter, the borated paraffin converter and the internal standard were neglected. Near the reactor building the ratio of the count rates in the 477 keV line from the two borated paraffin converters was measured as 1.20 (1 ± 0.05). The probability for the thermalization of fast neutrons, calculated by the ANISN code, and the cross section of the corresponding converter are 0.42 (1 ± 0.02) and 104 cm^2 respectively [2]. Finally, the relation between the count rate in the 558 keV line and the flux density of fast neutrons is:

$$\Phi_F = \frac{n(558)}{0.57(1 \pm 0.08)\;cm^2} \quad .$$

To test the response of the spectrometer to small neutron fluxes measurements were performed in the vicinity of a nuclear power reactor at two distances from the reactor building. The measurements lasted approximately 50 and 500 minutes respectively. The flux densities of fast neutrons at the two locations were found to be 0.068 (1 ± 0.17) cm^{-2} s^{-1} and 0.006 (1 ± 0.5) cm^{-2} s^{-1}, respectively.

To test the reliability of the neutron flux measurements with the cadmium converter a measurement was performed at a location where the thermal and fast neutron fluxes had also been measured using boron converters [2]. The measurements gave 0.13 (1 ± 0.08) cm^{-2} s^{-1} and 0.86 (1 ± 0.06) cm^{-2} s^{-1} respectively. The measurement with the cadmium converter gave a slightly lower value of 0.76 (1 ± 0.09) cm^{-2} s^{-1} for the flux of fast neutrons.

DISCUSSION

It should be noted that in an asymptotic fission spectrum, as well as in the spectrum of naturally occurring neutrons, the flux density of thermal neutrons is much smaller than the flux density of fast neutrons [2,5]. Only in this case can the contribution of thermal neutrons to the 558 keV line be neglected. The results show fair agreement between the fast neutron flux density measured by boron converters, where the thermal neutron flux is accounted for, and the flux density measured with the cadmium converter.

The time required to measure the natural neutron background flux is about ten hours. This time may be shortened by using a larger semiconductor detector and by shielding the spectrometer against background gamma-ray radiation.

CONCLUSION

It was shown that a high-resolution gamma-ray spectrometer can be converted to a neutron flux meter by using a suitable cadmium converter. A converter containing neutron moderating and absorbing materials, in the present case water and cadmium, is sensitive to fast neutrons. It should be noted that for a quantitative relation between the neutron flux and the count rate in the 558 keV line the neutron spectrum must be supposed. The agreement between the flux measured with the cadmium and boron converters near the reactor building, as well as the agreement between the natural neutron background and the flux measured further away reflects the fact that the spectra of fast neutrons far from a fission source and of natural neutrons are similar [2,6].

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ON THE DISTINCTION OF SMALL EXTRA DOSES FROM THE NATURAL BACKGROUND

DESIGN AND TESTING OF A SENSITIVE TL-DOSEMETER

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INTRODUCTION

An important issue in environmental dosimetry is the detection of artificial doses added to the environmental dose by installations, working with ionising radiation (e.g. hospitals, reactors, in this article called a 'facility'). Dutch legislation sets a limit for the extra radiation dose rate added to the environment by a facility to a maximum of 40 μSv/a (1). Straightforward environmental dosimetry isn't able to distinguish such a small artificial dose from the fluctuating natural background within a reasonable measuring time.
A special method and dosimetry system, SEAD (Sensitive Extra Ambient Dosemeter) (2) which takes into account fluctuations in natural background and is sensitive enough to detect an extra dose rate of 40 μSv/a is presented in this study.

FLUCTUATIONS IN NATURAL BACKGROUND

Natural background radiation is build of three components: cosmic radiation, terrestrial radiation and radiation due to airborne and deposited radioactivity (radon and progeny). Terrestrial radiation varies in the Netherlands from 105 μSv/a to 683 μSv/a (3). Temporal fluctuations in airborne radioactivity and cosmic radiation occur on an daily basis and depend on rainfall, temperature, air-pressure and windvelocity; short risings due to rainfall can be up to 80 nSv/h compared to an average of 70 nSv/h. Typical temporal fluctuations are shown in figure 1. Due to temproal fluctuations weekly ambient doses can differ more than 2 μSv. Translated to a dose per year, this gives a difference of 100 μSv.

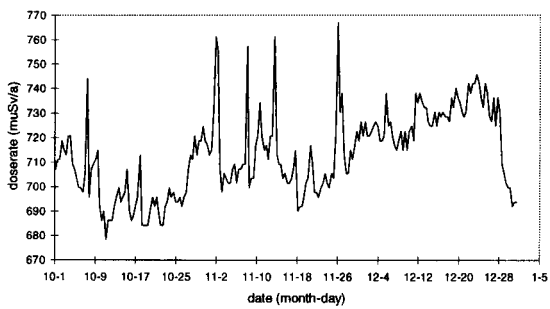

Figure 1: Fluctuations of the natural background based on 10-hour measurement periods at Delft. Dose rates are converted into yearly doses.

CUSTOMARY METHODS TO DISTINGUISH ARTIFICIAL DOSES FROM NATURAL BACKGROUND

There are a few different methods in use to detect an artificial dose added to the environment by a facility, all with the purpose to estimate the natural background and subtract this estimation from the measured dose rate around the facility, leaving an estimation of the artificial added dose rate.

The first way to estimate the natural dose rate is by measuring the dose rate at a place where there's no influence of the facility on the background radiation. Because the terrestrial component of the natural background differs strongly form place to place, this method is unlikely to produce an unbiased estimation of the 'natural background around the facility'. The second method is only usable in the case of facilities where the source(s) of ionising radiation can be switched off. In this case the background can be estimated by a measurement at a time the facility doesn't contribute to the natural background. Figure 1 shows clearly that it's rather unlikely that also this method provides an unbiased estimation within a practical measuring period (2 weeks). A reasonable improvement, but only for facilities where the sources can be switched off, can be made by combining these two methods. Two different measurements at the facility can be corrected for temporal fluctuations by measurements on a reference location (outside the influence of the facility). A last method estimates the fluctuations in natural background by mathematical modelling. This estimation needs a lot of precise meteorological data on air-pressure, rainfall, temperature at a high spatial resolution and is particularly difficult in the case of rainfall (wash-out of radon-progeny).

A NEW METHOD FOR MEASURING THE EXTRA ARTIFICIAL DOSE

Considering the disadvantages of the customary methods a different method is introduced. This method doesn't give an estimation of the natural background but uses the angle dependence of the incoming radiation and is based on the application of a radiation shield. By placing two detectors on both sides of a *radiation-shield*, one side faced to the facility and one to the opposite side, it's possible to measure a natural component (*not* natural background) and a natural component + artificial component in the same time and on the same place. In this way temporal and spatial fluctuations are taken into account and the difference between the two detectors provides a unbiased estimation of the extra dose added to the environment by the facility

DETECTOR DESCRIPTION

The detector SEAD (figure 2) is composed of two dosemeters of ten TL- samples each. The TL-material is LiF: Mg, Cu, P (GR200A), which is very sensitive to photons and has some other outstanding properties for ambient dosimetry (like low-fading-rate and low hygroscopicity). Using ten samples in each dosemeter improves the statistics of the experiment and thus lowers the lowest artificial dose which can be discriminated from the dose due to natural background. The samples are shielded with 5 mm PTFE .The shielding materials consist of 10 cm thick lead, which is enough to shield 6 MeV gamma-radiation for 98,7%. The lead is specially selected for its low specific activity (old) and is covered with 4 mm Copper to provide for as less signal as possible due to the lead-shielding. The cover of the detector is made of 1,5 mm Aluminium which transmits more than 80% of 30 keV photon-radiation and isn't a impediment for higher energies. The cover protects the dosemeters against extreme weather conditions.

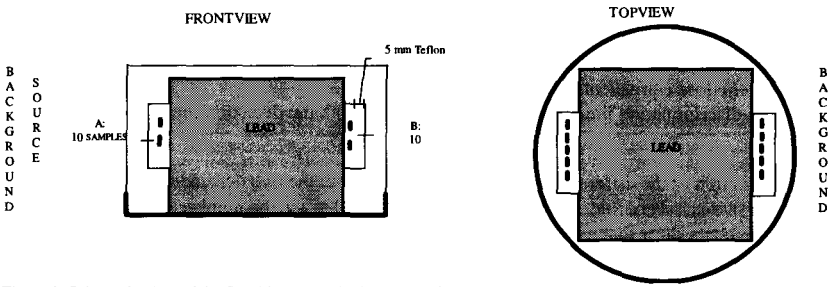

Figure 2: Schematic view of the Sensitive Extra Ambient Dosedetector

PERFORMANCE

Annealing and read-out The TL-material GR200A has an excellent performance, but only when specific and highly reproducable annealing and read-out procedures are applied. It's shown that an annealing temperature 240 °C for 10 minutes followed by fast cooling (>150 K/min) gives a good sensitivity. The *read-out procedure* consists of a plateau on 150°C for 5 seconds, followed by a heating-rate of 3 K/s to 240°C and a second plateau at 240 °C for 13 seconds. The integrating limits for TL-light are from 155°C to *the end* of the second platform. In this way the, non-stable peak 2 (peak temp ≈120 °C) is left out of the integration and peak 4 (peak temp 220 °C) is included as much as possible.

Uniformity and reproducibility To gain a batch of GR200A-samples with a great uniformity a selection of these 40 samples is made out of 100 'new' samples with a manufacturer-guarantee of max. 5% deviation. The standard deviation of the readings of 40 samples after equal irradiation with 10 mR is 0.8%. The standard deviation of 6 readings of each sample after 6 equal irradiations with 60 μGy is less than 2%.

Fading Three groups of 4 samples were placed outdoors for 14 days. One group was irradiated with 3,65 mR before the storage and one group was irradiated after the 14 days with 4,35 mR. After accounting for the different doses and the environmental dose in 14 days (measured with the third group) a fading of (2 ± 4) % (1 st.dev. of the mean) was calculated.

Zero-dose. The zero-dose of all samples was measured by reading them directly after annealing. Most of the readings of the zero-dose didn't exceed the minimum detectable value (determined by the variance in reader-background). The samples which did have a detectable zero-dose were left out of the selection.

Length of field-cycle. The length of the field cycle was determined by irradiating a group of 6 samples which different (low) doses. It was proven that the relative standard-deviation of the mean of 6 readings remained constant when the samples were exposed to 4 mR or more. This can be explained by writing the relative standard deviation σ_N as a function of the total counts n and standard deviation of the reader-background σ_B.(4)

$$\frac{\langle \sigma_N^2 \rangle}{N^2} = b + \frac{1}{n} + 2\frac{\sigma^2_B}{n^2} \qquad (1)$$

where b is a constant. In the case of GR200A and the used reader (modified Harshaw 2000) b>0.001, σ_B=10 and n > 4000 counts for a dose of 4 mR, leaving b, a constant, as the only significant contribution to the relative standard deviation. A dose of 4 mR is equal to the environmental background in 14 days. Fig. 4 shows the relative standard deviation of a TL-measurement as a function of the measurement time in a natural background.

Hygroscopicity. After 14 days storage the difference of the mean between 5 samples placed in a closed can, filled with water and five samples placed in a dry environment was non-significant (0 ± 2)%.

Energy response The relation of measured counts and irradiated dose (mR) is plotted in fig. 3. The defined dependence between ambient dose equivalent (cSv) and exposure (R) as given by ICRU (5) is plotted in fig. 3.

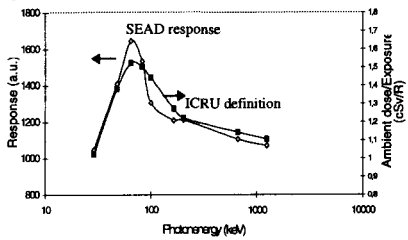

fig 3: TL-response of SEAD and ICRU definition

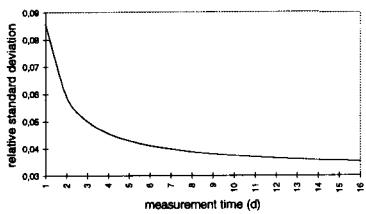

fig 4: Relative standard deviation as a function of measuring time

From fig 3 it's clear that the combinaton of lead, 1,5 mm Al and the 5 mm PTFE gives an excellent agreement with the ICRU definition of ambient dose equivalent.

Discrimination limit The discrimination limit, O_D, (not to be confused with the detection limit) is defined by

$$O_D = t_{\alpha/2}(n+m-2)S_d \qquad (2)$$

Where S_d is the standard deviation of the difference of the mean readings from the two dosemeters and $t_{\alpha/2}$ is the Student-t critical value for a two-sided confindence interval of 1-α and n and m are the number of samples in dosemeter A and B. Measurements show that n=m=10 is sufficient to reach the desired discrimination limit.

Field measurements Results of 4 measurements with SEAD are shown in table 1, two of them at one place at the site boundary of a research reactor and three at different places at the site boundary of a hospital in which radiation facilities are operated.

Table 1: Resluts of measurements with SEAD.

	Reactor	Reactor	Hospital 1	Hospital 2
Extra artificial dose (mSv/a)	0.032	0.037	-0.017	-0.032
'Background' (mSv/a)	0.73	0.68	0.86	0.87
Discrimination limit (95%) (mSv/a)	0.030	0.021	0.033	0.029

DISCUSSION AND CONCLUSION

SEAD is capable of distinguishing an artificial dose rate of 40 μSv/a from the natural background in a measuring time of 14 days. The performance tests show that SEAD shows excellent stability, in particular because of the use of (well-selected) GR200A-samples. Table 1 shows, however, that in absence of any artificial radiation source a difference in natural dose rate can be up to 32 μSv/a (hospital 2). This difference isn't a result of diffence in soil-type (both dosemeters are virtually on the same spot) but is due to diffences in surrounding elements (buildings, trees etc.) on both sides of the detector. This difference can only be established precisely when measurements around a facility can be performed with 'the source switched off'. In other cases the artificial dose rate as established by SEAD must be considered as best estimate, because it eliminates by far more biasing factors than methods used up to now.

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PLASTIC SCINTILLATOR WATER MONITOR RESPONSE TO ^{32}P

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INTRODUCTION

Water monitors equipped with a plastic scintillator are widely used to measure the radioactive concentrations of β-emitters in liquid waste. Plastic monitors are highly sensitive to high energy β-emitters like ^{32}P, but they are insensitive to low energy β-emitters like ^{3}H, ^{14}C. This is a result of the detector's surface being covered by a thin protective layer. A computerized plastic scintillator water monitor system was devised for high energy β-emitters at our facility. The response to ^{32}P was investigated so that optimum conditions for operating the monitor could be achieved.

MATERIALS AND METHODS

Fig.1 shows a schematic diagram of the sistem that is composed of a personal computer, plastic scintillator detector, controler and pipe lines. A computerized sequence for operating the monitor, background (BG) measurements, waste liquid sampling, sample measurements, and detector cleaning, was programmed. Signals of the detector were fed to a multichannel scaler(MCS) to analyze the time variations of the count rate.

^{32}P solution of 0.04-0.4Bq $\cdot$ cm^{-3} prepared in a radioactive liquid waste storage tank of100m^3 was used to measure the detector response, which was also analyzed by using a short cut circuit connected to a 20 ℓ polyethylene tank containing ^{32}P solution of 5Bq $\cdot$ cm^{-3}. The activity in the water sampled near the detector was measured by a liquid scintillation counter(LSC), and was compared with those values measurd by the water monitor, MCS, and a multichannel analyzer(MCA) . This was done to confirm and/or adjust the different methods giving the same values as LSC.

RESULTS AND DISCUSSION

Figure 2 represents an example of time variation of count rates measured at a dwell time of10sec/Channel. by MCS. The counting efficiency was 5.1 $\pm$0.1%. The time to replace BG water with waste

liquid was about 6 min. and a cleaning time was about 1min. This difference is due to the the dissimilarity of both pipe line lengths. Some measurements showed that the count rates after cleaning was a little higher than that of BG, because of the ^{32}P in the residual waste liquid in the pipe near the detector after the first cleaning. This means that contamination occasionally remains on the detector's surface after cleaning contingent on the pipe arrangement. The detector was dismantled and the surface was completely decontaminated by wiping it with soft tissue paper. Second cleaning also completely washed out any residual waste liquid. The detector was then free of any contamination.

Replace times, cleaning times, and repeat times should be experimentaly confirmed for each monitoring system, since each system differs in pipe line arrangement as well as length.

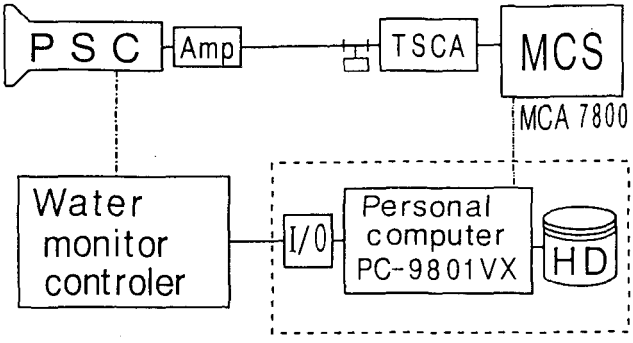

Fig.1. A schematic diagram of the computerized water monitor equipped with a plastic scintillator.
PSC; Plastic scintillator detector, MCS; Multichannel scaler
HD; Hard disk

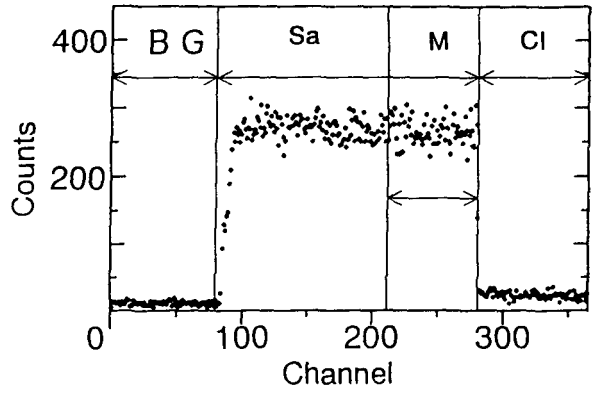

Fig.2. The time variation of the count rate of each step measured by MCS.
Dwell time; 10sec/Channel, BG; Background, Sa; Sampling, M; Measurement, Cl; Cleaning

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Environmental samples are assayed against commercially available standards of various constructions including anodized, electroplated, polymeric membrane, and filter paper. In order to cover a large range of beta energies C-14, Tc-99, Cs-137, and Sr-90 are investigated. Differences in assay results are discussed in terms of beta spectra and average beta energies. Substantial discrepancies are demonstrated when when improper standards are used for low energy beta assays.

NEUTRON MEASUREMENTS BY IN-SITU GAMMA-RAY SPECTROMETRY USING CADMIUM CONVERTERS

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Abstract. Ambient fast neutron fluxes were measured with a neutron-gamma ray converter consisting of an aqueous solution of cadmium nitrate. The gamma rays were detected by a portable gamma-ray spectrometer. For fission neutrons with an asymptotic energy spectrum fair agreement between the flux density measured with the cadmium converter and converters containing boron was achieved. Also for the natural neutron background agreement between the measured and predicted flux density was established.

INTRODUCTION

The presence of neutrons is reflected in gamma-ray spectra in peaks belonging to neutron capture and inelastic scattering on materials in the vicinity of the sensitive volume of the detector. These peaks are usually present in low-level background measurements and contribute to the background [1]. In in-situ measurements neutron-induced peaks cannot be observed in the spectra because of the high background. However, the sensitivity of the measurements to neutron induced radiation can be enhanced by surrounding the sensitive volume of the detector by a neutron-gamma ray converter. Such converters are most effective at thermal neutron energies because of the high cross sections for neutron reactions. The best materials boron and cadmium [2]. ^{11}B has a high cross section for the n,α reaction. After α particle emission the residual nucleus is formed in its first excited state with a probability of 0.94 and emits deexcitation gamma rays at 477 keV which are Doppler broadened. In case of a cadmium absorber neutrons are captured by the isotope ^{113}Cd. If the converter is mixed with a moderating material, the measurements are also sensitive to fast neutrons. This offers the opportunity to measure thermal and fast neutron fluxes separately.

Ambient neutron fluxes were measured on-site in a nuclear power plant using boron converters [2]. Also the natural cosmic-ray induced neutron background at an altitude of 300 m was measured [3]. However, measurements to determine the natural background with a relative uncertainty of some tens of percent last approximately one day. To improve the sensitivity of the measurements, boron in the converter was exchanged by cadmium. From the point of view of sensitivity, boron exhibits several disadvantages as compared to cadmium:

- Its line is Doppler broadened to about 14 keV;
- The energy of the gamma ray, characteristic of absorption of neutrons in boron, is the same as that belonging to the decay of cosmic-ray produced ^{7}Be and
- The Compton edge of gamma rays belonging to the decay of ^{137}Cs induces an uneven background at the energy of 477 keV.

All these effects contribute to the uncertainty of the peak area and the sensitivity. Therefore, cadmium converters are more suitable for measurements of small neutron fluxes. The line at 558 keV from the capture of neutrons on ^{113}Cd is not interfered by lines from naturally occurring radionuclei and is not Doppler broadened.

METHODS AND MEASUREMENTS

The measurements were performed with a 25% semiconductor detector and a cadmium converter with a volume of approximately 2 litres. The converter thickness was 5 cm at the front of the detector and 4 cm at the side of the detector. The converter was prepared as an aqueous solution of cadmium nitrate. The amount of cadmium dissolved in the converter is about 255 g. The count rate in the 558 keV line is given by:

$$n(558) = b(558)\,[\eta_T(558)\,\Phi_T + \eta_F(558)\,P_F\,\Phi_F] \quad .$$

Here $b(558)$ denotes the probability for the emission of gamma rays of energy 558 keV which is 0.73, $\eta_T(558)$ the probability for the detection of gamma rays with the energy of 558 keV from capture of thermal neutrons, $\eta_F(558)$ the probability for the detection of gamma rays from capture of fast neutrons, P_F the probability for thermalization of fast neutrons and Φ and Φ the fluxes of thermal and fast neutrons through the converter surface. Since the average free path of thermal neutrons in the converter is about 1.7 cm, thermal neutrons entering the converter through the surface are absorbed far from the detector sensitive area. Therefore, the probability for the registration of the corresponding gamma rays in the detector is small. On the other hand, since the diffusion length of fast neutrons during moderation is larger than the thickness of the converter [4], the source of gamma rays from fast neutrons is homogeneous. Neglecting the contribution of thermal neutrons, the flux density of fast neutrons is expressed as:

$$\Phi_F = \frac{n(558)}{b(558)\,\eta(558)\,P_F S} \quad ,$$

where S denotes the cross section of the converter.

The response of the converter-spectrometer system to neutrons was determined from the response of the same spectrometer to neutrons using two borated paraffin converters. The first had the same geometry as the cadmium converter, and for the second the thermalization probability was calculated with the ANISN code for asymptotic fission neutron spectrum [2]. The ratio of count rates in the 477 keV line in the measurement with the two converters is:

$$\frac{n_1(477)}{n_2(477)} = \frac{\eta_1(477)}{\eta_2(477)}\,\frac{P_{F1}}{P_{F2}}\,\frac{S_1}{S_2} \quad .$$

Since $S=S_1$ and assuming $P_F=P_{F1}$, because the converters are of equal dimensions, the flux density can be expressed as:

$$\Phi_F = \frac{n(558)}{b(558)\;\eta_2(447)\,\dfrac{\eta(558)}{\eta_1(477)}\,\dfrac{n_1(477)}{n_2(477)}\,P_{F2}\,S_2} \quad .$$

The efficiency of the second converter $\eta_2(477)$ was calculated from the measurement of an internal soil standard in the same geometry [2] and found to be 1.70 (1 ± 0.03). The efficiencies $\eta(558)$ and $\eta_1(477)$ refer to the same geometry. Their ratio was determined from the measurement of another internal standard in the geometry of the cadmium converter to be 0.88 (1 ± 0.02). This value was used for the calculation of the neutron flux density, and therefore the differences in the self-absorption of the cadmium converter, the borated paraffin converter and the internal standard were neglected. Near the reactor building the ratio of the count rates in the 477 keV line from the two borated paraffin converters was measured as 1.20 (1 ± 0.05). The probability for the thermalization of fast neutrons, calculated by the ANISN code, and the cross section of the corresponding converter are 0.42 (1 ± 0.02) and 104 cm^2 respectively [2]. Finally, the relation between the count rate in the 558 keV line and the flux density of fast neutrons is:

$$\Phi_F = \frac{n(558)}{0.57(1 \pm 0.08)\;cm^2} \quad .$$

To test the response of the spectrometer to small neutron fluxes measurements were performed in the vicinity of a nuclear power reactor at two distances from the reactor building. The measurements lasted approximately 50 and 500 minutes respectively. The flux densities of fast neutrons at the two locations were found to be 0.068 (1 ± 0.17) cm^{-2} s^{-1} and 0.006 (1 ± 0.5) cm^{-2} s^{-1}, respectively.

To test the reliability of the neutron flux measurements with the cadmium converter a measurement was performed at a location where the thermal and fast neutron fluxes had also been measured using boron converters [2]. The measurements gave 0.13 (1 ± 0.08) cm^{-2} s^{-1} and 0.86 (1 ± 0.06) cm^{-2} s^{-1} respectively. The measurement with the cadmium converter gave a slightly lower value of 0.76 (1 ± 0.09) cm^{-2} s^{-1} for the flux of fast neutrons.

DISCUSSION

It should be noted that in an asymptotic fission spectrum, as well as in the spectrum of naturally occurring neutrons, the flux density of thermal neutrons is much smaller than the flux density of fast neutrons [2,5]. Only in this case can the contribution of thermal neutrons to the 558 keV line be neglected. The results show fair agreement between the fast neutron flux density measured by boron converters, where the thermal neutron flux is accounted for, and the flux density measured with the cadmium converter.

The time required to measure the natural neutron background flux is about ten hours. This time may be shortened by using a larger semiconductor detector and by shielding the spectrometer against background gamma-ray radiation.

CONCLUSION

It was shown that a high-resolution gamma-ray spectrometer can be converted to a neutron flux meter by using a suitable cadmium converter. A converter containing neutron moderating and absorbing materials, in the present case water and cadmium, is sensitive to fast neutrons. It should be noted that for a quantitative relation between the neutron flux and the count rate in the 558 keV line the neutron spectrum must be supposed. The agreement between the flux measured with the cadmium and boron converters near the reactor building, as well as the agreement between the natural neutron background and the flux measured further away reflects the fact that the spectra of fast neutrons far from a fission source and of natural neutrons are similar [2,6].

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ON THE DISTINCTION OF SMALL EXTRA DOSES FROM THE NATURAL BACKGROUND

DESIGN AND TESTING OF A SENSITIVE TL-DOSEMETER

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INTRODUCTION

An important issue in environmental dosimetry is the detection of artificial doses added to the environmental dose by installations, working with ionising radiation (e.g. hospitals, reactors, in this article called a 'facility'). Dutch legislation sets a limit for the extra radiation dose rate added to the environment by a facility to a maximum of 40 μSv/a (1). Straightforward environmental dosimetry isn't able to distinguish such a small artificial dose from the fluctuating natural background within a reasonable measuring time.
A special method and dosimetry system, SEAD (Sensitive Extra Ambient Dosemeter) (2) which takes into account fluctuations in natural background and is sensitive enough to detect an extra dose rate of 40 μSv/a is presented in this study.

FLUCTUATIONS IN NATURAL BACKGROUND

Natural background radiation is build of three components: cosmic radiation, terrestrial radiation and radiation due to airborne and deposited radioactivity (radon and progeny). Terrestrial radiation varies in the Netherlands from 105 μSv/a to 683 μSv/a (3). Temporal fluctuations in airborne radioactivity and cosmic radiation occur on an daily basis and depend on rainfall, temperature, air-pressure and windvelocity; short risings due to rainfall can be up to 80 nSv/h compared to an average of 70 nSv/h. Typical temporal fluctuations are shown in figure 1. Due to temproal fluctuations weekly ambient doses can differ more than 2 μSv. Translated to a dose per year, this gives a difference of 100 μSv.

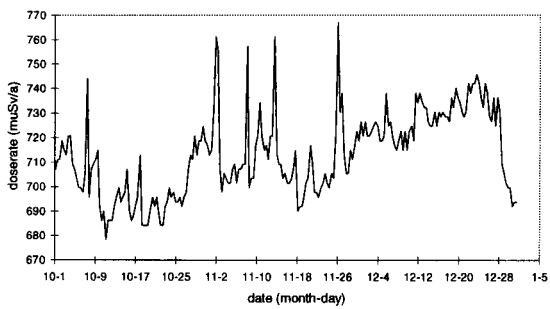

Figure 1: Fluctuations of the natural background based on 10-hour measurement periods at Delft. Dose rates are converted into yearly doses.

CUSTOMARY METHODS TO DISTINGUISH ARTIFICIAL DOSES FROM NATURAL BACKGROUND

There are a few different methods in use to detect an artificial dose added to the environment by a facility, all with the purpose to estimate the natural background and subtract this estimation from the measured dose rate around the facility, leaving an estimation of the artificial added dose rate.

The first way to estimate the natural dose rate is by measuring the dose rate at a place where there's no influence of the facility on the background radiation. Because the terrestrial component of the natural background differs strongly form place to place, this method is unlikely to produce an unbiased estimation of the 'natural background around the facility'. The second method is only usable in the case of facilities where the source(s) of ionising radiation can be switched off. In this case the background can be estimated by a measurement at a time the facility doesn't contribute to the natural background. Figure 1 shows clearly that it's rather unlikely that also this method provides an unbiased estimation within a practical measuring period (2 weeks). A reasonable improvement, but only for facilities where the sources can be switched off, can be made by combining these two methods. Two different measurements at the facility can be corrected for temporal fluctuations by measurements on a reference location (outside the influence of the facility). A last method estimates the fluctuations in natural background by mathematical modelling. This estimation needs a lot of precise meteorological data on air-pressure, rainfall, temperature at a high spatial resolution and is particularly difficult in the case of rainfall (wash-out of radon-progeny).

A NEW METHOD FOR MEASURING THE EXTRA ARTIFICIAL DOSE

Considering the disadvantages of the customary methods a different method is introduced. This method doesn't give an estimation of the natural background but uses the angle dependence of the incoming radiation and is based on the application of a radiation shield. By placing two detectors on both sides of a *radiation-shield*, one side faced to the facility and one to the opposite side, it's possible to measure a natural component (*not* natural background) and a natural component + artificial component in the same time and on the same place. In this way temporal and spatial fluctuations are taken into account and the difference between the two detectors provides a unbiased estimation of the extra dose added to the environment by the facility

DETECTOR DESCRIPTION

The detector SEAD (figure 2) is composed of two dosemeters of ten TL- samples each. The TL-material is LiF: Mg, Cu, P (GR200A), which is very sensitive to photons and has some other outstanding properties for ambient dosimetry (like low-fading-rate and low hygroscopicity). Using ten samples in each dosemeter improves the statistics of the experiment and thus lowers the lowest artificial dose which can be discriminated from the dose due to natural background. The samples are shielded with 5 mm PTFE .The shielding materials consist of 10 cm thick lead, which is enough to shield 6 MeV gamma-radiation for 98,7%. The lead is specially selected for its low specific activity (old) and is covered with 4 mm Copper to provide for as less signal as possible due to the lead-shielding. The cover of the detector is made of 1,5 mm Aluminium which transmits more than 80% of 30 keV photon-radiation and isn't a impediment for higher energies. The cover protects the dosemeters against extreme weather conditions.

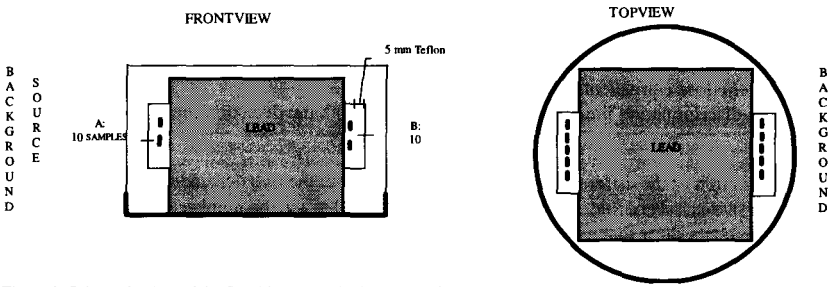

Figure 2: Schematic view of the Sensitive Extra Ambient Dosedetector

PERFORMANCE

Annealing and read-out The TL-material GR200A has an excellent performance, but only when specific and highly reproducable annealing and read-out procedures are applied. It's shown that an annealing temperature 240 °C for 10 minutes followed by fast cooling (>150 K/min) gives a good sensitivity. The *read-out procedure* consists of a plateau on 150°C for 5 seconds, followed by a heating-rate of 3 K/s to 240°C and a second plateau at 240 °C for 13 seconds. The integrating limits for TL-light are from 155°C to *the end* of the second platform. In this way the, non-stable peak 2 (peak temp ≈120 °C) is left out of the integration and peak 4 (peak temp 220 °C) is included as much as possible.

Uniformity and reproducibility To gain a batch of GR200A-samples with a great uniformity a selection of these 40 samples is made out of 100 'new' samples with a manufacturer-guarantee of max. 5% deviation. The standard deviation of the readings of 40 samples after equal irradiation with 10 mR is 0.8%. The standard deviation of 6 readings of each sample after 6 equal irradiations with 60 μGy is less than 2%.

Fading Three groups of 4 samples were placed outdoors for 14 days. One group was irradiated with 3,65 mR before the storage and one group was irradiated after the 14 days with 4,35 mR. After accounting for the different doses and the environmental dose in 14 days (measured with the third group) a fading of (2 ± 4) % (1 st.dev. of the mean) was calculated.

Zero-dose. The zero-dose of all samples was measured by reading them directly after annealing. Most of the readings of the zero-dose didn't exceed the minimum detectable value (determined by the variance in reader-background). The samples which did have a detectable zero-dose were left out of the selection.

Length of field-cycle. The length of the field cycle was determined by irradiating a group of 6 samples which different (low) doses. It was proven that the relative standard-deviation of the mean of 6 readings remained constant when the samples were exposed to 4 mR or more. This can be explained by writing the relative standard deviation σ_N as a function of the total counts n and standard deviation of the reader-background σ_B.(4)

$$\frac{\langle \sigma_N^2 \rangle}{N^2} = b + \frac{1}{n} + 2\frac{\sigma^2_B}{n^2} \qquad (1)$$

where b is a constant. In the case of GR200A and the used reader (modified Harshaw 2000) b>0.001, σ_B=10 and n > 4000 counts for a dose of 4 mR, leaving b, a constant, as the only significant contribution to the relative standard deviation. A dose of 4 mR is equal to the environmental background in 14 days. Fig. 4 shows the relative standard deviation of a TL-measurement as a function of the measurement time in a natural background.

Hygroscopicity. After 14 days storage the difference of the mean between 5 samples placed in a closed can, filled with water and five samples placed in a dry environment was non-significant (0 ± 2)%.

Energy response The relation of measured counts and irradiated dose (mR) is plotted in fig. 3. The defined dependence between ambient dose equivalent (cSv) and exposure (R) as given by ICRU (5) is plotted in fig. 3.

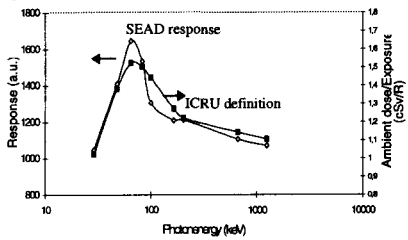

fig 3: TL-response of SEAD and ICRU definition

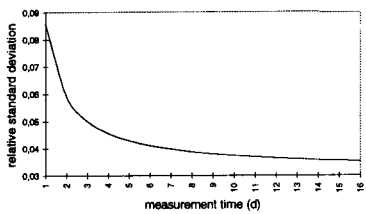

fig 4: Relative standard deviation as a function of measuring time

From fig 3 it's clear that the combinaton of lead, 1,5 mm Al and the 5 mm PTFE gives an excellent agreement with the ICRU definition of ambient dose equivalent.

Discrimination limit The discrimination limit, O_D, (not to be confused with the detection limit) is defined by

$$O_D = t_{\alpha/2}(n+m-2)S_d \qquad (2)$$

Where S_d is the standard deviation of the difference of the mean readings from the two dosemeters and $t_{\alpha/2}$ is the Student-t critical value for a two-sided confindence interval of 1-α and n and m are the number of samples in dosemeter A and B. Measurements show that n=m=10 is sufficient to reach the desired discrimination limit.

Field measurements Results of 4 measurements with SEAD are shown in table 1, two of them at one place at the site boundary of a research reactor and three at different places at the site boundary of a hospital in which radiation facilities are operated.

Table 1: Resluts of measurements with SEAD.

	Reactor	Reactor	Hospital 1	Hospital 2
Extra artificial dose (mSv/a)	0.032	0.037	-0.017	-0.032
'Background' (mSv/a)	0.73	0.68	0.86	0.87
Discrimination limit (95%) (mSv/a)	0.030	0.021	0.033	0.029

DISCUSSION AND CONCLUSION

SEAD is capable of distinguishing an artificial dose rate of 40 μSv/a from the natural background in a measuring time of 14 days. The performance tests show that SEAD shows excellent stability, in particular because of the use of (well-selected) GR200A-samples. Table 1 shows, however, that in absence of any artificial radiation source a difference in natural dose rate can be up to 32 μSv/a (hospital 2). This difference isn't a result of diffence in soil-type (both dosemeters are virtually on the same spot) but is due to diffences in surrounding elements (buildings, trees etc.) on both sides of the detector. This difference can only be established precisely when measurements around a facility can be performed with 'the source switched off'. In other cases the artificial dose rate as established by SEAD must be considered as best estimate, because it eliminates by far more biasing factors than methods used up to now.

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PLASTIC SCINTILLATOR WATER MONITOR RESPONSE TO ^{32}P

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INTRODUCTION

Water monitors equipped with a plastic scintillator are widely used to measure the radioactive concentrations of β-emitters in liquid waste. Plastic monitors are highly sensitive to high energy β-emitters like ^{32}P, but they are insensitive to low energy β-emitters like ^{3}H, ^{14}C. This is a result of the detector's surface being covered by a thin protective layer. A computerized plastic scintillator water monitor system was devised for high energy β-emitters at our facility. The response to ^{32}P was investigated so that optimum conditions for operating the monitor could be achieved.

MATERIALS AND METHODS

Fig.1 shows a schematic diagram of the sistem that is composed of a personal computer, plastic scintillator detector, controler and pipe lines. A computerized sequence for operating the monitor, background (BG) measurements, waste liquid sampling, sample measurements, and detector cleaning, was programmed. Signals of the detector were fed to a multichannel scaler(MCS) to analyze the time variations of the count rate.

^{32}P solution of 0.04-0.4Bq $\cdot$ cm^{-3} prepared in a radioactive liquid waste storage tank of100m^3 was used to measure the detector response, which was also analyzed by using a short cut circuit connected to a 20 ℓ polyethylene tank containing ^{32}P solution of 5Bq $\cdot$ cm^{-3}. The activity in the water sampled near the detector was measured by a liquid scintillation counter(LSC), and was compared with those values measurd by the water monitor, MCS, and a multichannel analyzer(MCA) . This was done to confirm and/or adjust the different methods giving the same values as LSC.

RESULTS AND DISCUSSION

Figure 2 represents an example of time variation of count rates measured at a dwell time of10sec/Channel. by MCS. The counting efficiency was 5.1 $\pm$0.1%. The time to replace BG water with waste

liquid was about 6 min. and a cleaning time was about 1min. This difference is due to the the dissimilarity of both pipe line lengths. Some measurements showed that the count rates after cleaning was a little higher than that of BG, because of the ^{32}P in the residual waste liquid in the pipe near the detector after the first cleaning. This means that contamination occasionally remains on the detector's surface after cleaning contingent on the pipe arrangement. The detector was dismantled and the surface was completely decontaminated by wiping it with soft tissue paper. Second cleaning also completely washed out any residual waste liquid. The detector was then free of any contamination.

Replace times, cleaning times, and repeat times should be experimentaly confirmed for each monitoring system, since each system differs in pipe line arrangement as well as length.

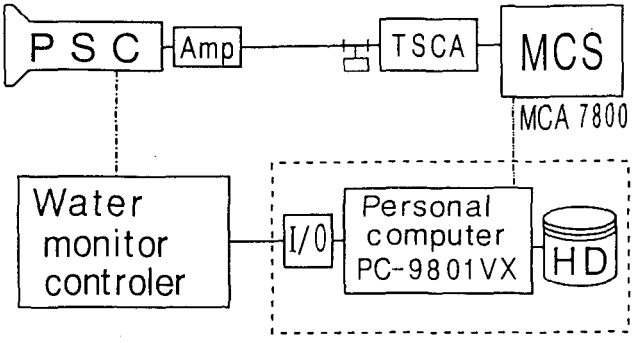

Fig.1. A schematic diagram of the computerized water monitor equipped with a plastic scintillator.
PSC; Plastic scintillator detector, MCS; Multichannel scaler
HD; Hard disk

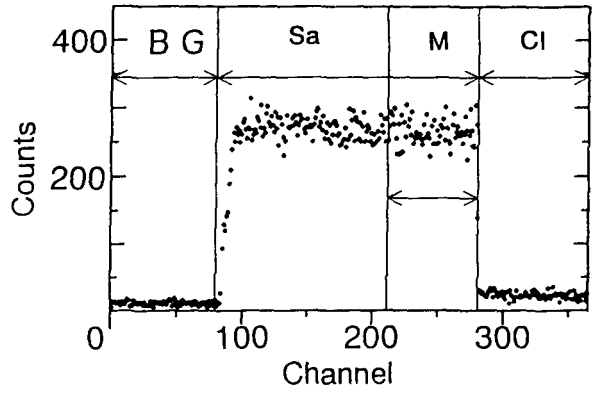

Fig.2. The time variation of the count rate of each step measured by MCS.
Dwell time; 10sec/Channel, BG; Background, Sa; Sampling, M; Measurement, Cl; Cleaning

SEPARATION OF THE INDUSTRIAL AND THE NATURAL COMPONENTS OF THE AMBIENT BACKGROUND BY A STATISTICAL METHOD

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INTRODUCTION

The continuously increasing number of industrial facilities involving nuclear tehniques, such as nuclear power stations, gamma irradiators, particle accelerators a.s.o., lead to new requirements concerning the control of the ambient background. By considering the ambient background as being composed by two components - the natural background and the industrial component - it is of great practical interest to measure separately this two components of different origins. As concerns the industrial component of the ambient background, it is represented mostly by the penetrating gamma-ray field, the other radiation fields being, more or less rapidly, absorbed after emission.

In this way the problem considered by us was to separate the natural background from the gamma -ray field of industrial origin (the usual method of subtracting the natural background from the mixed natural background + gamma-ray field is connected very often with some difficulties, as the unknown industrial gamma-ray field can not be canceled for the natural background measurement). For this purpose we decided to use the statistical discrimination method based on the ionization chamber as radiation detector.

PRINCIPLE OF METHOD

As it has been recently shown (1), the ratio between the variance and the mean value of the current of the ionization chamber, placed in a radiation field, $k = \sigma^2/\bar{I}$, depends on the radiation field type, through the mean value and the distribution function of the energy, lost by the ionizing particles in the chamber volume. Besides, for a given radiation field, the statistical factor k does not depend on the strength of the field, i.e. on the ionization current value. As a consequence, a new method of statistical discrimination of components in a binary mixed radiation field has been proposed (2). By following to the statistical discrimination method, for the case of the superposition of an industrial gamma-ray field (leading to an ionization current mean value $\bar{I}_\gamma$) and a natural background field (leading to an ionization current mean value $\bar{I}_B$), by measuring the total ionization current mean value $\bar{I}$ and its variance σ^2, the two components of the ionization current may be deduced from the following relationships:

$$\bar{I}_\gamma = \bar{I}\,(k_B - k)\,/\,(k_B - k_\gamma) \qquad (1)$$

and:

$$\bar{I}_B = \bar{I}\,(k - k_\gamma)\,/\,(k_B - k_\gamma) \qquad (2)$$

where k, k_γ and k_B represent the statistical factors for the mixed field, the gamma-ray field and the natural background field. As it has been also shown in (2), the range of use and the precision of the method are depending on the ratio k_B/k_γ between the statistical factors of the superposed fields.

In this way our first step in using the statistical discrimination method was represented by the measurement of the statistical factors k_γ and k_B. For this purpose we used a 10 l - atmospheric pressure air filled ionization chamber, biased at 400 V. The integration time constant of the electrometric scheme has been equal to 2 s. The variance and the mean value of the ionization current, corresponding to a given radiation field, were obtained by processing series of 300 readings.

We obtained for the natural background:

$\bar{I}_B = 9.5 \times 10^{-15}$ A $\qquad \sigma_B = 4.0 \times 10^{-15}$ A $\qquad k_B = 1.7 \times 10^{-15}$ A

By using a 370 MBq activity ^{60}Co gamma source, mounted at 10 cm from the ionization chamber wall, we obtained:

$\bar{I}_\gamma = 2.5 \times 10^{-10}$ A $\qquad \sigma_\gamma = 2.18 \times 10^{-13}$ A $\qquad k_\gamma = 1.9 \times 10^{-16}$ A

By looking at the measured values of the statistical factors, we have to emphasize the following points:

- the ionization current mean value corresponding to the gamma source is much higher than the natural background ionization current mean value and we shall consider to have in this case a practically pure gamma-ray radiation field.

- we don't know how pure is the natural background radiation field and the obtained ratio $k_B/k_\gamma = 9$ has to be considered as a lower limit of this ratio.

- at the same time the ratio value of about ten is proving the validity of the statistical discrimination method for this mixed field.

- the higher value of the statistical factor for the natural background radiation field could be qualitatively explained by the presence of ionizing particles of much higher stopping power than the secondary electrons produced by the gamma-ray atomic interactions; such ionizing particles could be, for instance, the μ -mesons or other cosmic rays components.

STATISTICAL UNCERTAINTIES

The relationships (1) and (2) are limited in use by the statistical uncertainties of determining the $\bar{I}_\gamma$ and $\bar{I}_B$ values. In order to deduce this statistical uncertainties, we have to estimate the experimental statistical errors associated with I and σ^2 and then to use the propagation error formulae. Let's emphasize the double role played by σ in this calculation, being at the same time an error (or a part of it) for the $\bar{I}$ value and a measured quantity with its own error. As concerns the statistical error on the σ value this one depends only on the number of readings (3) (for instance, the statistical error at a 95% confidence level is of about 10%, for 300 readings, and is varying as $n^{-0.5}$, where n represents the number of readings).

The statistical relative uncertainties of the statistical discrimination method, at a 95% confidence level obtained, in such a way, are the following:

$$\varepsilon_\gamma = \left(\frac{4k(1+\beta_\gamma)}{\bar{I}\,\sigma^2 n^2} + \frac{2.25\beta_\gamma}{n} \right)^{1/2} \qquad (3)$$

$$\varepsilon_B = \left(\frac{4k(1+\beta_B)}{\bar{I}\,\sigma^2 n^2} + \frac{2.25\beta_B}{n} \right)^{1/2} \qquad (4)$$

where:

$\beta_\gamma = k^2 / (k_B - k)^2$

$\beta_B = k^2 / (k - k_\gamma)^2$

n = number of readings

σ = integration time constant

In order to have a better understanding of the relationships (3) and (4), in figure 1 the dependences of ε_γ and ε_B on the gamma-ray absorbed dose rate in air for a given natural background absorbed dose rate in air equal to 10μrad/h (0.1 μGy/h), in the case of a 10 l ionization chamber with a τ = 30 ms integration time constant and n = 1000 readings, are shown.

From the practical point of view, we are highly interested, in a few absorbed dose rates. First of all the gamma - ray absorbed dose rate representing the detection limit, corresponding to a 100% statistical uncertainty, is equal to 0.02 μGy/h. The gamma - ray absorbed dose rate of 0.1 μGy/h, equal to the assumed natural background absorbed dose rate, may be measured with an acceptable statistical uncertainty of less than 20%. The gamma -ray absorbed dose rate of 0.5 μGy/h corresponding to a total absorbed dose rate in air of 0.6 μGy/h corresponding to the admissible dose rate for population (5 mSv per year), may be measured, with a statistical uncertainty of about 5%, while the statistical uncertainty for the assumed 0.1 μGy/h natural background absorbed dose rate measurement does not exceed 12%.

CONCLUSIONS. The conclusions are the following:

1) The statistical discrimination method provides for a simultaneous measurement of the natural background and of the industrial gamma-rays absorbed dose rates in good conditions.

2) For a 10 l - atmospheric pressure air-filled ionization chamber and 1 min total measuring time, the estimated gamma-ray detection threshold may be as low as 0.02 μGy/h, by assuming a 0,1 μGy/h absorbed dose rate in air for the natural background (see fig. 1).

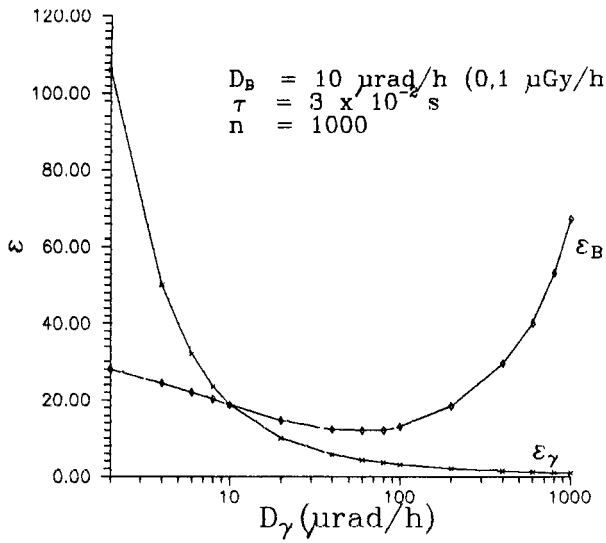

Fig. 1 - Statistical uncertainty of the method (example)

The results were checked by using a functional model with the block scheme shown in fig. 2. The agreement with the calculations was good.

IC	ionization chamber
EA	electrometric amplifier
S	sampling circuit
ADC	analog - to - digital convertor
P	microprocessor
DAC	digital - to - analog convertor
D	display

Fig. 2 - Block scheme of the functional model

3) The proposed statistical discrimination method leads to lower statistical uncertainties than those corresponding to the usual method, based on the subtraction of the natural background. For the same experimental conditions as those used in fig. 1, the method of natural background subtraction would lead to a 0.04 μGy/h for the gamma - ray detection threshold and to a statistical uncertainty of measurement for the 0.1 μGy/h gamma - ray absorbed dose rate in air equal to 47% (instead of 19%, given by the statistical discrimination method).

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HIGH SENSITIVITY DOSIMETRYCAL SYSTEM FOR ENVIRONMENTAL, NATURAL RADIOACTIVITY SURVEY

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The high sensitivity dosimetrycal system is destined for measuring of the absorbed dose in air which is produced of the X and gamma radiations within the interval 30 keV - 3 MeV.

This system consists in a dosimetrycal box which has a cylindrical form with the next dimensions: $\phi = 50$ mm, h = 10 mm, which contains five thermoluminescent detectors type $CaSO_4$:Dy.

The detectors are tablets with dimensions: ϕ = 10 mm and h = 0.8 mm.

The raw materials used for obtaining the thermoluminophor powder are the pure substances: $CaSO_4$ u.p.; H_2SO_4 e.g. and spectral Dy_2O_3.

The tablets are obtained pressing the thermoluminophor powder and teflon mixture. In this way the thermoluminophore substance is uniformly included in a teflone matrix.

$CaSO_4$:Dy thermoluminescent detectors (TLD), are obtained in our laboratory by the controlled dopping of the $CaSO_4$ crystal with the activant dysprosiu.

The original technology allows obtaining of the thermoluminescent crystals with high sensitivity leaving from very pure substances.

The measurement of the absorbed dose is based on the principle of the thermoluminescence, consisting in the occurence of excited states in the $CaSO_4$:Dy crystal during irradiation; when heating, thip crystal, following a pre-established thermal cycle, returns to its background state by emitting a fluorescence hight whose intensity in proportional with the absorbed dose [1]. We use for reading of the detectors a Victoreen 2800 equipment.

The dosimetric system is exposed to the environmental irradiation for a minimum period of 30 days and is placed at a standard height, 1 m from the soil.

The main dosimetric characteristics of the system are in accordance with the requirements of the International Standards [2].

They are presented in the following:

- the absorbed dose measuring interval has between (10^{-5} - 10^{-1}) Gy; linearity value R is defined by the ratio between to measured dose (D) and the conventionaly true dose EIR = D/E:

a) whitin the interval (10^{-5} - 10^{-4}) Gy, $0.5 \leq R \leq 1.5$;

b) whitin the interval (10^{-4} - 10^{-1}) Gy, $0.75 \leq R \leq 1.25$;

- the homogenity of the lot detectors from dosimeters is $\varepsilon \leq 30$ %; one lot has 50 detectors;

- the reproductibility of the irradiation during 10 complete cycles: irradiation, reading of the information, regeneration of the detectors for absorbed doses of 5.10^{-5} Gy and 2.10^{-4} Gy is defined by a standard deviation, $\sigma \leq \pm 7.5$ %;

- the dependence of the response with the radiations energy on the whole energetic interval is given by a permissible value of R, $R \leq 2$.

The high sensitivity dosimetrycal system can be used in environmental for measuring absorbed dose in air, he is not influenced by climatic factors.

We hope as with our dosimeter systems to participate to compare international.

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BETA TIME-SPECTROMETRY WITH MULTICHANNEL SCALING – A PORTABLE LOW LEVEL MONITOR FOR NATURAL AND ARTIFICIAL RADIONUCLIDES IN AIR, WATER AND SOLIDS

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Abstract: **The portable beta-/gamma-sensitive Low Level Monitor LLM 500 is equipped with hardware and software for automated repeated measurements of large-area samples (diskettes), up to 100 times for gate times between 1-100 min. Data stored and released to a printer or a PC yield a spectrum of activity vs time. The value of such time-spectra to discriminate natural and man-made radionuclides simply by their half-life, is evident. Three examples discriminating fresh and old radon, radon and thoron decay products, and radon decay products from filtration of snow water are illustrated and explained.**

INTRODUCTION

There is a need to discriminate natural and man-made radionuclides because of their different values both for radiation protection and for the public. In addition, such nuclide-specific measurements should be accurate, reliable, low-cost, and quick and the instrument should be easy to handle and the procedure easy to understand for laymen. There are two nuclide-specific properties which may be used to discriminate natural and man-made radionuclides: the energy of their gamma and alpha radiation, and half-life. New germanium and silicon detectors have made gamma- and alpha energy-spectrometry to be beautiful swans for the purpose stated and have left old time-spectrometry to be an ugly duckling. All natural and man-made radionuclides which need to be identified for practical purposes in public, such as the radon and thoron decay products Pb-214, Bi-214, Pb-212, and Bi-212, and reactor-made I-131, Cs-137 and Sr-90 have clearly different half-lives of minutes, hours, days and years and may thus be discriminated by time-spectrometry or in modern terms "multichannel scaling". In addition, the radionuclides listed are all beta emitters and are most likely to be deposited on large solid surfaces or collected from air or water on large-area filters. For such samples large-area, beta-sensitive, sealed proportional detectors of low cost and high sensitivity are a better choice than any alpha or gamma detector. Proportional detectors are well known from commercial hand-held contamination monitors and may be used for the purposes stated if endowed with appropriate hardware and software. Surprisingly, the Living Level Monitor LLM 500 of Münchener Apparatebau (1, 2) is the first and portable (7 kg, 28 cm x 29 cm x 19 cm) Low Level Monitor with automated time-spectrometry. Its main features and some practical results of three years experiences are reported in the following. The LLM 500 means a renaissance of beta time-spectrometry. Such decay curves to distinguish radon and thoron were first used by E. Rutherford, J. Elster and H. Geitel, and H. A. Bumstead in 1901 - 1904.

MATERIALS AND METHODS

The instrument LLM 500 contains a sealed proportional detector with a 7 mg/cm² aluminum window, argon-methane gas filling and an active area of 11 x 11 cm². The detector is very sensitive for beta and much less for gamma radiation. Its unshielded background rate is ~150 cpm at 70 nGy/h. The detector is connected to a pulse counter and to hardware electronics with software. The instrument has a front panel with a 4 x 4 array of numbered or symbolized buttons as keyboard, a PC-type opening for the measuring diskette, and a four-line, 20 digits each LC display, as well as data memory and output via a RS 232 interface. The LLM 500 is designed to accept samples in 135 x 200 x 2 mm³ diskettes. There are three types of diskettes. Type one is made of two pieces of cardboard each 1 mm thick, with an opening of 10 cm diameter, a glass fibre filter MN 85/90 (Macherey-Nagel, D-52348 Düren) in between and glued together. The diskette fits a patented holder attached to a Staplex high volume sampler with a flow rate of 800 L/min. With such diskettes, grab samples of air are drawn for subsequent measurement in the LLM 500, calibrated in Bq/m³ as equilibrium equivalent radon concentration EER (potential alpha energy concentration). Type two diskettes are made of PVC with an opening of 11.1 cm diameter. They are closed on one side with a 12 x 12 cm

piece of transparent adhesive tape cut from rolls of tape used to protect books. Thus a plate is formed to accept smear tests or filters of 11 cm diameter from water filtration. Such diskettes are also used for thin solid or powder samples between two tapes. Type three diskettes are plates made of white polystyrene and charged by friction to collect electrically charged radon decay products in air, called Philion Plates, described elsewhere (3, 4). All three types of diskettes fit in the slit of the LLM 500 like hand to glove.

After insertion the sample comes to rest under the proportional detector and above a 0.1 mm thick tungsten plate. Tungsten by virtue of its high atomic number is a good beta reflector and increases the counting efficiency for thin samples by about 20 %. The activity calibration factor k and the instrument efficiency e are defined in conformity with SI units by virtue of the dimensional analysis [k] = [Bq/cps] = [1] and [e] = [1/k] = [%]. As described elsewhere (1), k has been determined and confirmed by several different methods to be 3.0 ± 0.3 Bq/cps for two beta/gamma emitters such as the radon decay products in EER [Bq/m³]. The high efficiency of >30% for radon decay products (EER) means a high resolution both of activity (2 Bq) and time (less than 5 min). Proper functioning of the LLM 500 may be checked any time with a test diskette containing 1.9 g KCl with 30 Bq K-40 at k = 6.0 Bq/cps. The software and the simple keyboard of the LLM 500 allow to choose the gate time between 1 and 100 min and the number of repetitions between 1 and 100 or free running either for measurement of the background or of the sample , after subtraction of one background. The calibration factor may be changed between 0.1 and 9.9 and the volume or the weight of the sample is entered via the keyboard for calculation of activity concentrations. All data are stored and may be released either to a printer in columns or to a PC to be plotted as log of net count rates or activity concentrations vs time from which half-lives are extracted.

RESULTS AND DISCUSSION

The discriminating power of beta time-spectrometry is demonstrated by 3 practical examples simply drawn from air and snow water.

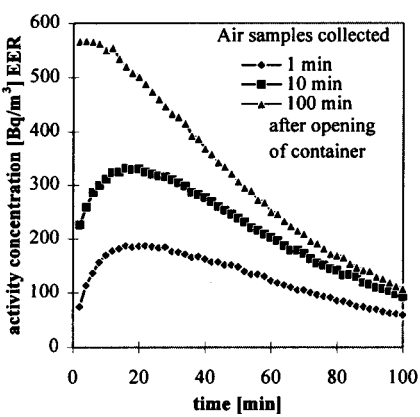

Fig. 1. Activity concentration on filter vs time after collecting sample. Note the difference between "fresh" radon (curve of 1 min after opening the container with uranium minerals) and old radon.

A problem often encountered is to locate a source of fresh radon, such as the entrance of radon from the soil or a water system into a room. Fresh radon has a low factor F of radioactive equilibrium, i.e. contains a larger than normal fraction of Po-218 (half-life 3 min) relative to Pb-214 (27 min) and Bi-214 (20 min). Such an atmosphere is conveniently created in a laboratory by opening an air-tight container with uranium minerals. 1 m³ of air was collected with a Staplex sampler within 75 s. Three samples were taken 1 min, 10 min, and 100 min after opening the container in a room of 100 m³. The glass fibre filters of 10 cm diameter held in cardboard diskettes were evaluated in two LLM 500. The time-spectra of total beta activity are quite different for the 3 samples. In fig 1, spectrum 1 shows a sharp rise of measured activity on the filter within the first 10 min. The large fraction of the alpha emitter Po-218 contributes only after its decay while the measurement is going on. Sample 3, collected 100 min after opening the container shows a normal decay curve for a normal factor of equilibrium of F ~ 0.6. The initial plateau for about 10 min is very practical. No correction to account for time of sampling, transfer and first measurement is necessary, as long as the total time does not exceed 10 min. Spectrum 2 is in between spectra 1 and 3. To obtain information on F, 10 points at 1 or 2 min intervals are quite sufficient.

Thoron (Rn-220) and its daughter Po-216 are both alpha emitters, with short half-lives of 55 s and 0.15 s resp. Their decay products Pb-212 and Bi-212 are beta/gamma emitters with convenient half-lives of 10.6 h and 1 h. Because of its short half-life, Rn-220 was believed not to leave the matrix of its mother Ra-224, such as building materials containing sand, frequently with monazite $CePO_4$, with some Th replacing Ce.

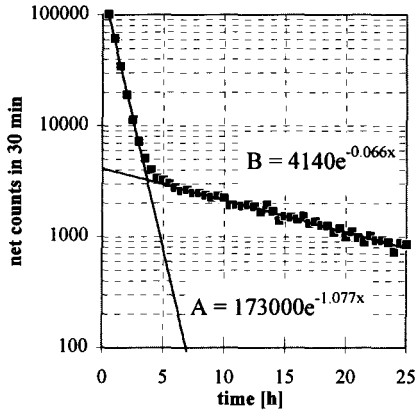

Fig. 2. Decay of a sample of 24 m³ of air, containing radon and thoron decay products. 20 x 30 x 24 = 14400 counts are equivalent to 1 Bq/m³. Note the two branches for radon and thoron decay products.

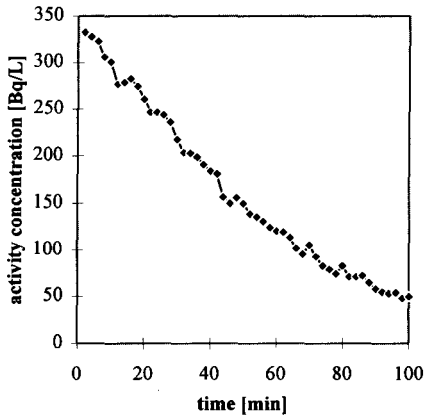

Fig. 3 Decay of a sample of radon decay products from 170 mL of melted snow, collected by simple filtration on a glass fibre filter. Half-life 40 min.

With beta time-spectrometry the presence of thoron decay products was demonstrated in normal room air. 24 m³ were collected on a glass fibre filter within 30 min as described before. 50 measurements automatically repeated at 30 min intervals were made with the LLM 500. Its calibration factor k = 3 Bq/cps means 20 cpm are equivalent to 1 Bq, or 20 x 30 x 24 = 14400 counts in 30 min for 24 m³ are equivalent to 1 Bq/m³. Background count was 4774 counts in 30 min, first net count was 101230. Fig. 2 shows the semilogarithmic plot of count rate vs time, yielding two equations: For radon decay products A = 173000 exp(-1.077 x) giving an activity of 12 Bq/m³ and a half-life of 38 min. Note this value is smaller than the value of 52 min taken from spectrum 3 in fig. 1, where Po-218 still contributes because of the short sampling and measuring time. For thoron decay products B = 4140 exp(-0.066 x) giving a nominal activity of 0.3 Bq/m³ and a half-life of 10.5 h.

Glass fibre filters were found to be very efficient (> 95 %) and reproducible for rapid filtration of radon decay products in water. This was unexpected and may be the reason why it was not tried before. No reference was found in the literature on ten different methods for measuring radon or radon decay products in water. Equilibrium (after 3 h) may always be assumed to exist in water. For water filtration we use the same filters as for air sampling, type MN 85/90, 90 g/cm², 0.40 mm thick, 0.5 µm retention (information from the manufacturer). With a filter of 11 cm diameter in a cylindrical Büchner funnel, filtration without a water jet pump takes only 5 min for 1 L. Wet filters are dried in 1 min on a 300 °C hot stage and measured on a plate-type diskette in the LLM 500. Neutral or alkaline state is good, acidic is poor for retention. The surface of glass fibre filters may act similar to the surface of glass electrodes (Nernst' potential) to provide negative ionic bonding to cations. For fig. 3 fresh snow was melted to 170 mL of water, showing a concentration of 330 Bq/L from snow-out of radon decay products. The decay curve in fig. 3 is for illustration, for a measurement the first value of 1 or 2 min is sufficient. In fresh rain concentrations up to 100 Bq/L radon were found. The detection limit is 1-2 Bq/L, found in any drinking water, for which a US EPA limit is 11 Bq/L.

CONCLUSIONS

Ten years after Chernobyl the hope is expressed, that the instrument described may not be required in the future to discriminate natural and reactor-made airborne radionuclides and that its use be reserved to research, teaching and services in natural environmental radiation.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
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PAPER TITLE AN ENVIRONMENTAL DOSIMETRY SYSTEM UTILIZING HYPER-SENSITIVE MATERIAL AND HOT AIR HEATING

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ABSTRACT (See instructions overleaf)

A completely portable and self-contained environmental dosimetry system, based on proven hot-gas TLD technology and enhanced with hyper-sensitive TL material, neural network dose computation, and requiring only electrical power for operation is presented and performance testing is discussed. The dosimeter is composed of a symmetric holder containing proper filtration to measure $Hp^*(10)$, $Hp^*(0.07)$ and discriminate low, intermediate and high energy environmental radiation. The TLD material is newly developed LiF:Mg,Cu,P mounted on Kapton® or sandwiched between Teflon® sheets. The new TLD material is tissue equivalent and is shown to be well suited for enviromnmental dosimetry with higher sensitivity, more uniform response with respect to energy, and negligible fade. The TLD Workstation is equipped with an Air Supply Unit and provides portable field readout capability. The workstation is composed of a TLD reader, associated application software system, and a personal computer. The workstation software provides instrument control, data acquisition and storage, QC monitoring, and dose calculation algorithm. A new aspect of QC monitoring is the glow curve analyzer, which provides automatic screening to identify abnormal glow curves. The recently developed neural network-based algorithm computes the desired dose quantities more accurately than a simple dosimeter reading or a decision tree dose algorithm. This environmental dosimetry system was designed to comply with the current ICRP requirements and the proposed ANSI requirements.

THE DETERMINATION AND OPTIMIZATION OF MULTIPLE MEASUREMENTS FOR ACTIVITY CALCULATIONS

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ABSTRACT

The calculation of the radioactive contents of a waste container is often performed by a dose rate per activity ratio. This ratio is determined for a selected detector location, gamma energy and activity distribution. If the spatial distribution of the activity is unknown, however, the resulting calculated assessment of the activity from a single detector may contain a high degree of uncertainty. In addition, the optimal location of the measurements becomes a function of the background radiation, the amount of activity present and the desired statistical confidence of the result. If a set of N measurements is considered an N vector response, the incorporation of multiple detector measurements can reduce the uncertainty of the calculated assessment. The multiple detector calculations are performed by applying dimensional measurement space and a unit response set to the problem of assessing and optimizing multiple independent detector measurements of various waste containers. The associated errors of multiple measurements are calculated for various detector placement configurations. The calculations indicate that multiple detector measurements and optimal placement of the measurements combined with appropriate mathematical techniques significantly reduce the assessment uncertainties due to lack of knowledge of the spatial distribution of activity.

INTRODUCTION

A number of methodologies have been utilized to reduce the uncertainty in the assessment of unknown spatial distributions of activity. Rotation of the source container, segmenting the measurement volume seen by a detector and linear combinations of detector responses have been demonstrated as methods in reducing the assessment uncertainty for such configurations as a point source and a small cylinder source within a waste container (1).

The use of a detector response space for multiple measurements has been previously introduced to the problem of measuring the largest uncertainty that a detector system will generate when assessing randomly distributed radioactive material (2,3,4,5,6). The mathematical framework is independent of the container volume and the type of detector present. In this paper, detector response space is applied to the problem of interpreting detector responses in the presence of randomly distributed radioactive material. The resulting method, the ray-method, can be utilized to assess the amounts of radioactivity that could result in the given detector response (7).

FUNDAMENTAL CONCEPTS

When accounting for the detector responses which are the result of random or unknown spatial distributions of activity, the use of sets becomes a convenient terminology. A number of definitions have been utilized which aid in describing the set of detector responses that result from the distribution of a fixed amount of activity (2). If f_n is the nth detector response from a unit amount of radioactivity then the set of N detector response functions can be represented as

$$f(x,y,z) = (f_1(x,y,z), f_2(x,y,z), \ldots f_N(x,y,z)). \qquad (1)$$

The set of detector responses from all possible locations of a single unit point source of

radioactive material is called the point source response set. If (X,Y,Z) represents the set all possible positions that the activity may be located, the point source response set is defined as

$$F=\{f: f=f(x,y,z) \text{ for all } (x,y,z)\in(\mathbf{X},\mathbf{Y},\mathbf{Z})\}. \quad (2)$$

Let $r(x_0y_0z_0)$ represent the density of the radioactive material at point $(x_0y_0z_0)$. The integrated response to a source density distribution $r(x,y,z)$ is

$$c=\int_V r(x,y,z)\,f(x,y,z)\,dxdydz. \quad (3)$$

The set of all detector responses from any spatial distribution of one unit of radioactive material is called the complete response set. The complete response set can be represented as

$$C=\{c: c=\int_V r(x,y,z)\,f(x,y,z)\,dV \text{ for all } \int_V r(x,y,z)\,dV=1\}. \quad (4)$$

The complete response set has been shown to be the smallest convex set (i.e. the convex hull) that contains the point source response set (2). This property of convexity allows the complete response set to be constructed from the point source response set and the set of all distributions of activity to be defined in terms of point distributions of activity.

METHODOLOGY

A given N detector response, $(h_1,h_2,...,h_N)$, resulting from some unknown amount of activity creates a line passing through the origin with slopes

$$s_1=\frac{h_2}{h_1},\ s_2=\frac{h_3}{h_1},\ \ldots,\ s_{N-1}=\frac{h_N}{h_1}. \quad (5)$$

Detector responses on the above line are the result of different amounts of activity with the same distribution. The intersection of this line with the complete response set C, forms a line segment which represents different distributions of a unit amount of activity that result in detector responses which satisfy equation (5). The respective distances from the origin of the endpoints of this line segment are the maximum and minimum distances from the origin that result in the above detector response proportions from one unit of activity. Two ratios are computed by dividing the respective distance from the origin of each of these endpoints into the distance from the origin to the given detector response h. The resulting ratios give the maximum and minimum activity that could result from the point $(h_1,h_2,...,h_N)$. The assessment equation is

$$\min_{g\in C}\sqrt{\left(\frac{h_1^2+h_2^2+\ldots+h_N^2}{g_1^2+g_2^2+\ldots+g_N^2}\right)} \le A \le \max_{k\in C}\sqrt{\left(\frac{h_1^2+h_2^2+\ldots+h_N^2}{k_1^2+k_2^2+\ldots+k_N^2}\right)} \quad (6)$$

where the points are subject to the constraints

$$f_2=s_1f_1,\ f_3=s_2f_1,\ \ldots,\ f_N=s_{n-1}f_1 \text{ for all } f\in C \text{ and all } (x,y,z)\in(\mathbf{X},\mathbf{Y},\mathbf{Z}). \quad (7)$$

APPLICATION

The above methodology was applied to a waste container problem. The necessary calculations were simplified by several assumptions. The radiation detectors are limited to two and are considered point detectors. Only the unscattered flux is taken into account. Self-attenuation and the attenuation coefficient outside the source container are considered negligible. The detector responses are proportional to the amount of radioactive material present for identical distributions of activity. The background levels of radiation are ignored. Statistical uncertainty of a detector response is considered negligible.

DISCUSSION

Calculations of the above application demonstrate that the addition of a second detector measurement can decrease the calculated uncertainty by approximately 30-fold. The application relied on the assumption that the activity is proportional to detector response. The assessment of nonproportional response sets (i.e. self-shielding) can be performed by generating successive sets of incremented activity and searching for amounts of activity for which overlap of the detector responses ceases (5). In some cases, a self attenuation function for limited distributions can be calculated which may give a functional relationship between the detector responses and the activity (8).

Although the above application relied on point source formulations, the utilization of standard shielding codes to construct the complete response set from the generated point source response set and subsequently to solve the interpretation of the given detector responses should pose few difficulties for calculations not involving self-shielding.

CONCLUSION

The application of response sets to the problem of multiple detector measurements allows the analyst and designer to calculate the actual and maximum calculated activity uncertainty due to spatial uncertainty of the activity distribution. With this information detector systems can be optimized to minimize the maximum uncertainty and measurements can be exploited to a full extent.

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LIQUID SCINTILLATION COCKTAILS COMPARISON FOR TRITIUM CONTAMINATION MEASUREMENTS

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INTRODUCTION

Liquid scintillation counting is one of the most used techniques for the measurements of tritium contamination [1]. Until few years ago a problem related to this kind of measurement was the potential toxicity of the liquid cocktails used to produce the required scintillation. Some new products that guarantee an almost negligible impact on the environment and that are no longer toxic for the operators are now available.

Some of this new scintillation cocktail are suitable to be used for tritium measurement. Due to the great benefit from the health point of view of these new materials a test of their scintillation performance has been done at the ENEA centers to select the product having the best characteristics for tritium measurement.

THE COCKTAILS

Three new generation cocktails (biodegradable) were used in our test together with an old toxic (xylene based) scintillation liquid widely utilised in the past. Some characteristics of the examined cocktails are given in table 1, where the new products are indicated with the generic names SAFE 1, SAFE 2 and SAFE 3.

Table 1 - Cocktails characteristics

Cocktail	Applications	General composition	Flash point (°C)
SAFE 1 *low toxicity*	Low level counting, high sample acceptance	___	> 130
SAFE 2 *low toxicity*	High counting efficiency	Di-isopropyl-naphthalene (DIN)	148
SAFE 3 *low toxicity*	Low level counting	DIPN	150
OLD *toxic*	Large variety of application	Xylene	___

The four cocktails are appropriate for low level counting when large amount aqueous samples are needed.

INSTRUMENTS, METHODS AND SAMPLES

A Wallac 1220 Quantulus counter was used for all the measurements. The counter includes lead shielding, cooling system, liquid scintillation guard counter and pulse analyzer, to reduce the background level.

The Zinsser 900 vials (teflon coated) that were used for the counting process.

The preparation of the samples in the case of the cocktails SAFE 1, SAFE 3 and OLD was carried out adding 10 ml of scintillation liquid to 10 ml of tritiated or bidistillated water, for the SAFE 2 cocktail 9 ml of water and 11 ml of cocktail were used.

The percentage of the scintillation liquid used in the sample solutions was always that suggested by the cocktail manufacturer.

The counting time for all the measurements was 120 minutes. Two independent energy windows were fixed for each sample; the larger one (the same for all the cocktails) was choosen to contain all the tritium energy spectrum and the other was selected to maximize the figure of merit for the specifically used cocktail.

THE TEST PARAMETERS

The parameters considered to compare the cocktail performances were the efficiency and the Minimum Detectable Activity (MDA). The former was determined measuring a standard tritium source. The uncertainty on the efficiency determination can be calculated taking into account the following components. A first source of error is the uncentainty in the tritium quantity used in the sample. The tritium source used in our test is an organic substance soluble into water prepared by the Amersham International and distributed by Wallac. The uncertainty in the activity of this standard source is about 7.9 %. Other statistical errors have to be taken into account for the samples preparation depending on the specific cocktail and counter. In table 2 the total error (combined standard uncertainty) for each cocktail is reported [2].

Table 2 - Statistical error in efficiency evaluation

Cocktail	Type B uncertainty (%)	Type A uncertainty (%)	Combined standard uncertainty (%)
SAFE 1	7.9	5.9	9.8
SAFE 2	7.9	5.5	9.6
SAFE 3	7.9	5.7	9.7
OLD	7.9	7.6	10.9

The MDA has been evaluated with the following formula [2,3]:

$$MDA = \frac{4.66 \cdot Sb}{\varepsilon \cdot V \cdot 60} \quad [Bq/l]$$

where:
S_b is the estimation of the standard deviation (cpm)
ε is the efficiency
V is the volume of the sample (l)

Table 3 - Cocktail performances

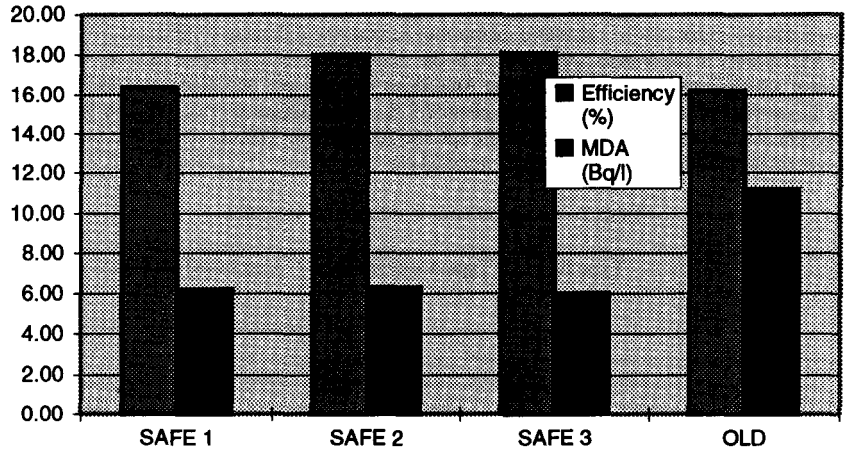
Cocktail	Window (channels)	Background ± σ (cpm)	MDA (Bq/l)	Efficiency (%)	FM	FMV^2
SAFE 1	20-270	2.23±0.13	6.2	16.4	120	12000
	20-120	0.63±0.08	4.5	12.8	255	25500
SAFE 2	20-270	2.48±0.13	6.3	18.0	130	10530
	20-130	0.70±0.08	4.4	15.1	325	26325
SAFE 3	20-270	2.51±0.14	6.0	18.1	130	13000
	20-130	0.76±0.10	4.9	15.1	296	29600
OLD	20-270	3.06±0.23	11.2	16.2	85	8500
	20-119	0.85±0.10	5.7	13.3	207	20700

RESULTS AND CONCLUSIONS

The results of the measurements were elaborated and the relevant parameters are presented in table 3.

Fig. 1 shows a graphic presentation of the efficiency and MDA for the various scintillation liquid tested. The conclusion that can be drawn from our results is that the three SAFE cocktails used show a similar behaviour and have better characteristics than the OLD cocktail

Figure 1 - Performance comparison



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A SIMPLE CALIBRATION METHOD FOR THERMOLUMINESCENCE DETECTORS USED IN ENVIRONMENTAL MEASUREMENTS

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INTRODUCTION

Recently, it has been proposed a calibration method for potassium halide thermoluminescence (TL) detectors (1). The procedure is based on the irradiation of potassium halide TL detector due to β particles, emitted by natural radioactive nuclide ^{40}K contained in the crystal itself as well as in a coat made of crystalline KCl salt surrounding the detector during a period of the calibration exposure time. The aim of this paper is to extend the proposed method to TL detectors manufactured from other materials. This is especially important for high sensitivity TL detectors used in environmental investigation, since the doses involved in the proposed method are comparable to the doses usually measured in environment.

GENERAL CONSIDERATION

To illustrate the method let us at first consider TL detector manufactured from KCl (for example, KCl:Eu) embedded into a coat of KCl crystalline salt as depicted in Fig. 1.

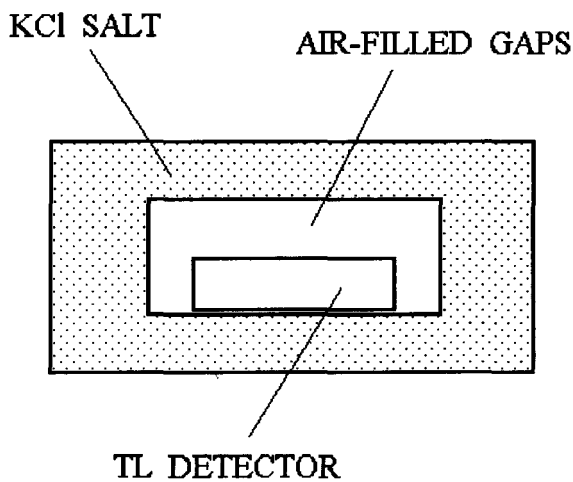

Figure 1. Schematic diagram for detector calibration exposure

If the thickness of the KCl coat is much larger than the range of the most energetic β particles, emitted by ^{40}K, and if the air filled gaps between detector and the inner surface of KCl coat are small enough, according to the cavity theory (2) the absorbed energy in the detector is equal to the energy produced per unit mass of KCl. Thus, the absorbed in the detector dose D is given by

$$D = C\,E\,t$$

where C is the activity of ^{40}K per unit mass of KCl, E is the average β ray energy per disintegration, and t is the time of exposure. Substitution of the values of C and E (3) in this expression gives the absorbed dose in the detector

$$D = 4.9\,t,\ \mu Gy,$$

where t must be expressed in hours. Therefore the exposure rate in potassium halide thermoluminescence detector embedded into a crystalline coat is 4.9 μGy/h and if the exposure time is equal 1 - 2 days, the absorbed dose must be equal to 100 - 200 μGy (10 - 20 mrad), i.e. the value, comparable to quarter dose usually measured in environment. Thus the calibration method is useful for detectors applied in environmental investigations. Unfortunately the fading of the KCl:Eu detectors according to recent investigation (4) is large, and because of that these detectors are not promising in environmental measurements. More perspective are LiF or Al_2O_3 detectors which show high sensitivity and low fading, adequate for environmental dosimetry.

RESULTS OF MEASUREMENTS AND ACCURACY OF THE METHOD

It is clear, that for LiF or Al_2O_3 detectors embedded into a KCl coat the exposure rate must be less (compared to potassium halide detectors) for lack of the self-irradiation effect. In order to determine the exposure rate for LiF and Al_2O_3 detectors we have made a comparison of TL signals, obtained from these detectors after theirs exposure in KCl coat, and TL signals, obtained after the detectors irradiation by using ^{137}Cs γ-ray source of well-known activity. The irradiation was made in condition of the existing of the electron equilibrium. In experiments were used detectors LiF:(Ti, Mg) and anion-defective α-Al_2O_3 :C single crystals, commercially grown in Russia as DTG-4 and TLD-500K, respectively. The size of the crystals were 5 mm diameter and 1 mm thickness. The crystalline KCl coat salt used for irradiation was with wall thickness around 1 g cm^{-2}. From the results of the comparison it was found that exposure rate in TL detectors LiF and Al_2O_3 embedded into a cavity in KCl crystalline block is 2.09 μGy/h and 1.73 μGy/h, respectively. These values of the exposure rates may be used for detectors calibration.

As was shown in (1), the main sources of errors in the proposed calibration method are: the impurities of the KCl salt which may change the content of ^{40}K per unit mass of salt as compared to that of pure KCl salt, the energy lost by the β particles because of the existing air-filled clearances between the surface of the detector and that of the cavity in the KCl salt, the ^{40}K γ radiation which may give a small contribution to the absorbed detector dose, variations of the natural radiation background during the exposure time, the statistical variation of the number of β particles which were absorbed by the detector during the

exposure time. In (1) was also shown that the total uncertainty of the absorbed detector dose due to all these sources of errors under certain conditions does not extend the common error of 1.5% - 2.0% of traditional calibration methods practiced in specially equipped metrological laboratories (5).

CONCLUSION

In conclusion, we have developed a very simple and highly accurate calibration procedure for LiF and Al_2O_3 detectors based on application crystalline KCl salt block as a source of β radiation due to the natural content of the radionuclide ^{40}K. The method does not require artificial radiation sources or precision equipment and may be carried out in any laboratory. The proposed procedure may be used in environmental investigation, since the doses involved in the method are comparable to the doses usually measured in environmental dosimetry.

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APPLICATION OF REAL TIME SPECTRUM MEASUREMENT TO RADIATION MONITORS

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ABSTRACT

A multichannel analyzer (MCA) and two realtime spectrum monitoring methods have been developed for use in radiation monitors. The new MCA was designed to be installed at a local site as a component of a radiation monitor. The MCA repeats spectrum measurement at short intervals (Δt) and, after each measurement, transmits a spectrum datum to the operation console. The authors applied two methods to process Δt spectrum counts for each channel for longer time interval.

One method of processing counts is the "running average (RA) method". The other method is the "exponential smoothing (ES) method", which simulates RC rate meters by subtracting a fraction corresponding to the accumulated counts. Relative standard deviations for each channel can be made the same by selecting an appropriate value. The response with the "ES" method is initially faster than that with the "RA" method, but the "RA" method allows a full response to be reached at a predictable time.

INTRODUCTION

In order to measure various radionuclides contained in typical radioisotope facilities, gamma spectrometry is necessary. The detectors of radiation monitors are installed at a local site, separately from a control room.

Radiation monitors must be able to operate continuously and respond quickly to changes in radiation levels. The MCA conventionally used in labolatory analysis is not designed to measure varying radiation continuously or as quickly as radiation monitors need. The conventional MCA is also expensive.

For these reasons, spectrum measurement has not been applied to radiation monitors. Only the count rates of the total spectrum have been measured until now.

We have developed a relatively inexpensive MCA and data reduction methods with a real time response to be used in radiation monitors.

REAL TIME SPECTRUM MEASUREMENT

Table 1 shows the specifications of the MCA.

The MCA contains an amplifier, a high voltage power supply, a temperature sensor and an optical interface for data transmission to the console. Detector bias voltage, amplifier gain and Δt are adjustable at the personal computer in the console.

Table 1. Specifications of the MCA.

Bias voltage supply	600 - 1,200 V
Amplifier gain	1.1 V/pc $\times$1,2,3
ADC conversion gain	512 ch
Counts per channel	$2^{16}-1$
Integral non-linearity	$\pm$ 0.5 %
Differential non-linearity	$\pm$ 2 %
Operating condition	0-45 ℃, 95 %RH

A specturm measurement is repeated at short intervals (Δt) in the MCA and, after each measurement, spectrum data are transmitted to the operation console via an optical fiber cable. The personal computer in the operation console processes Δt spectrum counts for each channel for longer time interval.

We applied two computing methods. Both methods have been utilized in digital count rate meters. One method is the "running average (RA) method". Each Δt spectrum datum transmitted from the preamplifier is stored in a rotating 10 stage memory stack. The most recent datum replaces the oldest cyclically. The total of all the counts for the same channel in the memory stack is updated each time a datum is replaced. Thus the spectrum is refreshed every Δt measurement. The counting time of the spectrum is $10 \times \Delta t$. Because the spectrum of the "RA" method is simply the mean value in $10 \times \Delta t$ interval, its response is delayed with fast radiation changes.

The other method is the "exponential smoothing (ES) method", which responds exponentially to radiation changes. It simulates the response of analogue RC rate meters by subtracting a fraction corresponding to the accumulated counts.

$$N_i = R_i^{-1} [N_{in} + (R_i - 1) N_{i(n-1)}], \quad (1)$$

$$\tau_i = (R_i - 0.5) \Delta t, \quad (2)$$

$$P = [2 \tau_i N_{i(n-1)}]^{-1/2} \times 100, \quad (3)$$

where

N_i = Counts of i-channel after refreshment;
R_i = Exponential smoothing coefficient;
N_{in} = Counts of i-channel transmitted;
$N_{i(n-1)}$ = Counts of i-channel before refreshment;
τ_i = Time constant of i-channel; and
P = Relative standard deviation which is common to whole channel.

The exponential smoothing coefficient R_i is derived from equations, (1),(2) and (3).

$$R_i = [2 N_{i(n-1)} \Delta t]^{-1} (100/P)^2 + 0.5 . \quad (4)$$

This method has been used in digital count rate meters. Although digital count rate meters have only one channel, there are i number of channels with spectrum monitoring. By setting the values of Δt and P beforehand, R_i is calculated with $N_{i(n-1)}$. Then, the count numbers of each channel are exponentially smoothed. The time constant for each channel changes according to the count for each channel. As the results, relative standard deviations of whole channel have the same value in the range of τ_i from 1 to 10 min. The range is limited from the the point of actual application to radiation monitors.

THE RESULTS OF MEASUREMENT

In the background radiation field, we put two radiation sources, ^{137}Cs and ^{60}Co, whose radiation intensities are less than 3.7 MBq, near to a 2"ϕ NaI(Tl) scintillator. After the spectrum reached equilibrium, we picked up both sources.

Figure 1 shows changes in the spectrum measured with the "RA" method with Δt=1 min. Figure 2 shows changes in the gross counts of the ^{137}Cs peak area.

The counts increase, then decrease according to the same ratio step.

Figure 3 shows changes in the spectrum measured with "ES" method when the relative standard deviation is set to 5 %. Figure 4 shows changes in the gross counts of the ^{137}Cs peak area when the standard deviation is set to 5%, 10% and 20%. It is recognized that each time constant varies according to standard deviations.

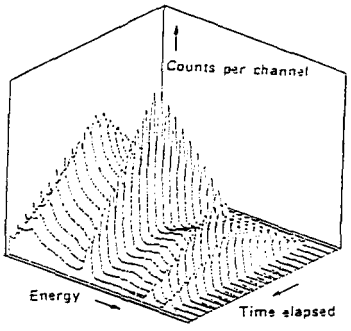

Figure 1. Changes in spectrum with running averaging.

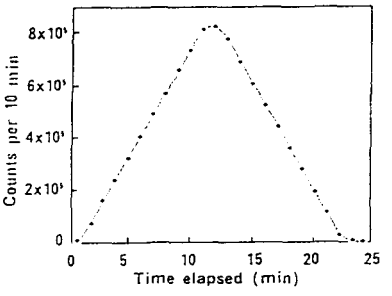

Figure 2. Changes in gross counts of ^{137}Cs with running averaging.

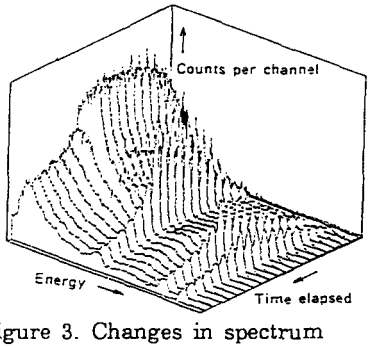

Figure 3. Changes in spectrum with exponential smoothing.

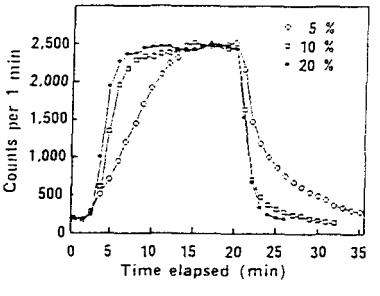

Figure 4. Changes in gross counts of ^{137}Cs with exponential smoothing.

APPLICATION TO RADIATION MONITORS

Real time spectrum measurement is applicable to area monitors, liquid effluent monitors and gaseous effluent monitors. NaI(Tl) scintillators are suitable for such radiation monitors. In liquid and gaseous effluent monitors, specific energy regions are set to determine the radioactive concentration of each radionuclide, which will be released from a radioisotope facility. The dose equivalent rate of an area monitor is calculated from the measured spectrum and energy weighting factors. Such an area monitor is more sensitive than one with an ionization chamber.

In routine radiation monitoring, only calculated radioactive concentrations and dose equivalent rates are stored in the personal computer. When those values exceed the preset levels, the spectrum at that time is also stored automatically.

SUMMARY

We have developed an MCA and two real time spectrum monitoring methods for use in radiation monitors. The "running average method" cannot keep up with radiation changes, but shows a precise spectrum. The "exponential smoothing method" has a fast response because of an automatically variable time constant.

The latter method is preferable for area monitors and gaseous effluent monitors.

MEASUREMENT OF THE AVERAGE PATH OF GAMMA-RAYS IN THE SAMPLE

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Abstract. It is shown that the valley-to-peak ratio in gamma-ray spectra bears information about the average path of gamma rays, thansversed in the sample. The measurements of the intrinsic valley-to-peak ratio, due to the response of the detector to gamma rays are presented. The average path in homogeneous sample, estimated from the valley-to peak ration is compared to the path, calculated from the derivative of the efficiency on the attenuation coefficient.

INTRODUCTION

In gamma-ray spectrometry the activity of gamma-ray emitting nuclides can be evaluated if the probability for the registration of gamma rays in the full energy peaks is known. This probability depends on the characteristics of the detector, the sample and their geometry. For homogeneous samples it can be measured by measuring calibrated sources [1], which resemble the samples in all characteristics. Therefore it is convenient to have a method to justify the supposition about the homogeneity of unknown samples. Hence, methods are sought to determine the inhomogeneity from the measured spectrum itself.

An indication for the presence of inhomogeneity can be obtained from the comparison of the measured average path of detected gamma rays in the sample with the calculated one under the supposition that the sample is homogeneous. When gamma rays pass the material they may by scattered for small angles. The corresponding energy loss is small and they register in the energy interval between the Compton continuum and the full energy peak. However, in this energy interval counts are present even if there is no material between the source and the detector, because of scattering on insensitive layers of the detector and since gamma rays may lose any amount of energy in the sensitive area before escaping. Therefore only the increase of the number of counts in the interval over its intrinsic value bears the information about the thickness of the transversed layer [2]. This information can be then used to assess the distribution of the emitter within the sample or to calculate self-absorption corrections.

METHOD

The relation between the number of gamma rays emitted from the vicinity of the point $\vec{r}$ which are registered in the "valley" between the Compton continuum and the full energy peak and the average path transversed in materials between the sample and the detector has two terms. The first describes the number of counts of the bare detector and the second the increase due to the scattering in the transversed material:

$$dN_v(\vec{r}) = K(\vec{r})\, dN_p(\vec{r}) + \eta'(\vec{r})\, P_1(\mu_a,\vec{r})\, dN(\vec{r}) \quad .$$

Here $dN_v(\vec{r})$ denotes the number of gamma rays emitted in the vicinity of the point $\vec{r}$ and registered in the valley, $dN_p(\vec{r})$ the number of counts registered in the peak, $K(\vec{r})$ the ratio of the number of counts in the valley and the number of counts in the peak for a source placed at the point $\vec{r}$ and in the absence of material between the source and the detector, $P_1(\mu_a,\vec{r})$ the probability for single scattering of gamma rays to the energy interval of the valley, $\eta'(\vec{r})$ the probability for detection of scattered rays with full energy and $dN(\vec{r})$ the number of rays, emitted from the point $\vec{r}$. The constant $K(\vec{r})$ represents an intrinsic property of the detector and must be measured in a similar way as the total-to-peak or peak-to-Compton ratios. The probability for single scattering is given by [2]:

$$P_1(\mu_a,\vec{r}) = K_1\,\mu_a\,\bar{d}(\vec{r})\,e^{-\mu_a\bar{d}(\vec{r})}\,P_\Omega(\vec{r}) \quad ,$$

where K_1 denotes the fraction of the total number of interacted rays, which are scattered into the energy interval of the valley, μ_a the total interaction coefficient, $\bar{d}(\vec{r})$ the average path of gamma rays through the material and $P_\Omega(\vec{r})$ the probability for transversing the space occupied by the scattering material. K_1 is given by the ratio of the integral of the Klein-Nishina cross section over the interval of angles, defined by the energy interval spawning the valley region, and the total interaction cross section. The number of gamma rays detected in the valley region is:

$$dN_v(\vec{r}) = K(\vec{r})\,dN_p(\vec{r}) + K_1'\,\mu_a\bar{d}(\vec{r})\,dN_p(\vec{r}) \quad ,$$

where we have used the relation $dN_p(\vec{r}) = \eta(\vec{r})dN$ and $K_1' = K_1\eta'(\vec{r})e^{-\mu\bar{d}}P_\Omega(\vec{r})/\eta(\vec{r}) = K_1\eta'(\vec{r})P_\Omega(\vec{r})/\eta_o(\vec{r})$. In the last expression $\eta_0(\vec{r})$ denotes the efficiency in the absence of the absorbing medium. It can be shown [3], that the dependencies on $\vec{r}$ in the expression on K_1' partially cancel out. The rest can be accommodated to the average path $\bar{d}(\vec{r})$ by its redefinition, so that the valley-to-peak ratio for a volume sample is obtained by integration of the previous equation over the sample volume and division by N_p:

$$\frac{N_v}{N_p} = \frac{\int_{V_s} K(\vec{r})\,dN_p(\vec{r})}{N_p} + K_1'\,\mu_a\,\frac{\int_{V_s}\bar{d}(\vec{r})\,dN_p(\vec{r})}{N_p} \quad .$$

Since the last factor in the last term represents just the average path in the sample of the scattered and detected rays, it can be expressed as:

$$\bar{d} = \frac{\dfrac{N_v}{N_p} - \dfrac{\int_{V_s} K(\vec{r})\,dN_p(\vec{r})}{N_p}}{K_1'\,\mu_a} \quad .$$

MEASUREMENTS

The valley-to-peak ratio of the detector $K(\vec{r})$ has been measured for a p-type detector of 35% relative efficiency at energies of 320, 661, 835 and 1115 keV on a grid of points within 5 cm in the radial direction and 4 cm in the axial direction from the centre of detector cap with the interval of 1 cm. In order to avoid the contribution of counts from the low-energy tail of the full energy peak the upper limit of the valley region was set to 25 keV below the energy of the gamma rays. The energy interval of the valley region is presented in Fig. 1. The measured energy dependence of $K(\vec{r})$ in the detector axis on the detector surface is presented in Fig 2.

The constant K_1' was measured with a parallel beam of rays, scattered on a layer of material of known thickness. To measure the average path of gamma rays in the sample four sources with aqueous solution of the ^{137}Cs isotope were measured. They were of cylindrical shape with dimensions of Φ60x10, Φ60x36, Φ90x16 and Φ90x39 mm. The comparison between the average path in the sample of the detected rays, calculated from the derivation of the volume-source efficiency on the absorption

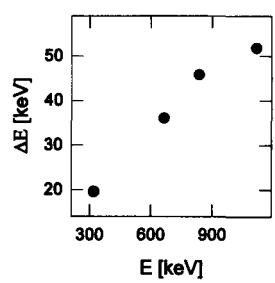

Figure 1. The energy dependence of the valley region.

coefficient [4], and the average path in the sample of scattered and detected rays is presented in Table 1. It may be observed that the path of scattered and detected rays is larger. This is due to the scattering angle, which permits the detection of rays which would not pass the sensitive volume of the detector.

DISCUSSION AND CONCLUSION

The valley-to-peak method was developed for use in in-situ gamma-ray spectrometry, to measure the average depth of the radionuclides deposited in surface layers of the soil. It has been shown that also in laboratory conditions it yields useful results. In laboratory conditions the difference between the paths of scattered and unscattered rays becomes measurable. This difference, since is due to the scattering angle, falls off with increasing energy of gamma rays. The comparison between both average paths offers the opportunity to check the supposition about the homogeneity of the sample on one hand and presents an independent check for the accuracy of the calculation of the influence of self-attenuation on the efficiency on the other. It should be noted also, that in laboratory conditions the spatial dependence $K(\vec{r})$ which reflects the variation of the detector response on the position of the source must be taken into account. Consequently, in measurement of volume samples the integration in Eq. (1) has to be carried out, in order to obtain unbiased results.

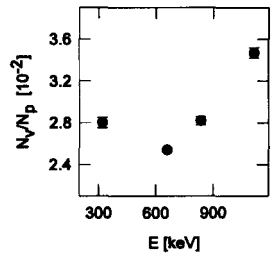

Figure 2. The energy dependence of the valley-to-peak ratio for a point source located on the detector surface on its axis.

TABLE 1. Comparison between the average paths in the sample of the detected and the scattered and detected gamma rays.

Geometry [mm]	Detected [mm]	Sc. & detected [mm]
60 x 10	5.4	4.9 ± 0.9
60 x 36	15.0	17.2 ± 0.9
90 x 16	10.0	11.2 ± 1.0
90 x 39	17.3	20.2 ± 0.7

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE

Development of radioactivity investigations in the environment using portable devices

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MAJOR SCIENTIFIC TOPIC NUMBER 4.1. (see page 7)

ABSTRACT (See instructions overleaf)

Today, the knowledge of the radioactivity level in a given environment is one of the essential questions that seems concerned all citizens. In the Université de Franche Comté, to reply for the different investigations in situ that we have been asking by institutional and academic institutions, industrial person or simple citizen, we have developed or adapted different techniques. The whole of these instruments is autonomous and portable. They localised in a specially mobile laboratory to realise measures rapidly. The available measure instruments are : - a proportional counter, specially developed to measure the radon concentration directly in every atmosphere, soil and aquatic point, - an alpha spectrometer of terrain using a silicon detector adapted to perform alpha emitter measurements on all surfaces and with the possibility to analyse the alpha emitters fixed on aerosols and collected by a filtering membrane, - a portable gamma spectrometer (8192 channel) using NaI (Tl) detector for monitoring gamma ray emission s from natural (potassium, uranium, thorium,...) or artificial (caesium, ...) radionuclides and - a device to analyse the concentration in ozone that is proportional to the rate of ionisation produced on the site of measure. All devices connected to a portable micro-computer that centralises the results and gives the cartography of the radioactivity level for the studied environment.
Different fieldwork's have shown the quality and the efficiency of implementation systems.

THE BGO ALPHA-BETA-GAMMA SCINTILLATION SPECTROMETER*

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Introduction

The NaI(Tl) alpha-beta-gamma (ALBEGA) scintillation spectrometer was invented by Kalish in 1972 [1,2]. It was introduced at an international conference in March 1973 [3] and further described at other conferences [4-10]. The first manual NaI(Tl) ALBEGA spectrometer was developed and constructed by Kalish in cooperation with Beta Research & Development Co. (Israel) during 1972-1974 and a prototype of an automatic version was displayed by Kalish and Beta Research & Development at the 5th International Conference on Medical Physics in August 1979 in Jerusalem.

The core of the NaI(Tl) ALBEGA scintillation spectrometer comprises a cylindrical NaI(Tl) crystal coupled to two photomultiplier tubes. At the center of the crystal, perpendicular to its major axis, is a cylindrical cavity covered with quartz. When a sample containing a beta- or alpha- emitting isotope is to be measured, it is dissolved in a liquid scintillator, and the vial containing the radioactive liquid scintillator is loaded into the cavity. The light emitted from the liquid scintillator when a beta or alpha particle interacts with it is piped through the NaI(Tl) crystal to the two photomultiplier tubes. Only when the pulses from both tubes coincide and only when the pulses are fast, as determined by a pulse-shape analyzer, is the signal stored in a multi-channel analyzer. The NaI(Tl) crystal serves not only as a light pipe but also as an anti-coincidence shield for reducing background counting rate. When a gamma, or cosmic, ray interacts with both the liquid scintillator and the NaI(Tl) crystal, the "false" pulse is not counted; such background pulses are rejected by the pulse-shape analyzer. The light emitted from the liquid scintillator and the NaI(Tl) crystal reaches the same photomultiplier tubes. However, the shape of a pulse originating in the organic liquid scintillator differs from that originating in the inorganic NaI(Tl) crystal: whereas the decay constant of a light pulse from the liquid scintillator is about 2 ns, the decay constant of a light pulse from the NaI(Tl) crystal is about 230 ns. By using a pulse-shape analyzer it is possible to ascertain the origin of the pulse.

When a gamma emitter is to be measured the NaI(Tl) crystal acts as a conventional gamma scintillation detector.

The BGO ALBEGA Spectrometer

The NaI(Tl) ALBEGA scintillation spectrometer is a universal spectrometer with which any kind of radioactive isotope can be measured. However, it is unsuitable for the measurement of samples containing low activities of tritium. With conventional liquid scintillation spectrometers the background in the tritium window is about 20 counts per minute (cpm), whereas with a NaI(Tl) ALBEGA scintillation spectrometer it is over 100 cpm [6]. This is a rather serious disadvantage as tritium is widely used, inter alia, in biological laboratories, being the only radioactive tracer of

hydrogen. Noakes and Spaulding [11], who experimented with a similar NaI(Tl) ALBEGA spectrometer, also got high background in the tritium window and could not explain its origin. Kalish suggested that the high background was caused by afterglow from the NaI(Tl) crystal and that the best way to solve the problem was to use BGO crystals instead [12-15]. While the afterglow from a NaI(Tl) crystal 3 ms after irradiation is 0.5-5%, the afterglow from a BGO crystal 3 ms after irradiation is only 0.005% [16]. In 1992, Kalish convinced R. J. Valenta (Packard Instrument Co.) to test the performance of an ALBEGA scintillation spectrometer using a BGO crystal. It was found that the BGO crystal did indeed solve the problem of high background in the tritium window extremely well. The figure of merit values for C-14 and tritium - as compared with those obtained with a conventional liquid scintillation spectrometer - showed dramatic improvement, from 1595 to 6300 (295%) for C-14, and from 544 to 1030 (89%) for tritium. In the measurement of a C-14 3.5 ml benzene sample in a 7 ml glass vial with the BGO ALBEGA system the background is very low, only 0.227 cpm and the figure of merit is 18,624.

Kalish now suggests a way to render the BGO ALBEGA spectrometer suitable for ultralow-level counting of radioisotopes that coincidently emit beta-gamma (e.g., I-129), beta-gamma-gamma (e.g. Co-60 and positron emitters), alpha-gamma (e.g., U-235), or alpha-gamma-gamma rays (e.g., Am-241). With this procedure, a sample containing a radioisotope that simultaneously emits a beta particle and a gamma photon is dissolved in a liquid scintillator. The beta particle interacts with the liquid scintillator and the gamma photon interacts with the BGO crystal. Each of the output pulses of the photomultiplier tubes is comprised of two components: a fast one, caused by the interaction with the liquid scintillator; and a slow one, caused by the interaction with the BGO. By analyzing the shape of the pulse with a suitable pulse-shape analyzer, the sizes of the fast and slow components may be determined. Only when these components are of the right size, is a pulse recorded. This, in a way, is a beta-gamma coincidence measurement. The probability for a background event to interact with both the liquid scintillator and the BGO such that the energy absorbed in the liquid scintillator would be within the right beta window and simultaneously the energy absorbed in the BGO would be within the right gamma window is expected to be very low. The same procedure may be used with alpha-gamma emitters. With a positron emitter, for example, the pulse output of each photomultiplier tube is resolved into its fast and slow components. The two fast components are added, and only when this summed pulse is within the beta window and is in triple coincidence with 0.51 MeV slow-component pulses from gamma interactions with the BGO, is a signal recorded. Again, the probability of such triple coincidence as a result of a background interaction is extremely low. Background levels with these procedures are expected to be less than 1 count per day.

Summary

The BGO ALBEGA spectrometer can measure samples of alpha-beta- and gamma-emitting radioisotopes. Gamma-emitting samples can be measured with or without a liquid scintillator. The figure of merit values for tritium and C-14 are superior to those of conventional liquid scintillation spectrometers. The author expects that BGO ALBEGA spectrometers will in future replace conventional gamma and liquid scintillation spectrometers.

* Patent pending

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NEW THREE-COUNT TECHNIQUE FOR SHORT-LIVED RADON DECAY PRODUCTS IN AIR

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ABSTRACT

Up to the present, radon and its short-lived decay products in air are usually monitored by means of α detection. But radon progenies, including RaB (^{214}Pb) and RaC (^{214}Bi) which are β and γ emitters, contribute about 90 % to the equilibrium equivalent radon concentration (EECRn). Therefore, this paper introduces a new three-count technique by a β detector in the light of radioactive decay law and its boundary conditions during sampling and counting times to solve the Bateman equation. β (even low level β) instruments have been fairly popularized domestically and internationally. It can be used not only as an instrument for radon and its daughters in air, but also as a monitor for β airborne activity in the environment. This new method taps further the latent power of the present instrument and realizes various uses for a unit.

Keywords : radon, three-count technique, equilibrium equivalent radon concentration (EECRn), thoron

INTRODUCTION

It is well known that natural radon and its short-lived decay products in airborne state may cause serious radiation harm to human health such as inducing lung cancer to those who were exposed to them for a long period. Therefore, detailed studies for how to monitor them were conducted. Today, along with the improvements in science and technology, the monitoring procedures for radon progenies have made an encouraging progress. Tsivoglou et al.[1] proposed the Tsivoglou method (a three-point method) for monitoring radon daughters in air on the basis of the count-ratemeter in 1950s. Thomas[2] then improved the three-count technique (or modified Tsivoglou method) on the foundation of the scaler in 1970s. In addition, there are some rapid methods for monitoring the radon progenies in air[3], which are all by means of α-particle detection. However, β and γ rays are emitted in the decay process of radon daughters and reports concerning the detection method of them were not seen so far.

PRINCIPLE

Radon and its short-lived decay products emit α, β and γ rays during the decay process. Their main decay diagram is shown in Fig. 1. Radon (^{222}Rn) decays to form ^{218}Po (RaA), ^{214}Pb (RaB), ^{214}Bi (RaC), ^{214}Po (RaC'), ^{210}Pb(RaD), etc., consecutively. The former four decay products are called radon short-lived daughters because their half-lives are less than 30 min. RaD and the subsequent nuclides have much longer half-lives, for instance 22.3 y for RaD. They are quite stable as compared with the former ones. For this reason, it can be regarded as if RaD and the latter are not to decay, and the multi-progeny decay for radon and its daughters may be simplified to a four-progeny decay. On the other hand, the half-life of nuclide RaC' is very short, only 164 μs. The radioactivity equilibrium between RaC and RaC' appears right away. It can be said that RaC is simultaneously emits one β- and one α-particles and decays immediately to RaD, not through RaC'. Hence, the four-progeny decay is further simplified to a three-progeny decay.

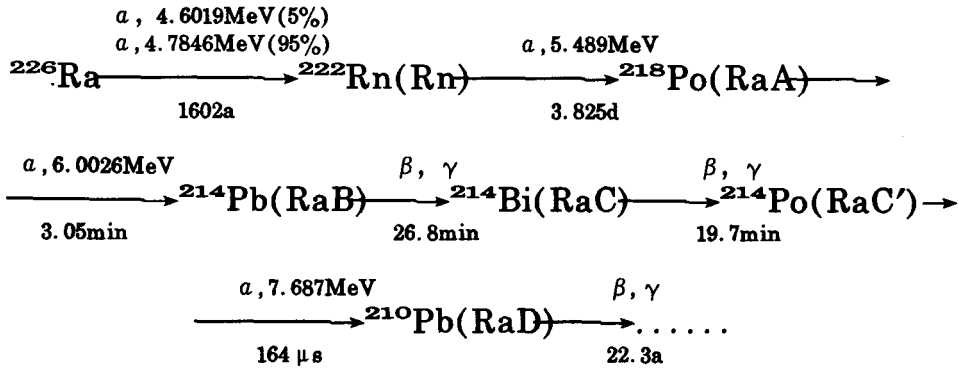

Fig. 1 Main decay diagram of radon and its short-lived decay products

Radon short-lived progenies emit two α-particles in all. For this reason, radon daughter monitoring in air is almost by means of α-particle detection because of its high α-energy and α-pulse signal amplitude, very low background, and the simple structure and easy manufacture of α-detector. On the other hand, the equilibrium equivalent radon concentration (EECRn) is given by:[4]

$$EECRn = 0.105\ C_{RaA} + 0.516\ C_{RaB} + 0.379\ C_{RaC}\ Bq/m^3 \quad (1)$$

where C_{RaA}, C_{RaB} and C_{RaC} are the concentrations of radon progenies, RaA, RaB and RaC, respectively. If the three radon daughter concentrations in air are known, EECRn can be obtained immediately from Eq. (1), and their dose to the public can be evaluated. The RaC′ concentration, $C_{RaC'}$, does not contribute to EECRn as seen from Eq. (1).

Two radon progenies, RaB and RaC, which contribute about 90 % to EECRn are not α-emitter, but are both β and γ emitters. Hence, it is unsuitable to say that α-particle counting for detecting RaA and RaC′ is the best choice because radiation doses are caused mainly by α-radiation. On the contrary, β or γ counting is an optimal selection.

Based on the radioactive decay law and relevant boundary conditions during sampling and counting periods, the Bateman equations were solved and got integral gross β counting which includes three unknowns hiddenly, namely the RaA, RaB and RaC concentrations in air. That is a ternary first-order equation. It is well known that only a set of three simultaneous equations can be solved to get a sole answer when there are three unknowns in it. For this reason, three equations with three observed values are required. Three activity measurements on the filter (i.e., three countings) are to be conducted in the experiment for three different time intervals after the end of sampling. Hence, this method is called the three-count technique. A set of simultaneous linear equations obtained from three countings was set up and solved by the inverse matrix method. The formulas for radon progeny concentration in the air to be measured by the gross β coun-ting method are derived via a computer. According to the UNSCEAR definition[5], the formula of Potential Alpha Energy Concentration (PAEC) is led in. The formula coefficients are a series of complicated exponential operations which are functions of sampling and counting times. A computer program was written to calculate all coefficients rapidly so long as each time parameter inputs are given. These coefficients are simplified constants and do not need to be computed every time for a certain procedure with fixed time parameters.

However, it is possible not to start and/or stop the countings on time in actual situations for a variety of reasons. The data will become invalid at the moment. It is well known that the radon and its daughter concentrations in air change with time and place. Reappeared them is impossible yet even though they are determined again. Therefore,

the above-mentioned program is onlined to realize the automatic measurements. Output data from various procedures can be typed out immediately in the light of both relative parameters and measured gross β countings.

RESULTS AND DISCUSSION

To compare conveniently with the Thomas three-count technique[2], the same procedures as his were used , i.e., 5 min sampling time and 2~5, 6~20 and 21~30 min countings after the end of sampling to obtain the three gross β countings, N_1, N_2 and N_3, respectively. If the total β counting efficiency G (including the instrument and filter efficiencies, self-absorption and backscattering corrections on the filter, etc.), sample flow rate V L/min and background counting rate N_o cpm (counts per minute) are known, the formulas for the radon progeny concentrations, PAEC and EECRn are:

$$C_{RaA} = (-60.9933N_1 + 29.6053N_2 - 27.9909N_3 + 20.4233N_o) / GV \ Bq/m^3 \quad (2)$$

$$C_{RaB} = (+8.50301N_1 - 4.71163N_2 + 5.32176N_3 - 7.44207N_o) / GV \ Bq/m^3 \quad (3)$$

$$C_{RaC} = (-3.08556N_1 + 2.90511N_2 - 3.84836N_3 + 3.22043N_o) / GV \ Bq/m^3 \quad (4)$$

$$PAEC = (-175.253N_1 + 97.9325N_2 - 90.9556N_3 - 26.6961N_o) / GV \ 10^{-10} \ J/m^3 \quad (5)$$

$$EECRn = (-3.18617N_1 + 1.77840N_2 - 1.65155N_3 - .475117N_o) / GV \ Bq/m^3 \quad (6)$$

Generally, the flow rate V, efficiency G and background rate N_o are determined previously, and their errors can be managed to be reduced as much as possible or neglected. Statistical counting errors can not be decreased and are dominant in the environmental level. The resultant concentration uncertainties are shown below according to the error resultant rule when only statistics are considered:

$$S_{RaA} = \sqrt{3720.19N_1 + 876.476N_2 + 783.490N_3 + 268734N_o} / GV \ Bq/m^3 \quad (7)$$

$$S_{RaB} = \sqrt{72.3011N_1 + 22.1994N_2 + 28.3211N_3 + 7295.81N_o} / GV \ Bq/m^3 \quad (8)$$

$$S_{RaC} = \sqrt{9.52067N_1 + 8.43965N_2 + 14.8099N_3 + 2939.46N_o} / GV \ Bq/m^3 \quad (9)$$

$$S_{PAEC} = \sqrt{30713.5N_1 + 9590.78N_2 + 8272.93N_3 + 2826321N_o} / GV \ 10^{-10} \ J/m^3 \quad (10)$$

$$S_{EECRn} = \sqrt{10.1517N_1 + 3.16270N_2 + 2.72761N_3 + 932.190N_o} / GV \ Bq/m^3 \quad (11)$$

where S_{RaA}, S_{RaB}, S_{RaC}, S_{PAEC} and S_{EECRn} are the resultant uncertainties for the RaA, RaB and RaC concentrations, PAEC and EECRn in the air, respectively.

The experiments showed that the results of this new three-count technique agree well with that of the Thomas' method.

SUMMARY

Almost all monitors and dosimeters commercially available now for radon and its progenies in the air employ generally the method of detecting α-particle. Their efficiencies are low and detector areas are small normally although the method is good. It means these instruments are well suited for integrated or continuous long time measurements, but not for the rapid and sensitive grab ones at low levels in the environment. This paper develops for the first time a new three-count technique by gross β countings to determine the radon daughters in the air. The formula for calculating EECRn is led firstly in so as to realize the combination monitoring of radon and its progenies. This sdudy fully utilizes the high efficiency and large area of the β detector, and is a new pioneering work in radon and its daughter monitoring in the air. The new method taps further the latent power of these β instruments and realizes the various uses for a unit.

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COMPARISON OF LABORATORY AND IN SITU EVALUATION OF ENVIRONMENTAL TL DOSIMETERS

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INTRODUCTION

The passive environmental gamma-radiation dosimetry is mainly based on TL (thermoluminescent) dosimetry. This method offers considerable advantages due to its high precision, low cost, wide range, etc. (1). At the same time its application involves uncertainty caused by the dose collected during the transport from the point of annealing to the place of exposure and back to the place of evaluation. Should an accident occur read-out is delayed due to the need to transport to a laboratory equipped with a TLD reader.

A portable reader capable of reading out the TL dosimeter at the place of exposure (in situ TLD reader) eliminates the disadvantages mentioned above. A microprocessor based portable TLD reader was developed (2) by us for monitoring environmental gamma-radiation doses.

Using a portable reader for in situ evaluation there are several disadvantages as well. The method requires the transport of the reader instead of dosimeters. The portable reader should be battery operated with low power consumption. Due to this requirement the temperature stabilization of the reader requests different solution as in laboratory type devices.

The lowest power consumption can be achieved by bulb type dosimeters. The bulbs have high reproducibility due to constant heat contact. Construction of bulbs makes use of usual temperature regulation system practically impossible. Lack of temperature stabilization of the light detection system and the heating plate leads to some difficulty in use of portable, battery operated readers.

Both investment and running costs of environmental monitoring are different for laboratory and in situ readout systems.

Comparison of recently developed in situ and traditional laboratory evaluation methods of environmental TL dosimeters is given in recent paper. The comparison was made in the same conditions. The most characteristic - for environmental monitoring - numerical TL data (dose range, reproducibility, fading, self dose etc.) are given for manufactured by us $CaSO_4$:Dy bulbs (portable reader) and very advantageous, high sensitive Al_2O_3:C dosimeters (3) (laboratory evaluation).

PORTABLE TLD SYSTEM

At the beginning of the 80's the KFKI laboratory produced a great number of portable, battery operated TLD readers (named Pille - "Butterfly") (4). These devices used $CaSO_4$ bulb dosimeters and the evaluation technique was based on analogue timing circuits and analogue to digital conversion of the photomultiplier current with a read-out precision of 1 μGy. The measured values were displayed and manually recorded. The version with an external power supply was used for space dosimetry as an onboard TLD reader (5).

Based on a microprocessor an up-to-date version of the battery operated portable reader was developed in 1994-95 at the Atomic Energy Research Institute. The new system consists of a set of TL bulb dosimeters with built-in memory chips and the microprocessor based reader with main characteristics as follows:

- The $CaSO_4$:Dy bulb dosimeter is the same as the earlier one (5).
- The TLD bulb has a common case with the memory chip containing the identification number of the bulb. The heating supply and the high voltage supply are controlled via digital-analogue converters by the microprocessor thereby providing a possibility to program the time dependence both of the heating current and of the high voltage.
- The light output of the bulb dosimeter is measured by a photomultiplier, a wide range I/U converter, and a digital voltmeter. (The range of the light detecting system exceeds 8 orders of magnitude.) The built in, stabilized LED light source controls the light sensitivity of the reader in each measuring cycle.
- The four digit alphanumeric LED display indicates the measured dose in exponential form, the possible error codes and the menu/submenu points of the setup. The removable memory card can store up to 4000 measured data sets (dose, identification number, date and time, and digital glow curve).
- The mass of the battery operated version of the reader is about 2.2 kg, its dimensions are 190 x 155 x 70 mm. The rechargeable battery (9.6 V) provides capacity for about 200 read-outs. The reader also works from a 12 V or 24 V car battery in buffer mode.

The environmental temperature influences both the position and the area of the glow curve. Using the built in digital thermometer this effect can be taken into account.

The gamma-radiation energy dependence is relatively small due to 0.7 mm thick Sn filter to cut the well known oversensitivity of $CaSO_4$ below 200 keV. The angular dependence is significant only in case of irradiation along the axis of the bulb. The decreased response in environmental exposure in comparison with standard calibration can be taken into account using a correction factor.

The self irradiation of the bulb dosimeters is caused by glass envelope. For our case (glass with low potassium content) it was found equal to 5 ± 2 nGy/h.

Fading of the $CaSO_4$:Dy at usual environmental temperatures is negligible.

Environmental doses can be measured using in situ read-out of the dosimeters. The wide measuring range of the system (3 μGy - 10 Gy) provides the possibility to measure dose values from natural background up to very large accidental doses. The measuring error (σ) of individually calibrated dosimeters does not exceed $\pm 2.5\%$ above 50 μGy (typical monthly environment dose rate) at room temperatures. Having a temperature range from -10°C to +40°C for the reader the temperature dependence after corrections is below $\pm 2\%$. It means that the overall uncertainty (σ) is below $\pm 5\%$.

LABORATORY EVALUATION USING AL_2O_3 DOSIMETERS

Measurements on high sensitive Al_2O_3:C single crystal dosimeters (diam. 5 x 0.9 mm) were carried out on Harshaw 2000 AB TLD reader at the heating rate of 10°C/s. The dosimeters were linearly heated up to 300°C, the dosimetry peak (190°C) was integrated within the interval of 120°C to 250°C.

All measured data were calculated as the average of 5 dosimeters reading. During irradiation and handling the dosimeters were kept in dark polyethylene foil to exclude undesirable light exposure (6).

Self dose measurements and room temperature fading investigations were performed at a low background place, in an iron shield, having 20 cm thick walls. The dose rate within the shield was 20 nGy/h measured with high pressure ionization chamber.

For TL measurements the dosimeters were exposed using a calibrated ^{90}Sr-^{90}Y source.

One of the important parameters of the Al_2O_3:C is its 40 times higher TL sensitivity compared with common TLD-100 dosimeter.

The batch inhomogeneity (σ) of 20 pieces of dosimeters does not exceed 15% and the reproducibility of a single dosimeter on the basis of repeated (5 times) irradiation (1.75 mGy) and evaluation was found within 5 %.

The lowest detectable dose is 100 nGy.

Self dose effect on the dose responses of dosimeters could not be demonstrated by measurements of five months.

With storing the dosimeters in the iron shield place at room temperature for four weeks the fading was negligible.

Results obtained suggest that the Al_2O_3:C dosimeters are promising for environmental monitoring due to their high sensitivity, good reproducibility, low fading and easy handling.

COMPARISON OF LABORATORY AND IN SITU EVALUATION

Main characteristics of laboratory and in situ evaluation are given for Al_2O_3:C and $CaSO_4$:Dy bulb dosimeters are given in table.

Characteristics	Laboratory evaluation (Al_2O_3 dosimeter)	In situ evaluation ($CaSO_4$:Dy bulb)
Minimum detectable dose (3σ of the background), μGy	0.1	1
Reproducibility	$\pm$5%	$\pm$2.5%
Self dose, nGy/h	negligible	5$\pm$2
Environmental temperature effect	negligible	$\pm$2% with correction
Transport dose effect	may be significant	excluded
Time delay of read-out	significant	short
Investment cost, reader, USD	from 20 000	about 10 000
Investment cost, dosimeter, USD	3-5/piece	30-50/piece
Cost of readout/piece	low	medium
Recommended field of application	wide (national) scale monitoring	local scale monitoring around nuclear installations

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NEUTRON SPECTROMETRY FOR PROTECTION DOSIMETRY AT VERY LOW LEVELS

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INTRODUCTION

Dose limits for exposure of members of the public are significantly lower than those for designated radiation workers. The new ICRP 60 (1) recommendation for critical groups of the general public is an effective dose limit of 1 mSv per year which requires a measurement capability at levels down to about 100 nSv h^{-1}. Radiation protection dosimetry for neutrons at these levels is problematical, nevertheless, operators of nuclear sites are still required to demonstrate acceptably low radiation levels in areas accessible to the public.

In addition to the known poor dose equivalent response of available dosemeters, there is an added problem at low levels of inadequate sensitivity. Personal dosemeters are certainly not sufficiently sensitive, and the sensitivity of area survey instruments is such that they can only be used in integral mode. Even then, the statistical uncertainties are likely to be large. One further problem concerns the quantity measured. Survey instruments are designed to measure the operational quantity ambient dose equivalent, H*(10), which always tends to be an overestimate of the present limiting quantity effective dose equivalent (2). If in any situation exposure is near the limit, an estimate of H*(10) may not be sufficient to prove conclusively that levels are less than the statutory limit, and a direct estimate of effective dose equivalent may need to be made.

The only way of estimating effective dose equivalent is via an absolute spectral measurement. From such a spectrum any relevant dosimetric quantity can be estimated via tabulated fluence to dose equivalent conversion factors. (Certain quantities also require information about the angular dependence of the field - see later text). Spectrometry at such low neutron fluence levels is difficult, however, there is one instrument available which can perform the required measurements, and this is a well characterised Bonner sphere (BS) set.

MEASUREMENT SITES

The general public are very rarely exposed to neutrons, other than those from cosmic rays, however, neutron levels can occasionally be detected in the vicinity of nuclear power plants. The measurements described in this paper were performed at various points at distances around a reactor building which ranged from about 61 m to nearly 200 m from the building. All the measurements were performed out of doors although rudimentary temporary shelters were available for most of the sites and for the electronics which was centrally situated at a position of up to 100 m from the various measurement positions. At the site furthest from the reactor not even these facilities were available and measurements were made in the open air with the electronics located in a small van and powered by a portable generator.

SPECTROMETRY SYSTEM

The BS spectrometry set used for the present measurements consisted of eight spheres of high density polyethylene of diameters 3", 3½, 4", 5", 6", 8", 10", and 12"[†]. The central counter was a spherical ^{3}He proportional counter of diameter 32 mm (type SP9 manufactured by Centronic Ltd). In addition to using the eight spheres, measurements were also made with the thermal neutron counter, both bare and under a 1 mm thick cadmium cover, to provide a direct determination of the thermal neutron component of the field.

The derivation of a neutron energy spectrum from a set of Bonner sphere measurements is possible because of the different degree of moderation in each of the spheres. Each 'sphere plus thermal counter' combination has a different sensitivity to the various parts of the neutron energy spectrum. Small spheres are most sensitive to the low-energy part of the spectrum, whereas large spheres are most sensitive to higher energy neutrons. Provided the response functions, i.e. the response as a function of neutron energy, of all the spheres are known, the neutron spectrum can be unfolded from the count rates of the spheres.

[†] Bonner spheres are conventionally made to exact inch sizes, and the diameter in inches remains the accepted 'label' for each sphere.

Considerable effort has gone into the characterisation of the NPL BS set. The determination of the response functions for the spheres involved a series of measurements, with monoenergetic neutrons from 70 keV to 14.7 MeV (3,4), and also with thermal neutrons (5). These measurements are all traceable to national standards. In addition, two sets of calculations, one with the discrete ordinates transport code ANISN, and one with the Monte Carlo transport code MCNP, have been used. These data have all been combined to provide an optimised set of response functions.

Neutron spectra are unfolded from measured sphere count rates using the least-squares program STAY'SL. The difficulties of Bonner sphere unfolding have been much discussed in the literature (see for example reference 6). The problem is mathematically under-determined, however, STAY'SL has produced good results in spectrometry comparisons (7), and in unfolding code comparisons (6).

MEASUREMENTS

To measure the spectra, all spheres of the set were placed in turn at the measurement sites, and the count rates recorded for an appropriate length of time. The aim was always to count for long enough to obtain better than 1% statistical accuracy in the count rate; and this was achieved for the majority of spheres with the exception of the furthest distance from the reactor where the statistical uncertainties were higher, and no measurement was possible with the 12" sphere.

Because of the large distance from the source of neutrons the measurements were sensitive to changes in air scatter and attenuation caused by changes in atmospheric conditions such as temperature, and pressure. For this reason a monitor, consisting of a 5" BS, was positioned at a distance of about 150 m from the reactor and readings were corrected by the ratio of the mean monitor count rate over the total measurement period to the monitor count rate over the period of a particular sphere measurement. (The measurements covered a period of five days). Although the corrections were not large, the maximum being 3.8%, they were necessary because of the high sensitivity of BS unfolding to the ratios of the sphere count rates.

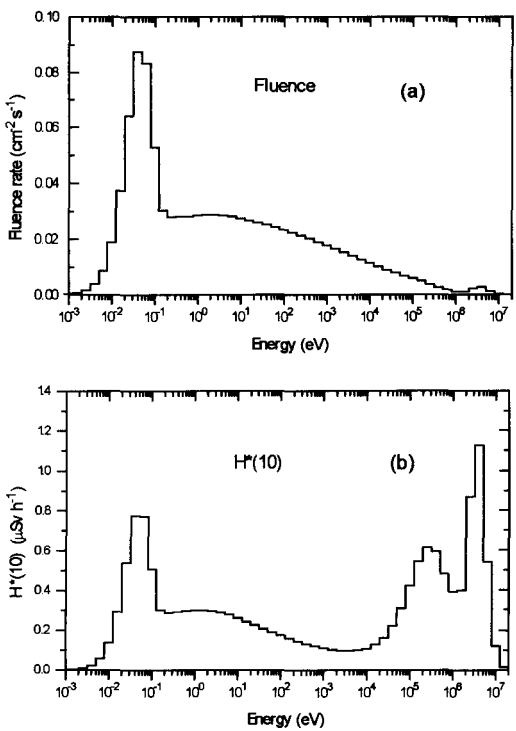

Figure 1. Average of the six measurement sites (a) fluence spectrum, (b) ambient dose equivalent spectrum.

RESULTS

Measurements were performed of the spectra at six sites at several distances in the range from 61 m to 191 m from the reactor building. The spectra at all sites were very similar, and an average of the six measurements, normalized to unit fluence rate, is shown in Figure 1 along with the corresponding ambient dose equivalent spectrum. (conversion factors were obtained from Wagner et al. (8)). The fluence is predominantly in the low and intermediate energy region with a substantial thermal peak, an intermediate energy distribution which decreases with increasing neutron energy, and a small component of fast neutron with energy greater than 1 MeV. When converted to H*(10), as shown in Figure 1 (b), the importance of the contribution from the small high energy component becomes apparent.

Values for effective dose equivalent for the six sites were calculated and these ranged from about 155 nSv h^{-1} to 16 nSv h^{-1}. These were for A-P irradiation of an individual. For all other angular dependencies of the fluence the effective dose equivalent is less and so these values can be considered as upper limits for the limiting quantity.

DISCUSSION

The present set of measurements were performed around a nuclear reactor, mainly within the site boundary where access was not possible for the general public. They did, however, include measurements outside the site boundary fence. These proved that measurements can be made with this type of BS system at extremely low levels well below the statutory limit for critical groups of the general public.

For the types of low energy spectrum seen, the quantity H*(10) over-estimated the limiting quantity effective dose equivalent by about a factor of about two for the assumption of A-P irradiation. Many area survey instruments overestimate H*(10) because of they over-read for intermediate energy neutrons. The net effect is thus an even greater overestimation of the effective dose equivalent. For the Harwell neutron dose equivalent monitor type 0949 the overestimation of H*(10) was approximately a factor of two resulting in a overall overestimation of the limiting quantity by a factor of about four. The Studsvik 2202D instrument did not significantly over-respond for H*(10) indicating that overestimation by this instrument in the intermediate energy region is compensated by under-reading in other regions. The spectrometry measurements give direct calibration factors for these instruments allowing them to be used with greater confidence in this particular environment in future.

The new limiting quantity recommended in ICRP 60, effective dose, was on average a factor of 2.3 higher than effective dose equivalent for these fields giving an even greater significance to the problems of the low sensitivity of neutron dosimetric instruments.

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BACKGROUND RADIATION SPECTRUM AND ITS INFLUENCE ON LOW LEVEL ACTIVITY MEASUREMENTS IN THE ENVIRONMENTAL SAMPLES

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ABSTRACT

The paper discusses the variations in the background gamma radiation spectrum as an important parameter in low level activity measurements. The background radiation spectrum has been recored from 1993 to 1995 on two HPGe detectors under equival environmental conditions but placed in differently constructed Pb cages. The results indicate that when one estimates low level activities of natural radionuclides in the environmental samples, background radiation can be a significant source of error, due to unadequate shielding or long counting time intervals. Special attention should be paid to the seasonal variations of the lines of Bi-214 and Pb-214, while errors due to intensity variations in K-40, U-235 and Th-232 lines in the spectrum are within the total eror of the method. Errors in intensity estimations in low and high energy range of the background spectrum should not be neglected , too.

INTRODUCTION

Background radiation spectrum is one of the basic Quality Assurance and Quality Control parameters in the evaluation of natural and man made radionuclides' activities in the environmental samples, especially when estimating low level activities of natural radionuclides. Intensity and stability of the spectrum are due to the technical characteristics of the detector itself, materials and design of the shielding, environmental conditions (ventilation ecc), construction of "low level chamber",ecc (1,2). Regular control of background radiation can point to the possible contamination of the detector or the environment (3). Some of the lines in the background spectrum can interfere with some of the lines in fission radionuclides spectrum or other complex spectra making the evaluation of the activities less reliable (4).

MATERIALS AND METHOD

Background gamma radiation spectrum has been recorded from 1993-1995 on two HPGe vertical detectors D1 and D2, relative

efficiency 23% and 20% respectively, both placed in a ground floor laboratory, with ventilation. Both detectors have Pb shieldings (10 cm) with layers of Cu (3mm) and Fe (10 mm D1, 6mm D2) on the inner wall of the cages (45cmx45cmx45cm). The cages differ in construction: the upper surface of D2 cage is covered with additional Pb movable plates, so there is no free space between Pb bricks. Detectors are used for radioactivity measurements in the environmental samples (soil, grass, food, building materials) in different geometries (standard Marinelli beakers, plastic bottles, planshettes).

Background has been recorded monthly and the means of count rates (imp/1000s) over the counting interval (250.000-400.000s) have been calculated. Energy calibration was performed with a standard set of point sources (COFFRET d'etalon gamma ECGS-2,Saclay with Ba-133,Co-57,60 & Cs-137).

RESULTS AND DISCUSSION

The results of the background radiation recordings on D1 and D2 detectors have been presented in Table 1. The data present the means ($\bar{X}$) of monthly counts rates (imp/1000s) for the three years and the coefficients of variation CV(%) for the 1993-1995 period (T) and for each year (93,94,95).

Table 1. Mean count rates $\bar{X}$(imp/1000s) and variations CV(%) in the background radiation spectrum for detectors D1 and D2 from 1993 to 1995

		Detector D1					Detector D2				
Radio-nuclide	E(KeV)	$\bar{X}$	CV(%) 93	94	95	T	$\bar{X}$	CV(%) 93	94	95	T
Th-234	63.3	5.2	4	11	4	3	1.9	11	11	5	8
PbX	74.8+77.1	11.5	47	29	52	6	6.3	22	42	58	16
PbX	89.7	18.2	10	10	16	3	8.7	7	17	19	7
U-235	185.7	8.2	19	24	19	11	3.6	8	7	16	8
Pb-212 +214	238.6 +241.5	8.6	28	14	36	5	6.4	9	24	26	9
Pb-214	295.2	11.6	49	48	57	7	7.1	24	52	60	16
Pb-214	351.9	18.9	51	30	54	10	11.5	25	52	61	18
anih.	511.0	15.0	4	9	3	1	10.9	2	4	1	4
Tl-208	583.1	1.1	17	22	24	11	0.8	22	20	8	6
Bi-214	609.3	14.2	47	29	54	10	8.7	27	48	59	19
Cs-137	661.6	1.5	12	10	10	3	2.7	32	18	5	12
Ac-228	911.1	0.6	24	24	31	17	0.5	17	20	20	2
Bi-214	1120.3	2.5	27	28	81	17	1.7	27	54	56	13
Co-60	1173.2	1.8	4	12	12	15	0.4	16	21	20	22
Bi-214	1238.1	1.2	35	36	55	13	0.7	30	46	44	19
Co-60	1332.5	1.6	12	17	8	14	0.4	27	17	22	22
K-40	1460.8	1.9	8	6	10	5	1.4	10	11	4	4
Bi-214	1764.5	2.3	23	27	51	13	1.4	25	44	54	16

The mean background peak rates counts recorded on D1 are generally higher than those recorded on D2 with better shielding construction (on the average the means of D1 are 1.5 time higher than the means of D2 and for energies up to 200 keV twice as high). Co-60 lines in both spectra are due to the amounts of Co in the shielding material, the shilding of D2 being much less contaminated. The presence of Cs-137 in the background spectra is due to the Chernobyl nuclear plant accident in 1986. However, higher peak counts on 661.6 keV of D2 point to the contamination of the detector while used for environmental samples activity measurements.

As for natural radionuclides, special attention should be paid to the unstable lines of Bi-214 and Pb-214 as their high seasonal variations within a year (due to variations of radon concentrations in air) could be a significant source of error when estimating low level activities of natural radionuclides in the environmental samples. Long counting time intervals (up to 400.000 s) can cause an additional error in activity estimation as some variations of the background radiation could occur in the interval. Averiging the background count rates over long counting time intervals can be another source of error, too. On the other hand, the lines of K-40, U-235 and Th-234 are relatively stable and their variations are within the total error of the method.

Relatively low variations (T<20%) over three years period for both detectors confirm the stability of the systems and of the environmental and measuring conditions.

Higher variations for lower values of the mean peak counts can be asigned to the errors in intensity estimation.

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International Intercomparisons for Radioactivity Measurements of Environmental Samples during Past Decade

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1. Introduction

During the period 1985~1994, our laboratory participated in a series of international intercomparison activities of radioactivity measurements sponsored respectively by WHO IRC, IAEA AQCS and IAEA MEL, with 20 samples measured in total. This paper describes the intercomparison activities concerned during past decade, methods of analysis and measurement, and measurement results. Some problems involved in measurements was discussed.

2. Samples

Among these 20 interncomparison samples, there were 12 samples from WHO IRC and 8 from IAEA. WHO IRC samples were liquid milk (No. 42L300), liquid effluent (No. 43E300), aquatic plant (No. 45V300), dry may blossom (No. 48V300), ground water (No. 56P300), mineral water spring (No. 49SH300), river sediment (No. 55SR300), white mine (No. 57VI300), pasteurized milk (No. 58L300), mineral water spring (No. 60SH300), liquid effluent (No. 59E300) and water (No. 61E300), which were required to analyses ^{3}H, ^{90}Sr, ^{58}Co, ^{60}Co, ^{134}Cs, ^{137}Cs, ^{106}Ru, ^{124}Sb, ^{110}m Ag, ^{226}Ra, ^{210}Po, ^{40}K, U and Th respectively. IAEA samples were Russian soil (IAEA-375), Russian grass (IAEA-373), Irish sea cockl flesh (IAEA-134), Irish Sea sediment (IAEA-135), Baltic Sea sediment (IAEA-300), Arabian Sea sediment (IAEA-315), black soil (IAEA-326) and Podssolic soil (IAEA-327), which were required to analyse transuranium elements, γ emitters and β emitters and to be given analysis of trace elements.

3. Analytical methods

Analytical methods used in the laboratory for measurement intercomparisons are shown in Table 1. Ge(Li) and HPGe gamma spectrometer were used for measurements of gamma rays emitting nuclides. Sample containers were employed according to the magnitude and shape of samples, including the original containers containing sample, 600 ml Marinelli Beaker, and ϕ 75 mm polyethylene container, about 25, 50 75 mm high respectively. For analyses of U, Th and K, chemical analytical methods were used in addition to Ge gamma spectrometer. To determine ^{134}Cs and ^{137}Cs in samples, direct measurement was firstly made by Ge gamma spectrometer and then so did again after radiological separation.

4. Reference sources and samples

Reference sources were from different laboratories in different countries as shown in Table 2. Standard point sources and reference point sources, containing ^{241}Am, ^{57}Co, ^{60}Co, ^{133}Ba, ^{203}Hg, ^{137}Cs, ^{54}Mn, ^{22}Na, ^{88}Y and ^{152}Eu, were purchased from China Institute of Atomic Energy, Britain Amersham Corporation and China National Institute of Metrology respectively. Standard solution and reference solution were from China Institute of Atomic Energy, WHO IRC and Amersham. Standard U, Th, U-Ra ore powder were received from Beijing Institute of Uranium Mine Geology.

Table 1. Analytical methods used in the laboratory for intercomparison

Nuclids	Methods
Gamma emitters	Ge(Li) or HPGe gamma spectrometer
^{3}H	Liquid sinctilation counter
U, Th	HPGe gamma spectrometer
U	Laser fluorescence or photospectrometry
Th	N-235 extractration photospectrometry, choromotography
^{226}Ra	HPGe spectrometer, radon emanation measurement
^{90}Sr	HDEHP choromotography or cation ion exchange chomoto-graphy, and beta measurement
$^{234}U, ^{235}U, ^{238}U$	Alpha spectrum analysis after evaporation and depletion
^{210}Po	Alpha measurement after evaporation and depletion
$^{134}Cs, ^{137}Cs$	Ge gamma spectrum meausrement after HPGe gamma reasurement or radiochemical analysis.

Table 2. Reference sources an samples

Type	Nuclide	Provided by
Point source	^{241}Am, ^{57}Co, ^{60}Co, ^{133}Ba, ^{208}Hg, ^{137}Cs, ^{54}Mn, ^{22}Na, ^{88}Y,	China Institute of Atomic Energy
Point source	^{241}Am, ^{57}Co, ^{60}Co, ^{133}Ba, ^{208}Hg, ^{137}Cs, ^{54}Mn, ^{22}Na, ^{88}Y,	Amersham, Britain
Point source	^{152}Eu	China National Institute of Metrology
Solution	^{60}Co, ^{137}Cs, ^{152}Eu, ^{241}Am	China Institute of Atomic Energy
Solution	^{58}Co, ^{137}CS, ^{60}Co, ^{226}Ra, U	WHO IRC
Solution	^{90}Sr, ^{137}Cs	Amersham, Britain
Ore powder	U, Th, U-Ra	Beijing Institute of Uranium Mine Geology
Soil	U, Th, Ra, K	China Institute of Radiation Protection
Spike sample (powder)	U, Th, Ra, K	China Institute for Radiation Protection
Spike sample (solution)	^{60}Co, ^{137}Cs, ^{152}Eu	China Institute for Radiation Protection
Spike sample (powder)	^{60}Co, ^{137}Cs, ^{152}Eu	China Institute for Radiation Protection

Radioactive point source is a basis of transferring national or international laboratory standard radioactivity quantity value, which transfers standard quantity value to reference materials or reference samples[1]. The first step for out laboratory transfers radioactivity quantity value is to check standard solution and reference solution. Measurement is made using Ge gamma spectrometer that was calibrated with standard and reference point sources to ensure system to have the capability of transferring standard quantity value.

In measurement intercomparison, radioactive standard solution and reference solution from various national laboratories are directly used as calibration standards of radiochemical analysis and ^{3}H measurements. Soil reference samples prepared by our laboratory were taken

from natural soils in Hengyang, Hunan Province, in which natural radioactive nuclide contents were determined by laboratories of the China Institute of Radiation Protection and the National Environmental Protection Agency. Additionally, our laboratory prepared soil spike samples and solution spike samples, which be directly used for measurement of gamma spectra. In measurements of solution sample, sediment samples and soil samples, ore powder separately containing U, Th, R-Ra and spike samples containing ^{137}Cs and ^{152}Eu were used.

5 Results

During these intercomparisons, the laboratory received 9 summary reports, 6 from WHO IRC and 3 from IAEA. Results shown a deviration within 10% of our reported value from WHO IRC reference values, with a few in exceed of 10%. There also were some data that shown no deviration from the WHO IRC value. Examples are ^{90}Sr and Ca in No. 42L300 milk in 1985, ^{235}U in No. 49SH300 mineral srping water in 1988, and U in No. 60SH300 milk in 1993. During the intercomparions of IAEA samples, the laboratory reported the analytical results of gamma secpectrometry, which were within the acceptable range. But no data were given for gamma emitting nuclides with low content in sample. Analyses of transuranium nuclides were not done.

6. Discussion

In the WHO IRC sponsored intercomparison, IRC presented reference values. Data from various laboratories were for quality assessment on the basis of a comparison with reference value. IAEA AQCS and MEL organized intercomparison haven't presented reference value, but taking median or total average as recommended value or informative value. So IAEA organized mesurement intercomparison, in fact, was the measurements for determining value, which raised higher requirements to those laboratories participating in intercomparison. In particular, reference sources or reference samples used for calibration must be national or international standards or tracable to national or internatinal standards. Careful analyses of measurement errors and correction factors will result in acqusition of reliable data which may be used as recommended or reference values.

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A NEW ALPHA SPECTROMETRY FOR INFINITELY THICK SOURCES USING SI DETECTORS.

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INTRODUCTION

In nuclear energy related facilities, contamination by transuranic nuclide must constantly be monitored.

As an example of the monitoring methods, thin electrodeposited sources prepared after the chemical separation processes, give high sensitivity in α spectrometry by Si detectors or gridded ionization chambers; thus, often used for α contamination analysis of environmental samples.

Because of the complicated source preparation processes, this method can hardly be recommended for analyzing many samples in a short period of time. For the cases of large area contamination, such as nuclear accidents, simple and rapid analysis is a prime importance.

As a simple α detection method, so-called gross counting technique by ZnS scintillation counters, is well known. But this method does not give α energy information at all and is insufficient to identify the nuclide. Consequently, α background by natural nuclide cannot be subtracted from the measured gross counts. And further, it is impossible to find the effective discrimination level for α particle counting; therefore, α counting efficiency for thick samples can only be determined by the actual measurement for the identical sample configuration and for the identical nuclide.

The purpose of this work is to develop a new α spectrometry method for thick samples.

Experimental results and mathematical analysis have confirmed that qualitative analysis of α nuclide in thick samples is feasible. And problems related to the analysis are also discussed in this report.

EXPERIMENTAL METHODS

Si Detectors

Si detectors used for experiments are Surface Barrier Detectors (SBD)-(1),fabricated from p-Si wafers. The vacuum evaporated Al face is the sensitive side for α particle injection. With the Al face up, fine powdered samples were directly mounted on the electrode, so that the Al layer thickness is 50 $\mu g/cm^2$ which is thicker than usual detectors. Particles which contain a small amount of acid or alkali, such as ion exchange resins, often result leakage current increase. To prevent the degradation of detector characteristics, outer edges of the electrodes are covered by Epoxy Resin as shown in Fig.1.

Measurements

^{210}Po of 6 Bq/mg was adhered to fine Ni powder of several grain sizes. The powder was directly mounted on the Al electrode of an SBD and the α spectrum was measured. Also measured in the same manner with another SBD, was the powder of low grade uranium ore. Another way of thick alpha sample spectroscopy is :a) filling

a sample dish, b)making the powder surface flat, c)placing an SBD close to the sample, and d)measuring the α spectrum. However, if soil samples were measured by this method, Rn and Tn daughters stick to the electrode and their contamination give problems to the spectrum analysis(3).
It was confirmed that the powder samples adhered to Al electrodes could be removed almost completely by soft brushing and/or methanol cleansing.
α energy spectrum measured at the surface of an infinitely thick sample (thicker than α ranges), in which α emitters distribute uniformly and α particle straggling is negligible, is given by the following formula(3);

$$dn(E)/dE = 0.25 \cdot S \cdot D/(dE/dx)_E \qquad [1]$$

where $dn(E)/dE$ [keV^{-1}] : differential detector counts
S [cm^2] : detector sensitive area
D [Bq/mg] : specific α activity in the sample, and
$(dE/dx)_E$ [$kev \cdot mg^{-1} \cdot cm^2$] : stopping power of the medium at E

EXPERIMENTAL RESULTS AND DISCUSSION

Fig.2 shows two groups of different spectra : 1)the measured α spectra of ^{210}Po adhered to three different grain sizes of indefinitely thick Ni powder samples, and 2)the α spectrum calculated by formula [1] with Ziegler's Stopping Power Data. In case the grain size is sufficiently smaller compared to α range the measured spectrum agrees well with the calculated results. While grain size of the powder increases, alpha spectra peaking at the high energy edges tends to be significant.This tendency was unexceptionally observed in other metal powders. This peaking phenomena can be explained as followings: if grain size increases, the portion of α particles which does not penetrate grain medium or go through short path to SBD, increase. Because all the α emitters are adhered on the surface of grains comparative to or larger than the α particle ranges. If noticeable peaking is observed with a sample, it means α emitters are not distributed uniformly in it ,and qualitative analysis by formula [1] can not be applied. It means some process has to be made. As shown in Fig.3 in the case of uranium ore, this can be easily solved by grinding the sample into fine powder in an agate mortar. Peaking is, however, advantageous to identify α nuclide. This is especially true for the identification of artificial α emitters, because they are, in most cases, attached on the surface of sand or ion exchange resin grains.
Total α counting rate for infinitely thick samples, as calculated by integrating formula [1], is shown in Fig.4. Since the medium is SiO_2, this result can be applied for soil samples as a good approximation.

CONCLUSION

(1)Without making any chemical separation process, α spectrometry can be made by using a Si detector and infinitely thick samples.
(2)Qualitative analysis for each nuclide can be made, if sample components are identified, by grinding samples into fine powder.
(3)Edge peaking in an alpha spectrum can effectively used for the nuclide identification.

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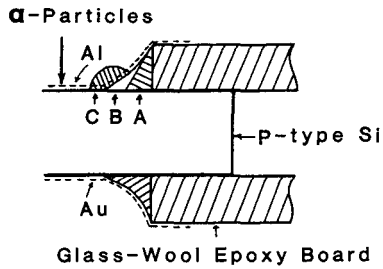

Fig.1
Cross-sectional view of Si detector.
A : Epoxy resin, with inorganic compounds
B : Epoxy resin, with amine type hardener
C : Epoxy resin, with polyamide hardener

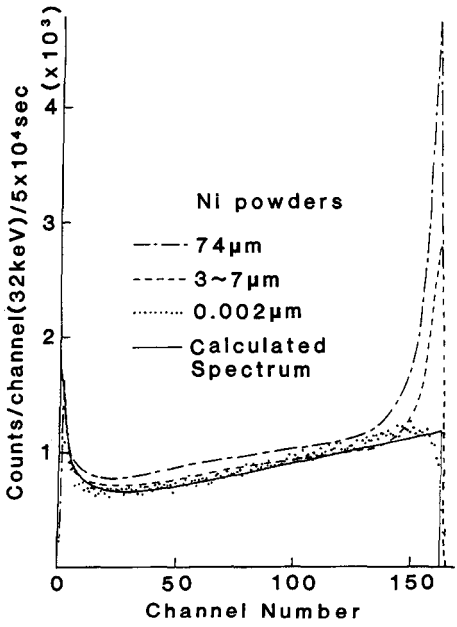

Fig.2
Three α spectra of infinitely thick samples (^{210}Po was doped to fine Ni powder of three different grain sizes) and the α spectrum calculated by Formula [1]

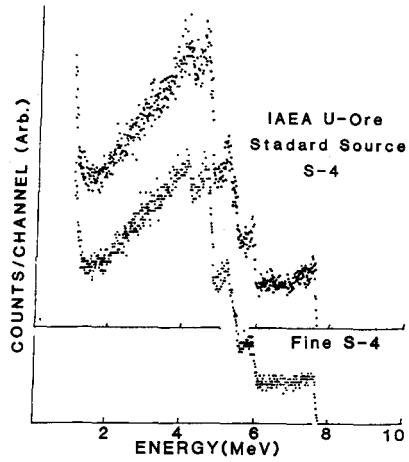

Fig.3
α spectrum of IAEA uranium ore (infinitely thick sample);
the spectrum at the lower section is, of the finely crashed powder ore by an agate mortar.

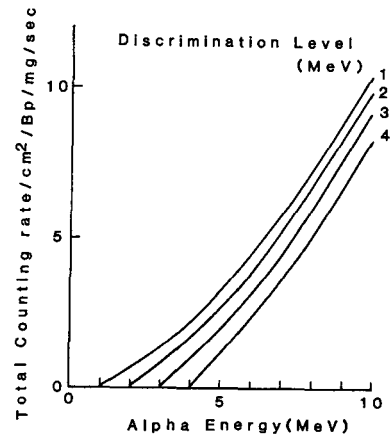

Fig.4
Total counting rate of α particles by integrating Formula [1], assuming SiO_2 as the medium. Detector area : 1 cm^2
Specific α activity : 1 Bq/mg
Numerical figures on curves indicate the lower energy limit of the integration.

AN IMPROVED METHOD OF IN-SITU MEASUREMENT OF ^{137}Cs CONCENTRATION IN SOIL TEN YEARS AFTER CHERNOBYL.

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Introduction

This contribution is aimed to the possibility of improving the in-situ gamma spectrometry to be independent on a priori knowledge about a depth distribution of ^{137}Cs in soil and sufficiently sensitive for the measurement of the post-Chernobyl ^{137}Cs at present, as well. The depth distribution of ^{137}Cs averaged over a large area of soil is obtained by unfolding of the detector responses to primary and in soil forward scattered photons. The proposed method employs detector with and without collimator.

Methods

The detector response N(i) to ith characteristic of gamma ray field above ground with radionuclide distributed homogeneously in horizontal direction is described by integral equation :

$$N(i) = \int \sigma(i,\zeta)\, A(\zeta)\, d\zeta \qquad i=1,2,...,n \qquad (1)$$

where $\sigma(i,\zeta)$ [$Bq^{-1}.m^2$] - detector response in gamma field of plane source of unit activity-per-area located at the depth ζ, $A(\zeta)d\zeta$ [$Bq.m^{-2}$] - unknown activity of the plane source to be determined by unfolding, ζ [$kg.m^{-2}$] - depth in the soil in units mass-per-area.

The responses of collimated and uncollimated detector $\sigma(i,\zeta)$ to primary 0.662 MeV and to 0.62-0.655 MeV photons scattered in the soil are considered in the proposed method. The detector responses are obtained by calculations in combination with experimental calibration.

Calculational procedures

The detector response $\sigma(i,\zeta)$ to ith characteristic of gamma field with flux $\phi(i,\zeta,\theta,E)$ per-unit-activity of plane source at the depth ζ in the soil can be described as follows:

$$\sigma(i,\zeta) = \iint \phi(i,\zeta,\theta,E).R(i,\theta,E)d\theta.dE \qquad (2)$$

where R(i,θ,E) - detector response in peak of total absorption for a parallel photon beam of energy E impinging at angle θ to the detector axis.

The Equation (2) can be simplified by assumption that the dependences R(i,θ,E) are in the energy range of photons from 0.62 up to 0.662 MeV constant:

$$\sigma(i,\zeta) = \int \Phi(i,\zeta,\theta).R(i,\theta)d\theta \qquad (3)$$

where $\Phi(i,\zeta,\theta)\, d\theta$ - photon flux at angle θ integrated over energy, R(i,θ) - angular dependence of detector response for a parallel photon beam of energy 0.662 MeV.

The photon flux Φ has been calculated by our Monte Carlo code SOILSC. Homogeneous and isotropic plane source of thickness 0.5 $g.cm^{-2}$ was simulated up to 85 $g.cm^{-2}$ of depths in the soil, starting with the plane source on the ground. The plane source radii were chosen 100 m to represent an infinitive plane soil contamination with radionuclide ^{137}Cs.

Experimental procedures

A portable N-type high purity germanium HPGe detector with relative efficiency of 12,5% and resolution of 1,7 keV for 1,33 MeV gamma rays has been used. The detector was supported by a tripod and the front of the detector was at the height 100 cm above the ground. The orientation of the detector is facing downward.

A cylinder-shaped collimator has been used to modify the angle distribution of photons impinging to the detector surface. The collimator is made from lead cylinder with the outer/inner diameter 19/10 cm and height 10 cm laying on aluminium cylinder with the outer/inner diameter 30/10 cm and height 2 cm. The bottom of aluminium disk was located at height of 100 cm above the ground.

The experimental calibrations were aimed to determine the responses $R(i,\theta)$ (in Equation 3) of collimated and uncollimated detector in the energy region 0.62-0.655 MeV and in peak of total absorption of the 0.662 MeV photons, as well, and to asses the background in the energy region 0.62-0.655 MeV due to photons from the natural sources.

Application to post-Chernobyl ^{137}Cs

The photon spectra were recorded over lawns in Southern Slovakia. In-situ gamma spectrometry was performed with and without collimation of the detector, which was located above the centre of the area used for soil sampling.

Figure 1 shows the pulse height distribution measured by the unshielded detector at the location No.2 in the energy interval 0.55 - 1.0 MeV. Exponential approximation of the spectrum in the interval 0.67 - 1.0 MeV, bold line in the figure, is used for assessment of background for 0.62-0.655 MeV photons scattered in the soil.

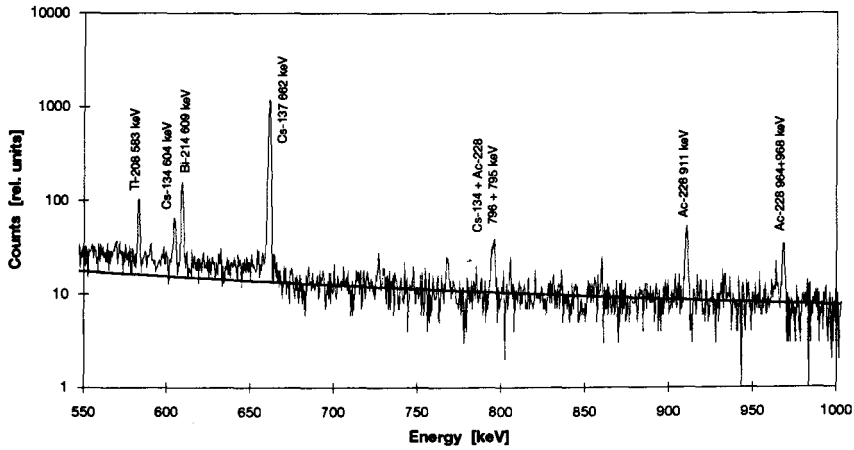

Figure 1. The pulse height distribution measured by the HPGe detector at the location No.2.

Figure 2 shows depth distributions of the ^{137}Cs activity in soil measured in surrounding of Bratislava, the places were in distance 20 km. The results obtained by presented method are compared to results by soil sampling. The relative standard deviation of the ^{137}Cs activity in soil samples were <10% for each depth of soil.

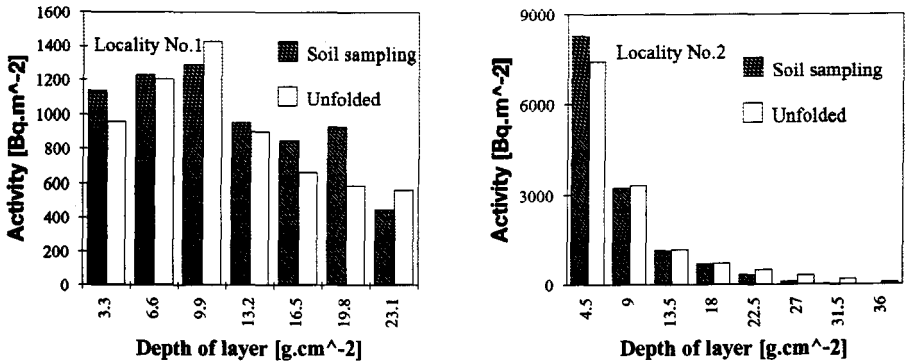

Figure 2. Distributions of the ^{137}Cs activity in soil profile of lawn in the locality No.1 and 2 determined by presented method and by soil sampling, as well.

The downward transport of ^{137}Cs have occurred to a peak concentration that lies below the surface of lawns. The differences between presented method and soil sampling of the distributions in Figure 2 are within 20%, except of distributions at the deeper depths where resolution of unfolding method is worse.

The ^{137}Cs activity-per-mass and/or activity-per-area in soil determined by presented method of deconvolution, by sampling method and by standard in-situ spectrometry using the exponential depth distribution in the soil (1), that has been determined by exponential least squares fit of results of soil sampling, are given in Table 1.

Table 1. ^{137}Cs activities in soil of lawns in Southern Slovakia determined by various methods.

No. of	Presented method		Soil sampling		Standard method	
locality	[kBq.m^{-2}]	[Bq.kg^{-1}]	[kBq.m^{-2}]	[Bq.kg^{-1}]	[kBq.m^{-2}]	[Bq.kg^{-1}]
1	6.3±1.6	274±68	6.8±0.7	258±26	-	307
2	13.2±3.5	-	13.8±1.5	-	12.5	-
3	1.2±0.5	39±16	1.1±0.2	36±7	-	33

Conclusion

In the presented method a fairly accurate concentration estimate for the top layer of the soil, down to a depth of about 30 g cm^{-2}, is obtained. The analysis of the spectra collected with the detector, used in this study indicates that ^{137}Cs concentration in soil, in time of 10 years after the Chernobyl accident, would be measurable using a middle HPGe detector (about 30% of relative efficiency) and a counting time of the order 1 hour or less. Even with smaller detectors, ^{137}Cs concentrations of 5 kBq m^{-2} are measurable and the depth distribution of ^{137}Cs of activities above 10 kBq m^{-2} in soil can be estimated by presented method when counting time of the order 3 hours is used.

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DETERMINATION BY IN SITU GAMMA SPECTROMETRY OF OUTDOOR AND INDOOR DOSE RATES IN POPULATED AREAS OF GREECE DUE TO NATURAL AND ARTIFICIAL SOURCES OF RADIATION

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INTRODUCTION

Knowledge of radiation levels in buildings is important in the assessment of population exposure (1). This is because most individual spend the majority of their time indoors exposed to the radiation from the radionuclides (mainly Ra-226, Th-232, their decay products and K-40) in walls, floors and ceilings. The rest of their time is spent outdoors on roads or near buildings, which exposes them to radiation from materials used for construction. In situ gamma spectrometry is a powerful tool to study indoor and outdoor dose rates. In the present work we report 300 indoor and outdoor in situ γ-spectrometry measurements performed in the two most populated towns of Greece: in Athens, the capital of Greece and in Thessaloniki, the second town of Greece.

EXPERIMENTAL PROCEDURE

The portable Ge detector used for the in situ measurements is a high purity Ge coaxial cylinder 44 mm in diameter and 41 mm in length with a relative efficiency of 10%. The spectrum is collected in a portable multichannel analyzer that also provides high voltage and pre-amp. to the detector. The first analysis of the spectra can be performed in a portable computer (notebook) connected to the multichannel analyzer with the use of a home-made software. The measurements are performed with a downward tripod-mounted facing detector at 1m above soil surface for a counting period of 2000s.

The calibration procedure in field spectrometry consist of not simply identifying the γ emitters present at a site but mainly to convert the measured count rates for each γ emitter to some meaningful quantity such as dose rate or radionuclide concentration. The procedure used in this work is the one proposed by Beck et al. (2) and Helfer and Miller (3). We have shown (4) that the field geometry factors derived from the calibration procedure can be applied to indoor geometry at least in the case of masonry structure which is typical in the most of the buildings and houses in Greece.

The γ emitters present in a typical measured spectrum are the various nuclides of the Uranium and Thorium series, the K-40 and Cs-137 mainly from the Chernobyl accident. In the calibration procedure used an equilibrium is assumed along the various nuclides in the U and Th series, so that the measurement of any one spectral line yields concentration values for the entire series, although averaging several spectral lines will give a more precise value. For the U-series we used the lines of 0.352 MeV (Pb-214) and 0.609 MeV (Bi-214) and for the Th-series the lines of 0.583 MeV (Tl-208) and 0.911 MeV (Ac-228). Disequilibrium in the U series due to exhalation of Rn-222 from the soil surface or the building materials can cause a percentage error in the U series exposure rate, however according to Beck et al (2) this error is smaller than 15%. The absolute error in the absorbed dose rate in air from U-series, Th series, K-40 and Cs-137 is of the order of 20% mainly due to the accuracy of the semiempirical calibration factors used for the calibration of our Ge detector for the indoor and outdoor in situ measurements.

RESULTS AND DISCUSSION

In Thessaloniki and surroundings we have performed 143 indoor measurements, 19 outdoor measurements on roads and pavements, 37 outdoor measurements over soil. The mean value of the indoor dose rate in air is $\dot{D} = 70 \pm 15$ nGy/h with a relative contribution of 23%, 34%, 42% and 0.7% due to Uranium series, Thorium series, K-40 and Cs-137 respectively (Table 1).

For the houses and buildings constructed before 1986 the actual indoor Cs-137 gamma radiation is only due to the presence of Cs-137 from the Chernobyl accident in the nearby outdoor environment. The radiocesium contamination in Thessaloniki is mainly due to the Chernobyl accident, since the contribution from the nuclear weapons fallout can be neglected (5), (6). For the houses and buildings constructed after 1986 the actual indoor Cs-137 gamma radiation is due not only to the presence of Cs-137 from the Chernobyl accident in the nearby outdoor environment but also to the possible presence of radiocesium in the building

materials. In Figure 1 is shown the indoor dose rate in air due to Cs-137 as a function of the construction year of the house or building. It is impressively shown a pronounce peak for the year 1986 where the Chernobyl accident occurred. This fact indicates that for houses and buildings that were under construction in 1986, the Cs-137 act also as an indoor source term, however with small contribution to the total dose rate.

THESSALONIKI

	number of measurements	mean dose rate	Contribution (%)			
			Uranium Series	Thorium Series	K-40	Cs-137
Indoor	143	70±15	23	34	42	0.7
Roads & Pavements	19	33±20	26	26	40	8
Over Soil	37	52±16	17	32	36	15

ATHENS

	number of measurements	mean dose rate	Contribution (%)			
			Uranium Series	Thorium Series	K-40	Cs-137
Indoor	79	25±10	31	27	41	0.3
Roads & Pavements	8	14±8	35	17	47	1
Over Soil	12	22±11	28	27	39	6

Table 1. Absorbed dose rate in air (nGy/h)

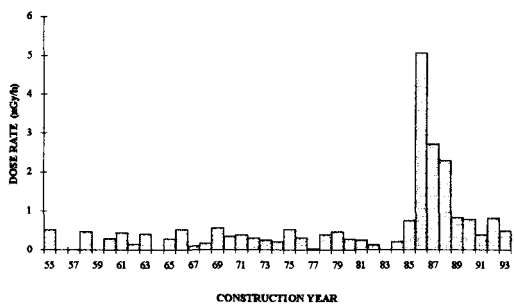

Figure 1. Indoor dose rate due to Cs-137 as a function of the construction year

The fact that for the houses and buildings constructed before 1986 the actual indoor Cs-137 gamma radiation is only due to the presence of Cs-137 in the nearby outdoor environment, gave the opportunity to measure in real conditions the attenuation of the gamma radiation by the building materials. For such buildings the dose rate due to Cs-137 was determined outdoor, in the vicinity of the building, as well as inside the building. These values gave dose attenuations, due to the shielding of the buildings, between 75% and 100%. In Figure 2 is presented the distribution function F(x) of the dose attenuation for these houses and buildings.

The mean value of the outdoor dose rate in air deduced from in-situ outdoor measurements on roads and pavements is $\dot{D} = 33 \pm 20$ nGy/h with a relative contribution of 26%, 26%, 40%, 8% due to Uranium series, Thorium series, K-40 and Cs-137 respectively. It can be seen that the contribution of Cs-137 is one order of magnitude higher than the one deduced from the indoor measurements and that the total outdoor dose rate $\dot{D}$ is smaller compared with the indoor dose rate. The higher value of the indoor dose rate compared with the outdoor dose rate can be easily explained from the fact that for the indoor environment we have a 4π source geometry compared to the 2π source geometry of the outdoor environment. As the number of country-wide indoor surveys is relatively small compared to those conducted outdoors, the indoor to outdoor ratio has been commonly used for the estimation of the average indoor absorbed dose rate in air from the outdoor values.

The mean value of the outdoor dose rate in air deduced from 37 in situ outdoor measurements on soil is $\dot{D} = 52 \pm 16$ nGy/h with a relative contribution of 17%, 32%, 36% and 15% due to Uranium series, Thorium series, K-40 and Cs-137 respectively. The Cs-137 contribution to the dose rate $\dot{D}$ is not any more negligible (15%) and is comparable to the contribution of the Uranium series (17%). The Cs-137 contribution to the outdoor dose rate over soil is two times higher than the one measured on roads and pavements. The influence of building, road and other materials used in urban areas affect significantly the external dose rate. Therefore estimation of dose rates based on measurements of soil samples do not apply, in general, to urban areas.

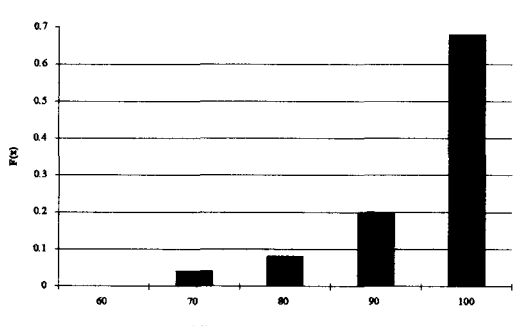

Figure 2. Distribution function of dose rate attenuation between indoor and outdoor measurement in the same place (26 measurements

From the above results, and taking

into account that 80% of people's time is spent indoors, a mean annual equivalent dose of 0.4 mSv for the γ-dose radiation can be deduced, for Thessaloniki.

In Athens and surroundings we have performed about 100 in situ measurements. The mean dose rate in air and the relative contribution of the Uranium series, Thorium series, K-40 and Cs-137 for the 79 indoor measurements, 8 outdoor measurements on roads and pavements, 12 outdoor measurements on soil are presented in Table 1. The presented mean indoor and outdoor dose rates are smaller by a factor of about 2.5 compared with the corresponding values in Thessaloniki. For all measurements the highest contribution to the D dose rate is due to K-40. The Cs-137 contribution to the indoor and outdoor dose rate in Athens is one order of magnitude smaller compared with the value obtained in Thessaloniki. This is explained from the fact that the Cs-137 deposition in Thessaloniki (~20 kBq/m^2) due to the Chernobyl accident was considerably higher than the deposition in Athens (~2 kBq/m^2).

	Indoor average absorbed dose rate in air nGy/h	Outdoor average absorbed dose rate in air nGy/h	indoors to outdoors ratio
Austria	71	43	1.65
Norway	95	73	1.6
Germany	70	94	1.36
Poland	-	87	1.2
Sweden	96	-	-
France	99	81	1.11
world populated weight average	80	57	1.4
Athens	24.5	18.8	1.3
Thessaloniki	70	52	1.64

Table 2. Comparative mean indoor-outdoor dose rates.

From the indoor and outdoor dose rates a mean annual equivalent dose of 0.14 mSv for the γ-dose radiation in Athens, assuming that 80% of people's time is spent indoors.

In table 2 the mean indoor-outdoor dose rates for Athens and Thessaloniki are compared with National averages from different countries and with the world populated weighted average (1).

CONCLUSIONS

The main conclusions of the present work are:

- First results of 212 indoor and 86 outdoor in-situ gamma dose measurements at the two most populated towns of Greece were presented. The mean absorbed dose rate in air and the relative contribution to the dose rate of the Uranium series, Thorium series, K-40 and Cs-137 were determined. From these results a mean annual equivalent dose of 0.14 mSv and 0.4 mSv for the γ-radiation could be deduced for the population of Athens and Thessaloniki respectively.
- The enhancement of the outdoor exposure rate due to radiocesium deposition from the Chernobyl accident is less than 15% for parks and gardens and less than 8% for roads and pavements.
- Estimation of dose rates based on measurements of soil samples do not apply, generally, to urban areas.
- For houses and buildings under construction in 1986, the year of the Chernobyl accident, radiocesium acts also as indoor source term however with small contribution to the total dose rate.
- The attenuation of the outdoor radiation (661.6 keV) observed, due to the shielding of the houses, was at least 75% up to 100%.
- The mean indoor-outdoor dose rates in Athens and Thessaloniki are smaller than the world indoor-outdoor population average and National averages of European countries.

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IRPA9
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April 14-19,1996
Vienna, Austria

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Abstract No.

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Mini-Presentation

PAPER TITLE Airborne Gamma-Ray Spectrometry in Environmental and Accidental Monitoring

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MAJOR SCIENTIFIC TOPIC NUMBER 4 (see page 7)

ABSTRACT (See instructions overleaf)

The airborne gamma-ray scintillation spectrometry enables to detect photons, emitted by natural radioelements (K, U, Th) as well as man-made contaminants, dispersed in the soil surface layer. Proper mathematical processing of experimental spectrometric data is needed to calculate superficial / specific activities of individual radionuclides and air kerma rates in the reference level 1 meter above ground. On-line processing is desired for mapping this values over the scanned region of interest.

The new data processing method for ENMOS airborne gamma-ray scintillation spectrometry system (produced by PICODAS Group Inc.) was developed. Using the Monte Carlo method, the spectrometer responses for natural radioelements, selected artificial contaminants (including depth distribution models) and set of flight altitudes were calculated. Two basic spectrometer respose matrixes for energy interval up to 3 MeV and mostly used altitudes were prepared. Different methods for the spectra decomposition to contributions of individual radionuclides were tested and impact of the statistical and systematic errors was analysed. Main advantages of this method are: 1) no experimental calibration and corrections of the energy dependence are necessary, 2) calculation of responses can be provided for any experimental arrangement.

The method can be effectively applied for the airborne gamma-ray spectrometry data interpretation in environmental monitoring, operational or accidental monitoring / mapping of the radioactive contamination and for geological mapping. Some examples of monitoring results are also presented.

AEROGAMMA-SPECTROMETRIC MONITORING OF THE EARTH SURFACE

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V.S.Danilov, V.V.Filimonov (A.A.Logachyov Institute, VIRG-Rudgeophisika)

The basic task of radioecological monitoring is in receiving a reliable information about radiation conditions, and,first of all, their changes for estimation of radiation safety in the location of people's residence and activities. The aerogamma-spectrometric (AGS) method is the only one to provide for a reliable and efficient information about radiation conditions on large territories. This method has been successfuly applied in Russia since 1963, originally being used to solve problems in geology. In the course of the AGS method improvement in Russia, a special attention was paid to increasing the sensitivity and ensuring the stability of equipment parameters, as well as to the metrological assurance of AGS measurements as a whole.

A high sensitivity is ensured by using a set of identical detecting units (their number is limited only by the possibilities of an aircraft), while stability of parameters is provided by an effective system of energy scale stabilization. As the experience shows, when using an aerogamma-spectrometer with 22 NaJ(Tl) crystals, 200 mm in diameter and 100 mm in height, without any adjustment, during 3 - 5 months, the ambient temperature changing from -5° to $+50^{\circ}C$, the energy scale for each crystal and the whole detecting unit is not changed by more than 0.5 % from the initial value.

When advanced equipment is available, methodical and metrological support of the work plays the most important role for receiving a reliable information in AGS monitoring. Aerogamma-spectrometers have the same measurement assurance as energy spectrometers. A high quality of the final result is achieved by using a set of measurement techniques. Natural testing grounds certified in the rank of State standard materials of the Russian Federation, are used as a refernce standard. Several of such testing grounds have been already certified for the content of thorium, uranium and potassium in rocks and the radiocesium in the upper soil layer. Small-volume (up to 10 l) working sources are used to adjust and control the stability of equipment parameters.

Measurement results are reduced to the Earth surface level taking into account the distortions of the energy spectrum of rock gamma-radiation by the atmosphere and the differencies in passing of rock gamma-radiations and technogenic radionuclides through the atmosphere. Preliminary results have been obtained on accounting for the local landscape peculiarities and the nature of penetration of radionuclides in the soil.

The AGS technology (VIRG-Rudgeophisika) as a whole (equipment, methodical and metrological support, application packages, etc.) has made a good showing to solve both various geological and radioecological problems. Specifically, this technology was widely and successfully used for mapping of the radioactive contamination of the territory after the accident at the Chernobyl atomic power station. A high reliability of the AGS results is proved by earth sampling.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Abstract No. 90891
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Mini-Presentation

PAPER TITLE ENVIRONMENTAL GAMMA DOSE RATE MONITORING STATIONS: DISCUSSION ON DOSE ~~RATE~~ QUANTITIES; SITE LOCATION AND CALIBRATION SCHEMES

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2. (see page 7) or 4.1

ABSTRACT (See instructions overleaf)

ENVIRONMENTAL GAMMA DOSE RATE MONITORING STATIONS: DISCUSSION ON DOSE QUANTITIES, SITE LOCATION AND CALIBRATION SCHEMES

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In view of decreasing tolerance levels for ionizing radiation, there is a great effort all over the world to advance environmental monitoring equipment close to its physical and technological limits. The author proposes a new concept for high precision monitoring of environmental gamma dose rate, together with decision schemes for the "efficient" operation of these devices.

- ☐ Which of the dose quantities (as preferred in different countries today) is most appropriate: exposure (USA), air kerma (F), photon dose equivalent (D) or ambient dose equivalent (potential future standard) ?
- ☐ Where exactly is an appropriate site for the placement of an environmental gamma dose rate probe? Which "structures" in the vicinity of the probe should be avoided?
- ☐ How can environmental measurement systems be calibrated in-situ? What about the cosmic components of natural background radiation? Which methods for automatic quality assurance can be provided?

AN ORIGINAL METHOD TO MEASURE THE NATURAL RADIOACTIVITY IN THE OUTDOOR ENVIRONMENT BY MEANS OF A MOBILE LABORATORY.

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ABSTRACT

The Nuclear Protection and Safety Institute has developed a special tool, a mobile laboratory, to perform measurement of natural radioactivity (gamma radiation and radon activity in the air) in the outdoor environment. With this mobile laboratory, it is possible to study the temporal and the spatial variations of the natural radioactivity. For that, measurements are made either by stopping the mobile laboratory at one point, or while the vehicle is moving. This last configuration allows to obtain cartographies of gamma radiation and radon level, that can be correlated with the geological characteristics of the site under study. This methodology can be applied to study the impact of radium-rich by-products storage site in the outdoor environment. It can provide very rapidly information useful for the site operator, the local authorities and the population.

Measurements made in France with this mobile laboratory show significant various areas as a function of the geological parameters. For example, the gamma dose rate is comprised between 60 and 300 $nGy.h^{-1}$ and the radon activity between 5 and 100 $Bq.m^{-3}$ and even higher.

INTRODUCTION

In the open air, the activity of radon gas depend locally on the exhalation rate and on the conditions of atmospheric diffusion. We can observe temporal and spatial variation in the range of a few $Bq.m^{-3}$ to several hundred $Bq.m^{-3}$ (1-2).
In fact, depending on the thermal structure of the atmosphere, with the presence or absence of temperature inversion, and on others meteorological parameters, the observed concentrations vary at a given site by a factor from 10 to 100.
Radon concentration also depends on the geological and pedological characteristics of the studied site (for example granitic or sedimentary area and parameters of the soil such as porosity, humidity, permeability, ...).So, we observe spatial changes in radon concentration due to the type of soils and not to the variation of meteorological parameters.

DESCRIPTION OF THE MOBILE LABORATORY

To perform measurement of natural radioactivity (gamma radiation and radon activity in the air) our Institute has developed a mobile laboratory. The vehicle is a diesel powered, cross-country four-wheel drive. The inside is fitted out with racks. These are equipped with silentblocs, selected according to the weight of the instruments, to eliminate low frequencies emitted by the vehicle engine or present in road noise. The measuring equipment is modular to accommodate the various tasks the mobile laboratory may be assigned.

It is equipped with the following:

- an electrical power generation system which provides 24 hours of measurement autonomy,
- a navigation system,
- a data acquisition and processing unit,
- ionisation chambers, for continuous measurement of the gamma dose rate, at approximately 2 m off the ground, and of the radon activity concentration. The air samples are taken at approximately 1.5 m off the ground.

The mobile laboratory is used to measure the temporal and/or spatial evolution of a pollutant on times scales ranging from a few minutes to several hours, and distance scales ranging from a hundred metres to tens of kilometres. Several measurement procedures have thus been developed.

Measurements are made as follow:

- at fixed stations, at several points of the site, over periods of several hours to several days and under various conditions of atmospheric diffusion. This reveals variations according to time of the day which are related, for example, to temperature changes at night;

- along various points of paths determined according to the site's characteristics (i.e. road infrastructure, housing, orography, etc.) and according to weather conditions under various conditions of atmospheric diffusion. This reveals spatial variations;

- at semi-fixed stations : the mobile laboratory is immobilised for 4 to 5 minutes at a point of known co-ordinates (x, y and z) then moved by a few hundred metres (in accordance with the orography) to carry out new measurements.

RESULTS

Figure 1 shows the results of cartographies made in east of France. These itineraries cross trough a granitic area (top of the map) and a sedimentary area (bottom of the map). Accordingly we find more radon in the north than in the south. The same phenomena is observed for the gamma dose rate. During the night the radon activity reached 200 $Bq.m^{-3}$ in the north for some points. For a same point of measurements, we observe that the radon concentration is increased by a factor of 2 between night and day.

This methodology can be applied:

- within the framework of studies on radon in buildings, to identify the radon-prone areas. Thus, some mitigation technics can be applied to reduce radon at home;.
- to study the impact of radium-rich by-products storage site in the outdoor environment. It can provide very rapidly information useful for the site operator, the local authorities and the population (3);
- to study the radiological regional background.

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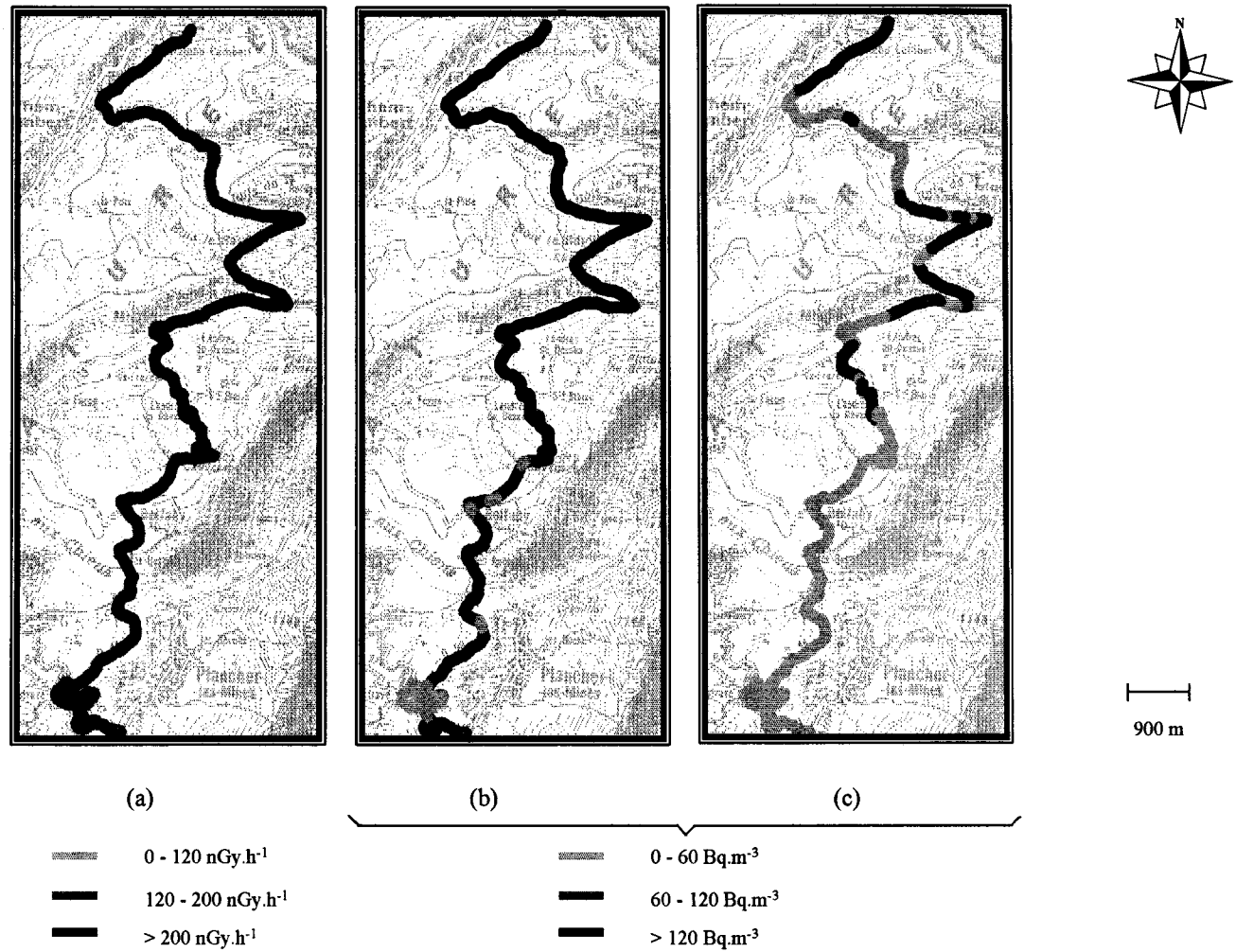

Figure 1: Cartographies of: (a) gamma dose rate, (b) radon concentration during the night, (c) radon concentration during the day

MOBILE LABORATORY FOR MEASURING RADIOACTIVITY IN THE ENVIRONMENT

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The emergency contingency plans of nuclear power stations provide for the measurement of radioactivity in the environment and, particularly, using mobile laboratory vehicles.
The originality of these vehicles is that they have been specially designed for nuclear power plants. EDF was able to achieve this design thanks to the standardisation of it nuclear power plants. This system is designed to enable non specialist technician to make reliable measurements easily.
The use of these vehicles is an integral part of the fixed environmental monitoring system of each nuclear power plant and includes the task of taking measurements in the field in order to corroborate the impact of radioactive releases into the environment in comparison with theoretical forecast calculations.

These vehicles are equipped mainly with the following:
- A radiation meter, which monitors activity continuously and the data from which is processed by a PC.
- A gamma spectrometer with a NaI probe for the specific measurement of Iodine and Cesium.
- Automatic beacons: these are truly portable environmental monitoring stations. They are used to gather aerosols and rain water and are fitted with a built-in radiation meter to measure exposure levels.
- A 220V electrical power unit supplied by a converter, which is supplied by a battery capable of operating for 48 hours.

All data collected are processed by the on-board measurement system. The data acquisition and processing system is designed to limit operator interventions to the minimum:
- Input data are processed automatically.
- Data and results are saved automatically without the intervention of the operator.

In continuous operating mode, the value of the dose rate is calculated every 10 seconds using data from detectors mounted on the side of the vehicle. The results are displayed in the form of a histogram and data are saved to the hard disk periodically.

The operator can «mark» certain points on the curve by assigning comments to them.
The gamma spectrometry measurements are taken automatically. Energy calibration is performed on power up and the system includes an automatic spectrum stabilisation function.
In order to start data acquisition, the operator installs the sample and selects the appropriate geometry.
The results are displayed on the screen and are saved to the hard disk on completion of the acquisition phase.

The parameters of the automatic beacons can be set. Sampling can be triggered either according to dose rate or according to time setting.
The measurement of exposure levels is continuous.
The sampling device consists of two identical pumping circuits, each of which is fitted with a removable cell which contains a filter with 1 or 2 active charcoal filter cartridges.
Rain water samples are also taken by a device consisting of two bottles.
Once the system is connected to the computer, data are downloaded to the computer automatically.

In addition, various items of sampling equipment are installed in the vehicle. The samples thus collected by them are subjected to gamma spectrometry analysis later in the various laboratories of the power plant.

ENVIRONMENTAL RADIOACTIVITY MONITORING IN LATVIA

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The first measurements of radioactivity in Latvia were performed in 1967. During the following 20 years the network of the radioactivity control were organized for all regions of Latvia. Until 1991 in Latvia the sample collection were solely performed, whereas the measurements were done in Obninsk (Russia). At present only a small research reactor with the thermal power of 5000 kW and the depository of radioactive waste are located in Latvia. The nearest nuclear power stations are situated in Ignalina (Lithuania), St.Petersbourg, Finnland and Sweden. A power of our reactor is small, therefore the increasing of radiation background can be caused only from the neighbour nuclear power stations or transport of radioactive materials.

After Chernobyl reactor accident a great attention from different organizations was paid to the control of radiation situation in Latvia. In this connection the determination of radionuclide concentration in environmental and food samples were organized in the Laboratory of nuclear spectroscopy at Nuclear Research Center of Academy of Sciences of Latvia till 1986 (1,2).

At present there are several organizations and paralel systems which ensure the control of the radioactivity in our republic.The first control system was organized in the Latvian Hidrometeorological agency. Now the system for the continuous measurements and processing of the data is developed.

At present the 46 stations for measurement of gamma radiation (approximately one in 1390 km^2) are established in Latvia. The largest density of such stations is found inside of the 100 km zone around the Ignalina nuclear power station..

The control program consists of registration of gamma radiation dosis rate from 46 stations, measurement of total beta activity of aerosols from the surface layer of the atmosphere and the precipitations and determination of the radionuclide concentration in various samples.

Since 1992 the concentration of natural and artificiel radionuclides in the aerosols, water, precipitation and sediments were measured in the Nuclear Research Centre of the Latvian Academy of Science using the methods of low background gamma spectroscopy and radiochemistry.

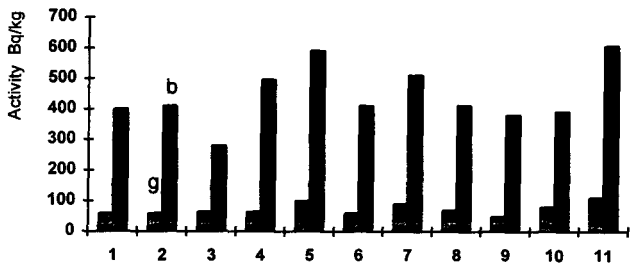
All data obtained from the network are transmitted to the several state institutions and are used for the international programs MORS and HELCOM. The information is obtained from the telecommunication network and forwarded by the direct telephon line BALTMET Riga-Norrkoping. Every day the infomation of gamma dosis rate is transmitted to Sweden.

The measurements of total beta activity of aerosols and precipitations from the reactor control zone were performed by the group of dosimetry of our nuclear reactor. These measurements were performed since 1967.

During the last years a new control system were organized in the Environmental Data Center at Ministry of Environment. In order to measure the gamma radiation the

automatic system was developed in the Environmental Data Center in cooperation with the Swedish Institute of Radiation Protection. Since December 1993 eight Rados radiation detector RD-02 were instaled in Daugavpils, Salaspils, Salacgriva, Talsi, Liepaja, Madona, Rezekne and Ventspils. This system has two signal levels consisting of a warning level at 0.2 μSv/h and an alarm level at 0.5 μSv/h. In order to evaluate the concentration of radionuclides the spectrometrical analysis of aerosol, soil and precipitation samples is forseen in a case of the first signal level. Similar systems are instaled inside of other Baltic states allowing to perform the data exchange. This year these data will be sent to IAEA, too. In order to colect the samples of plants and soils for spectrometrical analysis the control area are organized near the four cotrol stations located at Daugavpils, Salaspils, Ventspils and Liepaja. The results of measurements show that the total gamma activity of the soil samplesnear Daugavpils corresponds to 82-120 Bq/kg. The 1.Fig. shows the total beta and gamma activity near Liepaja.

Fig.1. The total gamma and beta activity of soils from Liepaja district



Number of sample
g-gamma b-beta

1. Bernati	5. Siksni	9. Ziemupe
2. Jurmalciems	6. Barta	10. Grini
3. Nica	7. Dubeni	11. Durbe
4. Rucava-Pape	8. Akmenrags	

The following conclusions can be proposed from the measurements obtained during the last four years.

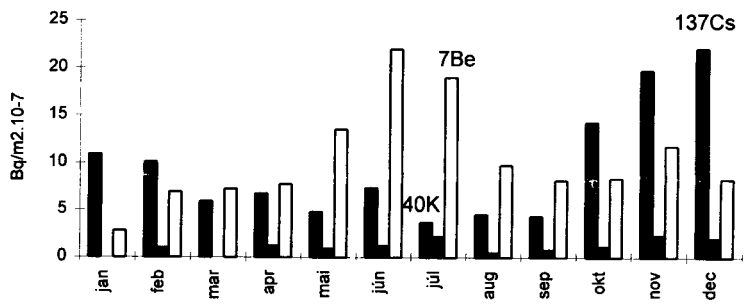
1.The mean values of exposure gamma radiation dosis rate per year were ranged from 9 to 13 μR/h, the minimum was at 6 μR/h (Liepaja) and the maximum at 19 μR/h (Kuldiga, Madona, Kraslava). The mean value of 7 μR/h was obtained from the automatic gamma stations within the last 5 months in 1994.

2.The mean monthly values of the total beta activity during the last four years decrises slowly from $8.8.10^{-5}$ to $6.3.10^{-5}$ Bq/m^3. Newertheless, we can see an insignificant increase of the total beta activity in the summer months.

3.The mean monthly values of radioactive pollution in 1993 in the precipitation vary from 0.5 Bq/m^2 in Ventspils to 2.5 Bq/m^2 in Daugavpils.

4.The concentration of radionuclides in aerosols and precipitation were determined in the Laboratory of nuclear reactions of Nuclear Research Center with the methods of gamma spectroscopy. Only ^{7}Be, ^{40}K and ^{137}Cs were detected in the samples of aerosols and ^{40}K and ^{137}Cs in precipitations. The mean concentration of ^{137}Cs in aerosols was $0.07.10^{-5}$ Bq/m^3 and 0.7 Bq/m^2 in precipitations in 1993. The highest ^{137}Cs concentration in aerosols (three times above the mean concentration) was observed in May and September of 1993. Such concentration would be connected with the air mass movement from Byelorussia during the forest fire and can be caused from Chernobyl accident.
5.The water samples were taken from the four biggest rivers of our republik, the Gulf of Riga and the Baltic Sea. The concentrations of ^{90}Sr and ^{137}Cs were determined. The mean concentration of ^{90}Sr observed for the rivers, the Gulf of Riga and sea water was 25, 46 and 21 Bq/m^3, respectively. The concentration of ^{137}Cs was smaller than 0.03 Bq/l.
6.The concentration of ^{137}Cs is 2.7 Bq/kg in the river sediments, nevertheless, the value of 41Bq/kg in the Gulf of Riga and 97 Bq/kg in the Baltic Sea were measured. These values are practically the same as in 1992.

Fig. 2. Concentration of radionuclides in aerosols of Riga in 1994



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A REGULATORY PROGRAMME TO ASSESS LICENSEES' ARRANGEMENTS FOR MONITORING THE PUBLIC IMPACT OF DIRECT RADIATION FROM CIVIL NUCLEAR SITES IN THE UK

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ABSTRACT

Civil nuclear sites in the UK are licensed by HM Nuclear Installations Inspectorate (NII) which forms part of the Nuclear Safety Division of the Health and Safety Executive. Licence conditions include a requirement for site operators to have an adequate safety case for any operation that may affect safety. NII expects these safety cases to include arrangements that ensure that doses received by members of the public from direct radiation from nuclear sites are acceptably low. In 1993 NII decided that there would be advantages in supplementing existing techniques for assuring itself of the adequacy of the licensees' arrangements by making independent measurements of direct neutron and gamma radiation dose-rates near the perimeter fences of nuclear sites. A five-year programme was established covering all civil licensed nuclear sites in the UK.

This paper presents the measurement protocol adopted and discusses some early results from this monitoring programme. Technical aspects of the interpretation of measured air kerma rates in terms of operational dose quantities are outlined. Appropriate conversion factors are applied, and 'occupancy' data considered, in order to assess the effective doses likely to be received by the most exposed members of the public. These are considered in the light of the current UK statutory public dose limit, the ICRP-recommended limit of 1 mSv per year and other factors, including the basic safety limits and objectives established in NII's published Safety Assessment Principles (SAPs) for Nuclear Power Plants.

INTRODUCTION

The regulatory background to this work has been described by Bacon et al[1] in 1994. The purpose of this work has been to increase NII's confidence in the adequacy of licensee's arrangements for ensuring that direct radiation doses to the public from site operations are acceptably low. In order to achieve this, we have ourselves assessed licensees' arrangements and independently measured gamma dose rates at site perimeter fences. At selected sites we have also arranged for independent measurements of neutron dose rate to be carried out by the National Radiological Protection Board (NRPB), and for surveys of local public occupancy to be performed by the Ministry of Agriculture, Fisheries and Food (MAFF), in each case under contract to NII.

PROTOCOL

Gamma dose rates were measured using Mini Instruments Type 6.80 environmental gamma dose rate monitors. We subtracted background count rates due to instrument noise and cosmic radiation and applied a calibration factor to obtain measurements in terms of the quantity air kerma, Ka, relative to ^{226}Ra. Air kerma rates due to terrestrial background radiation were assessed (by reference to published data[2] and to our own limited background measurements) to obtain an estimate of the air kerma rate from direct gamma radiation from the site. Energy-dependent conversion factors were then applied to interpret measured air kerma rates in terms of the quantities ambient dose equivalent ($H^*(10)$) and effective dose equivalent (H_E). Our contractors, NRPB, measured neutron dose rates at specified locations near the perimeter fences of relevant sites using a Studsvik 2202D dose rate meter calibrated in terms of $H^*(10)$ with respect to ^{241}Am/Be. The response of this instrument is dependent on neutron energy and a knowledge of the neutron energy spectrum is therefore required in interpreting measurements. The highest neutron dose rates encountered in our monitoring programme have been associated with Magnox power stations. We consider that the neutron energy spectra near these stations are such that the Studsvik instrument provides a reasonably accurate response in terms of effective dose equivalent.

Occupancy surveys carried out by MAFF, under contract to NII, have sought to identify those members of the public who comprise the critical group for direct radiation from the sites, evaluate the annual period of exposure to elevated dose rate fields from the site and consider any shielding effect of building construction materials. This work, which has sometimes involved the measurement of gamma dose rates inside and outside occupied buildings, has enabled us to intepret the results of our dose rate measurements in terms of public impact.

PROGRAMME

The following programme has been established:

YEAR 1		YEAR 2		YEAR 3		YEAR 4		YEAR 5	
Bradwell	*nc*	Hinkley Pt	*nc*	Trawsfynydd		Oldbury	*nc*	Wylfa	*nc*
Dungeness	*nc*	Sizewell	*nc*	Heysham	*nc*	Hartlepool	*nc*	Berkeley	
Chapelcross	*nc*	Calder Hall	*nc*	Hunterston	*nc*	Sellafield	*c*	Torness	*nc*
Amersham		Springfields		Cardiff		Capenhurst		RR Derby	
Winfrith		Rosyth		Dounreay		Devonport		Harwell	
								Barrow	

n: Neutron Monitoring (NRPB) *c*: Occupancy Survey (MAFF)

UK PUBLIC DOSE CRITERIA

The statutory dose limit for members of the public in the UK is specified in the Ionising Radiations Regulations 1985 (IRR 85)[3] at 5 mSv per year (in terms of the quantity effective dose equivalent, H_E). More recent guidance from ICRP[4] has recommended a principal dose limit to members of the public of 1 mSv in any single year (in terms of the quantity effective dose, E). NRPB[5] has endorsed this view but has gone further to say that a dose constraint of 0.3 mSv from a single source should be a realistic target for most UK nuclear sites. The ICRP advice is also reflected in NII's Safety Assessment Principles for Nuclear Plants[6] (SAPs). Principle 14 sets a Basic Safety Limit (BSL) of 1 mSv per year and a Basic Safety Objective (BSO) of 0.02 mSv per year for doses to the public. These values are related to risk figures published by HSE in the Tolerability of Risk from Nuclear Power Stations paper[7]. NII expect licensees to have an adequate safety case to show that this BSL is not exceeded and that doses in excess of the BSO are kept as low as reasonably practicable (ALARP).

CASE STUDIES

It is our policy to make our findings available to members of the public via local liaison committee meetings. At most sites we have found that no member of the public receives more than 0.1 mSv per year from direct radiation. However, higher dose rates were found at the three sites which are discussed below.

Nuclear Electric plc, Dungeness Nuclear Power Station (gamma, neutron monitoring and occupancy survey)

This site features relatively high gamma and neutron dose rates at the eastern perimeter within 100 m of two inhabited cottages. These aspects have previously attracted licensee and regulatory attention as part of NII's assessment of the licensee's long term safety review for the Magnox power station at Dungeness "A"[8]. Our independent gamma and neutron dose rate measurements results were broadly consistent with those reported by the licensee. Our findings provided confidence in the licensee's assessment at that time, based on comprehensive measurements and conservative assumptions regarding occupancy and reactor operating conditions, that the most exposed occupant of these cottages might receive an effective dose equivalent of approximately 0.83 mSv y^{-1}. The licensee has since arranged for neutron dose rates to be reassessed following the installation of additional polythene shielding on the roof of one of the reactor buildings. This work has shown that neutron dose rates at the cottages have been significantly reduced. We now believe that the most exposed member of the public is unlikely to receive an effective dose equivalent greater than 0.53 mSv y^{-1}. Considering the licensee's neutron energy spectrum data in the light of the revised radiation weighting factors for neutrons, as recommended by ICRP, we have estimated that this corresponds to an effective dose, E, of approximately 0.64 mSv y^{-1} (0.44 gamma, 0.20 neutron). Our work at Dungeness has highlighted a number of interesting aspects. For example, the gamma energy spectrum at Dungeness "A" is dominated by a component of energy ~ 6.3 MeV (from decay of N^{16} in the external ductwork of the reactor). We therefore needed to allow for the over-response of our instruments to this component and to apply modified factors in converting from Ka to H*(10) and H_E. We found that there was virtually no significant terrestrial gamma dose rate above the shingle surrounding the Dungeness site and that gamma dose rates measured in some buildings near the site were higher than those measured immediately outside. We consider that the elevated dose rates are probably attributable to natural radioactivity in the building construction materials.

Amersham International plc, Amersham Laboratories (gamma monitoring only)

As at Dungeness, occupied buildings are located very close to part of the perimeter fence of the Amersham site. The results of our gamma dose rate measurements were broadly consistent with those reported by the licensee (there being no neutron component). As a result of our work, the licensee has completed an assessment of the dose to the public using revised occupancy data and is making further improvements to shielding. We are currently examining the licensee's assessment that no member of the public is likely to receive an effective dose equivalent exceeding 0.2 mSv y^{-1} from direct radiation from the site.

British Nuclear Fuels plc, Chapelcross Nuclear Power Station (gamma, neutron monitoring and occupancy survey)

As with the previous two case studies, occupied buildings are located very close to the nuclear site. The radiation field features gamma and neutron contributions, including a high-energy gamma component of ~ 6.3 MeV. NII's independent assessment has broadly confirmed the licensee's safety case. This shows that the most exposed members of the public are residents of a nearby farm who might receive an effective dose equivalent of up to 0.12 mSv y^{-1} (0.10 gamma, 0.02 neutron). Key features of this work were the identification of the appropriate terrestrial background, the inclusion of a neutron component, and assessment of additional members of the public. The licensee is currently carrying out further investigations, which will include neutron energy spectrum determination, at this site.

SUMMARY

We have found that licensees have employed a variety of approaches and technical methods in seeking to ensure that doses received by members of the public from direct radiation from nuclear sites are acceptably low. Our independent assessment has, in some cases, prompted licensees to review and refine their arrangements for assessing the impact of direct radiation from their sites and to seek means of reducing off-site dose rates, in accordance with the ALARP principle. We have found that members of the public living close to nuclear sites have received effective dose equivalents (and effective doses) significantly less than 1 mSv per year as a result of direct radiation from the site. This work has provided useful supplementary information in support of discussions on ALARP. Nothing has emerged which has reduced our confidence in the adequacy of the licensees' arrangements for meeting public dose limits. No significant deficiencies, requiring regulatory enforcement, have been identified.

The views expressed in this paper are those of the author and do not necessarily represent those of HSE.
The author wishes to acknowledge the assistance of his colleagues Mr I F Robinson, Mr J S Griffiths, Dr D N Simister and Miss E A McCullough, and the co-operation of the licensees, in the course of this work.

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ENVIRONMENTAL MEASUREMENTS AROUND

FRENCH NUCLEAR POWER PLANTS

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Electricité de France generates 75 % of its electricity in nuclear power plants with pressurized water reactors (PWR). Theses plants comprise 34 units of 900 MW and 20 units of 1300 MW, the first of which was connected to the grid in 1977, and the last in 1993. Three other units of 1400 MW are under construction.

The environmental measurements are performed in two complementary ways :

1. Routine regulatory monitoring carried out by the operator according to a programme and procedures drawn up by the Ionizing Radiation Protection Office (OPRI), which is the State monitoring authority in France. The OPRI checks the results against those obtained with its own samples.

   Around each power plant, the following are monitored :

   - Ambient γ-radiation (continuously) at 8 points around the site within a radius of 5 km ;
   - Aerosols in the air (once per day) at 4 points within a radius of 1 km ;
   - Rainwater and groundwater (monthly) ;
   - Surface waters (whenever there is a liquid radioactive discharge) ;
   - Milk and vegetables (monthly) at 2 points in the area close to the site.

   The plant has an off-site laboratory, two specially-adapted vehicles and e team of three chemists.

   These measurements are quite separate from those conducted on radioactive wastes.

2. Annual and ten-yearly radioecological measurement campaigns around the sites in order to improve scientific knowledge of the environmental impact of the plants.

   A ten-yearly campaign consists of "radioecological photographs" which are compared with the "initial zero point". About 40 samples are taken and various analyses performed (total β, α and γ spectra, strontium, carbon-14, free and organic tritium). The choice of samples and the places where they are taken depend on the zero point and the special features of the region. Items sampled include drinking water, irrigation water, ground moss, vegetables, fruit, field crops, field soil, humic gley soil, meadow grass, milk, sewage sludge, wine, sediments, water or marine plants, fish or shellfish.

The annual campaigns around each site enable a picture to be built up of the radiation situation in time and space from γ-spectrometry measurements on 27 samples selected as being the most suitable indicators (sediments, bryophytes, fish, moss, fruit, drinking water, milk, wine, soil).

The programm involves around 600 samples and 800 measurements per year and is conducted, at our request, by the Institute for Radiation Protection and Nuclear Safety (IPSN) which has established the methodology for the sampling and measurements and has the capacity to ensure the continuity indispensable for this type of measurement..

The results of all these measurements are published and communicated to the public through the most appropriate media.

AN ASSESSMENT OF THE RADIOLOGICAL SIGNIFICANCE OF CONSUMING WILD FOODS COLLECTED NEAR THE SELLAFIELD NUCLEAR FUEL REPROCESSING PLANT.

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1. Dose estimates and modelling of food pathway

Extensive monitoring of conventional agricultural produce in the vicinity of the BNFL Sellafield plant is undertaken, by both the operator and the Ministry of Agriculture, Fisheries and Food, to determine levels of radioactivity and doses[(2)] arising to the consumer. Monitoring is also undertaken[(3)], albeit less extensively, for market garden and domestic produce. By contrast, few data exist with respect to levels of radioactivity in 'wild foods' (e.g. hedgerow fruits, field mushrooms etc.) or associated consumption habits. It has been postulated that such foodstuffs could contribute an appreciable radiation exposure dose to groups of high level consumers, potentially including members of the existing identified critical group for local agricultural produce. This paper assess the actual radiological significance of wild foods collected near Sellafield.

2. The Habit Survey

A previous survey of the consumption of locally produced foods had provided useful information for the assessment of doses from food produced close to the Sellafield nuclear plant[(1)]. In the current project, the habit survey had the main objective of identifying wild foods which could contribute to the ingestion dose. The survey was carried out during the spring and summer of 1994 by CN Research (Carlisle, UK). A total of 72 households were surveyed, and data collected for 181 individuals; comprising 3 infants (taken to be up to 2 years old), 21 children and 157 adults. Information was obtained on a total of 49 foods, including 9 species of fungi, 11 herbs, 16 fruits, 12 types of game, as well as nuts and honey; collected from areas near to Sellafield.

Table 1 Consumption Rate Data for some Key Foods

Food	Number of Consumers	Consumption rate (kg/year)			
		Mean	Max	Ratio max/mean	97.5th percentile
Venison	23	8.16	134.4	16.5	72.3
Blackberry	129	1.45	10.24	7.06	9.72
Honey	49	1.29	6.72	5.21	6.27
Elderberry	6	2.59	6.7	2.59	6.35
Pheasant	72	1.44	7.17	5.00	5.08
Fungi	64	0.57	4.98	8.74	2.06
Mallard	29	1.27	5.82	4.58	5.82
Nuts	30	0.49	2.56	5.22	2.56
Goose	10	1.08	2.69	2.49	2.59
Rabbit	7	0.86	1.34	1.59	1.34

In many cases, the maximum consumption rate of a food was much greater than the average value for all consumers of that food. In the most extreme case the maximum consumption rate for venison, by a game keeper, was 134 kg per year: 16.5 times the average value of 8 kg per year for all 23 consumers. This difference between the extreme rate and the average is, for wild foods, greater than for most conventional agricultural products, and requires special attention in the assessment of radiation doses from the consumption of wild foods.

In this survey, the consumption of wild fungi, although including a variety of species, was fairly low, being naturally limited by availability. The consumers of all wild foods were mainly adults, with generally insignificant consumption by infants and children.

3. Measurements of radioactivity in wild foods and consumption doses arising.

The information collected in the habit survey was used as a guide in the collection of samples of fruit, nuts, wild game (such as venison, rabbits, pheasant, duck and goose), as well as honey and fungi. Samples were measured for gamma emitters, actinides (plutonium and americium) and, on selected samples, ^{14}C and ^{129}I.

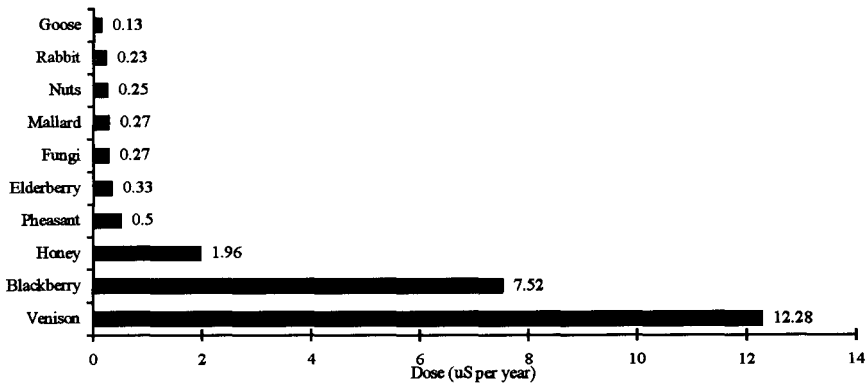
The dominant gamma emitter was ^{137}Cs; the highest concentrations being found in some samples of honey. Pollen analysis identified these honey samples as containing significant proportions of honey derived from heather. In West Cumbria heather grows on upland organic soils which received significant deposition from the Chernobyl accident in 1986. Table 2 compares two samples of honey from the area close to Sellafield.

Table 2 Comparison of caesium content and primary pollen source of upland and lowland honey from West Cumbria

	Upland honey	Lowland honey
^{137}Cs (Bq/kg)	254	21
Pollen Species:		
Calluna (heather)	71%	
Trifolium (clover)	9.2%	48%
rowan, hawthorn and cherry		20%

Figure 1 shows the contributions to the dose from wild foods consumed at the 97.5 percentile rates, assuming maximum measured radionuclide concentrations for each food, demonstrating the relative importance of blackberries and venison. In this assessment the upland honey samples were omitted, in order to avoid the inclusion of contributions from Chernobyl deposition through the heather-honey pathway.

Figure 1 Contributions to dose from different wild foods for the 97.5th percentile consumption rates



The dose from wild food can be compared to the critical group dose from locally produced agricultural products (e.g. milk, vegetables and meat); where the highest such dose was 24-54 μSv/year (Committed Effective Dose) to an infant in 1994, of which 7-36 μSv/year is from milk[(2)]. For an adult, more relevant for comparison to this project, the dose from local agricultural products is calculated to be 10-24 μSv/year committed effective dose (CED) of which 3-9 μSv/year is from milk.

The contribution to ingestion dose by individual radionuclides present in the wild foods is presented in Table 3. It can be seen that the dominant nuclide overall is ^{137}Cs, which may derive in this locality from Sellafield emissions, the Chernobyl accident and/or weapons test fallout.

Table 2: Committed effective dose based on 97.5th percentile consumption by an adult

	Effective dose (μSv/y)						Total	Collective dose*
Food	^{14}C	^{129}I	^{134}Cs	^{137}Cs	^{239}Pu	^{241}Am		μmanSv/y
Venison	0.00	0.00	0.45	11.82	0.00	0.00	12.28	282
Blackberry	0.57	0.49	0.22	5.99	0.11	0.24	7.62	983
Honey	0.08	0.00	0.12	1.67	0.10	0.003	1.96	96
Elderberry	0.00	0.31	0.00	0.02	0.01	0.00	0.33	2
Pheasant	0.03	0.17	0.00	0.30	0.00	0.00	0.50	35.8
Fungi	0.00	0.00	0.01	0.27	0.00	0.00	0.27	18
Mallard	0.00	0.00	0.00	0.27	0.00	0.00	0.27	8
Nuts	0.00	0.00	0.00	0.25	0.00	0.00	0.25	8
Goose	0.00	0.003	0.00	0.13	0.00	0.00	0.14	1
Rabbit	0.00	0.00	0.01	0.22	0.00	0.00	0.23	2
Consumption dose by isotope (μSv/y)	**0.68**	**0.97**	**0.81**	**20.9**	**0.21**	**0.24**	**23.8**	
%	**2.88**	**4.08**	**3.38**	**87.79**	**0.89**	**1.10**	**100**	

** Collective dose is calculated only for the consumer group for each foodstuff: i.e. the 23 consumers of venison each receive just over 12 μSv per year, or collectively 282 μSv per year. This is a very narrow definition of collective dose, used only to illustrate the relative importance of each of the foodstuffs analysed.*

Considering both collective and individual dose, venison and blackberries dominate the consumption pathway, contributing 282 and 983 μmanSv per year respectively to the group of consuming individuals. Blacberries are of particular importance in calculating collective dose, because of their ubiquitous distribution and consumption.

4. Conclusions

The project has highlighted the importance of blackberries, which are easy to collect and are consumed by large numbers of people. Game (e.g. pheasant) is also important because of the concentrations of radiocaesium in game reared close to Sellafield and fed on locally grown grain. Venison is an example of unusual consumption patterns by a very small number of individuals. The radioactivity in honey was very variable, with some samples containing radiocaesium from Chernobyl through the soil-heather-honey pathway, whilst other samples contained very little anthropogenic derived radioactivity.

Overall, the extreme doses from wild foods are comparable with doses to the critical group for consumption of conventional agricultural produce from close to the Sellafield site, but the 97.5th percentile of the individual effective doses from wild foods are generally lower. All doses are well below applicable dose limits, even when the doses are assessed additively.

This study was funded by the Ministry of Agriculture, Fisheries and Food (Food Science Division). The authors are grateful to David Weir of English Heritage, London, for the pollen analysis of honey samples.

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THE KGE COMPUTER SYSTEM

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A computer system has been developed to assist crisis management teams of the Command and Control Centres (PCC) of PWR power plants in evaluating releases and the radiological impact of such releases on the environment in the event of an accident.

This industrial application, known as GEEE (Gestion de Effluents Et de l'Environnement - Waste and Environmental Management) is supported by the KGE system, in accordance with encoding of the basic systems of nuclear power plants.

Each EDF Nuclear power plant (CNPE) is equipped with the KGE computer system.

This computerised tool is also installed in the technical control room of the national crisis centre.
Environmental specialists can link up with the KGE computer of the plant in which the accident has occurred and that of a neighbouring site or sites.
Certain parameters relating to all sites can also be consulted under normal conditions.

The evaluation of the radiological impacts on the environment is made on the basis of the condition of the installation in three stages, both in terms of diagnostic evaluations and prognoses.
These three stages are as follows:
* evaluation of the source; that is, the level of activity released into the environment for each major group of radionuclides (rare gases, iodines and cesiums)
* evaluation of the transportation and dispersion of this level of activity into the environment
* calculation of doses for the various forms of transport at various distances from the release source.

In order to monitor releases in real time (diagnostics), the KGE computerised system is connected to:
* the computer and data processing system (KIT) of the unit which provides thermo-hydraulic measurements and various activity measurements from the plant radiation monitoring system (KRT)
* to the environmental monitoring system (KRS) which provides data on wind speed and direction, the level of precipitations and the atmospheric stability index.
* to the radioactivity measurement and monitoring station at 1 and 5 km from the plant (RSN radiation meter beacons).

Therefore, the evaluation of the source is obtained through continuous radioactivity measurements from the plant radiation monitoring system (KRT) (beta total gas) and the ventilation rate from the nuclear auxiliary building ventilation system (DVN). Evaluation of the transport of waste is obtained through the continuous acquisition of meteorological measurements for the site from the environmental monitoring system. The evaluation of the dispersion of waste into the environment is obtained through the use of a "burst" diffusion model, which takes into account the weather variables at the time of the release.

In order to forecast the consequences (prognosis), the evaluation of the source is obtained using predetermined values based on the type of accident, the transport evaluation is obtained using average weather data supplied by the inter-regional division of the French weather service. The evaluation of the dispersion of activity into the environment is obtained using a monogram model from the IPSN.

The dose conversion factors set out in ICRP 29 are used for calculate the full organism dose, the thyroid dose equivalent and the dose rate, whether for the prognosis or diagnostic evaluation.

The application enables the results of the evaluation of the impact of gas releases (environmental message , impact map for the entire organism or thyroid, etc.).

The level of precision achieved for the areas affected by the release enable the targeting of activity measurements in the field on the one hand and the establishment of priorities for the implementation of measures to protect the population (confinement, evacuation) on the other.

The Automatic Radiation Monitoring Distributed System SRM-256C

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The Controlled and Supervised Areas (CSA) of UNK proton accelerator builded in Protvino (IHEP, Russia), borrows the significant area (more than 15 sq. km.) [1]. Submitted in work [2] results of accounts show, that by major factors, influencing to a radioecological conditions in region at the expense of work UNK are: 1) the output of pulsing radiation on day-time surface; 2) the radionuclides receipts with drinking water; 3) the pollution of radioactive air from system of ventilation. At normal mode of UNK operation the equivalent dose rates on day-time surface of pulsing radiation will changes in limits from 0.5 mcSv/hours near overmines buildings up to 0.1 mcSv / hour on the CSA border [2]. The average equivalent dose rates per year due to internal irradiation at use (intakes) of drinking water will not exceed 5 - 50 mcSv/years [2]. The Maximum equivalent dose rates on day-time surface, caused by pollution of radioactive air does not exceed 0.01 ... 0.03 mcSv/hour in limits of CSA, and the average equivalent dose per year caused by internal irradiation does not exceed 0.05 mcSv / years [2]. At emergency-free operation the maximum degree of the UNK influence in limits of CSA is estimated in terms of average equivalent dose per year at levels, not exceeding 0.05 ... 0.10 mSv/year [2]. For maintenance of integral environmental monitoring of total external radiation levels in limits of CSA network of passive monitor stations [3] will be developed.The UNK influence on radioecological condition of environment , according to resulted estimations, will insignificant and can be determined only by long measurements, as before UNK start , and during its operation. One of differential monitoring components [4] of environment in UNK region will be The Automatic Monitoring Distributed System (SRM-256C). Its based on network of Sensors, recording radiation parameters of the environment, connected through Input/Transmitting Stations (IOS) via the communication line (the switched telephone communications or the non-switchable cable communications of radial organization) with managing computer (IBM PC/AT compatible).The IOS executes reception, accumulation of the information from sensors and transmission of data in Communication Line (CL) on inquiry from managing computer. The data are received by managing computer through specialized modem. The program shell of SRI-256Å routes cyclic inquiry of IOS , reception and decoding of the information from them, executes the control of reliability of distributed sensors , archiving of the received information in specialized database, mathematical processing and graphic representation of data with binding on the controlled area , recall and notification under given telephone number list in case of dose rate limits exceed. The first developed variant of IOS (IOS1) consists of following main functional units: power supply unit (PSU); unit of the interface with communication line (ICLU); the timer unit (TU); the RAM unit (RAMU) (RAM with capacity 4 Kbytes); unit of address logic formation (ALFU); the unit of counters (CU); unit of the reliability control of detectors (RCDU); the unit of the universal shifting register (USRU). The IOS1 permits to accept, to store and to transfer data in line on inquiry from two detectors (are used modified industrial detector BDMG-08R for gamma - radiation

detecting (the measuring range of equivalent dose rate are 0.1 mcSv/hour ... 100 mcSvhour) and the DPN neutron detector (the measuring range of equivalent dose rate are 2.5 mcSv/hour ... 25 mSv/hour .) for registration of neutron radiation [3]. The signals from sensors are recounted with CU and through installed timing interval are recorded in higher word (with address W:2047) of RAMU. ALFU and TU controls the work of USRU and organize arithmetic shift of whole data file in RAMU to lower addresses. The changeable range of a timing cycle of record in unit the RAMU makes Tc = 1 minute ... 4 hours, thus minimum data storage time (without loss) makes (1 mines o 2048) 2048 minutes. With detecting of an asynchronous inquiry in CL by ICLU a cycle of sensors reliability testing (the testing radioactive Sr/Y sources deliverance are organized to a sensitive volume of detectors) is initiated, and then ALFU and RCDU give out in consec utive CL data file from RAMU and results of detectors diagnostics . The data are broadcasted to CL with installed rates of transfer: in telephone line - up to 2048 baud; to coaxial cables - up to 9600 baud. Taking into account experience of IOS1 operation the second variant (IOS2) was developed. Basis of IOS2 is the one-crystal microprocessor 80Å31AI of clone Intel MCS-51 (USA). The software of IOS2 is realized with TASM for MSDOS (Speech Technology Inc.) cross-tools.The IOS2 carries out all functions of IOS1, however the IOS2 use less of chips and is simple in set-up. The amount of served sensors may reached 8. The updating of functional opportunities for IOS2 is reduced to reprogramming of ROM (capacity of 4 Kbytes) The specialized modem is developed for ISA system bus (AT-bus) of IBM PC/AT - compatible computer. The modem cyclically recalls the IOS, accepts data, together with has an opportunity to give out the speech information on telephone lines. The software of system ÅDI-256Å is realized in environment of programming shell Borland C ++ and permits to serve up to 256 of IOS.

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CENTRALIZATION OF ENVIRONMENT MONITORING MEASUREMENTS

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According to French legislation, the operators of nuclear installations are required to install a certain number of monitoring devices to ensure that such installations have no radioecological impact on the natural environment and the surrounding population.
For this, the most recent nuclear power plant, that is the CHOOZ power plant, which belongs to the N4 series, has been equipped with a new system which centralizes the measurement data from sensors installed in the environment around the installation.
This system gathers radiation meter, meteorological and physico-chemical data measured in the environment together with a certain number of parameters related to the nuclear units themselves, such as power and continuous stack release activity measurements.

All these data are then centralized and exploited using a supervision mode which performs the surveillance, statistical, maintenance and communications functions.

The surveillance function enables the operator to display data concerning the environment around the power plant under the form of different screens for each measurement field:

- Radiation meter,
- Meteorological analysis,
- River monitoring,
- SODAR measurements.

Data are displayed under various forms, either under the form of a map showing the location of sensors in particular or under the form of tables of graphs.
The colour of the name of each sensor changes according to its status.
All maps show an indicator, which shows the direction and strength of the wind as well as atmospheric stability.
In the event of a pre-set value being exceeded and since the parameters of each sensor can be modified, a local signal is transmitted automatically to the control room of the power plant, where an alarm is generated.
Any screen can be printed out at any time.
In addition, each screen displays a band, which indicates the actual status of the network and maintenance operations on the system, in real time.

The statistic function is used for the management and storage of data as well as to print out graphs and tables showing particularly interesting points.

The CPU also transmits data to the various users on site and transmits data to the various organisations concerned; such as the national crisis centres or the national meteorological service.

The system can also act as a server and transfer certain processed data to other applications; such as the computerised waste release management system.

This system has been in service in the CHOOZ power plant since the summer of 1995.

ENVIRONMENTAL STRAY RADIATION MONITORING AT THE NSRL

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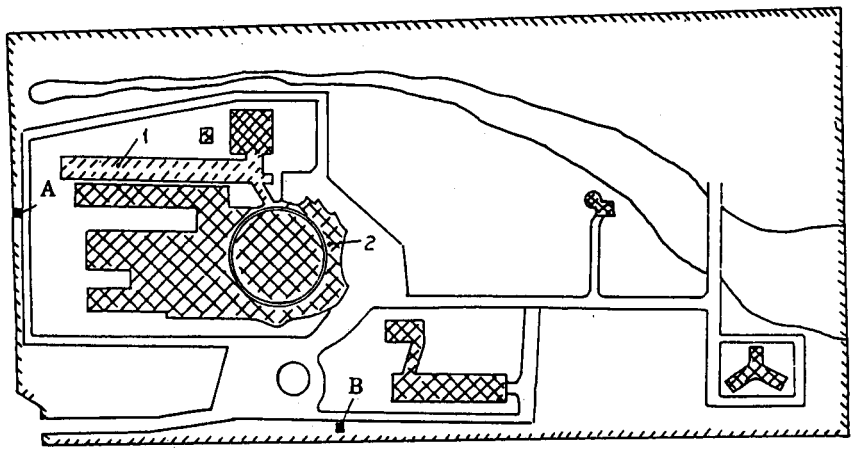
THE NSRL AND ITS ENVIRONMENTAL RADIATION FIELD

The first dedicated synchrotron radiation light source in China is fixed in the National Synchrotron Radiation Laboratory (NSRL). A 200 MeV electron linac and an 800 MeV storage ring are its main equipment. The linac works as the injector and the ring provides VUV synchrotron light for various kinds of optical experiments. Fig.1 is the sketch map of the NSRL.



1. Tunnel 2. SR Hall A. E.M. Station A B. E.M. Station B

Fig.1

The environmental radiation field around it is distinctive and its variation is related closely with the operation states of the synchrotron radiation machine. Its displays are as follows:

1. It is a mixed field. Not only bremsstrahlung but also neutron exists in it. Both of them have wide spectrum. These two components distribute quite differently. The former is directive while the latter is isotropic.
2. It is a prompt field with small duty factor(1×10^{-7}). The high instantaneous value and low average of the radiation dose are the most distinctive differences compared with the steady field.
3. Since the linac is in an underground tunnel covered with three-meter thick soil, no observable radiation dose caused by it can be recorded during the environmental monitoring. The additional environmental stray dose mainly comes from the injection of the storage ring. It skips every time the electrons from linac are injected into the ring. As mentioned above, although the peaks can be observed clearly on the curve recorded, the accumulated dose is limited.

It is necessary to monitor, record and analyze such a characteristic radiation field. To avoid the shortcomings of the discontinuous monitoring and TLD method, an intelligent environmental monitoring system has been set up for this radiation field.

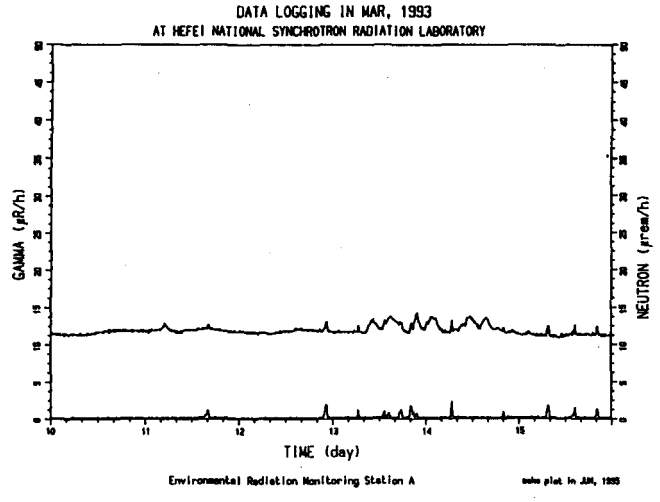
ENVIRONMENTAL RADIATION MONITORING

The intelligent environmental radiation monitoring system includes environmental monitoring stations, control computer and several peripheral devices. In Fig. 1, the points marked A and B are the locations of two environmental monitoring stations. Station A is at the west border of the NSRL area where is close to the linac. Station B is near the south border where is about 104 meters far from the injection point and exactly towards the injection direction. There are detectors and data acquisition device in every station. The gamma detector is a 8.5 liter volume pressurized ionization chamber filled with argon. Its sensitivity is $1.03\times10^{-4}A/(C\cdot kg^{-1}\cdot h^{-1})$. The neutron detector is a BF_3 tube in polythene moderator. Its sensitivity is $(17.0\pm0.3cps)/(n\cdot cm^{-2}\cdot s^{-1})$. The data acquisition device is a microprocessing unit with M146805E CPU. It can work well with rechargeable batteries when power supply fails. The environmental radiation data can be stored in its RAM and transferred to the control computer's disk once a week. In past years, the plenty of data provided scientific basis for us to analyze the field's characteristics and make tables or charts. The software for special research purpose is programmed during the work, such as the program for calculating additional dose or injection time.

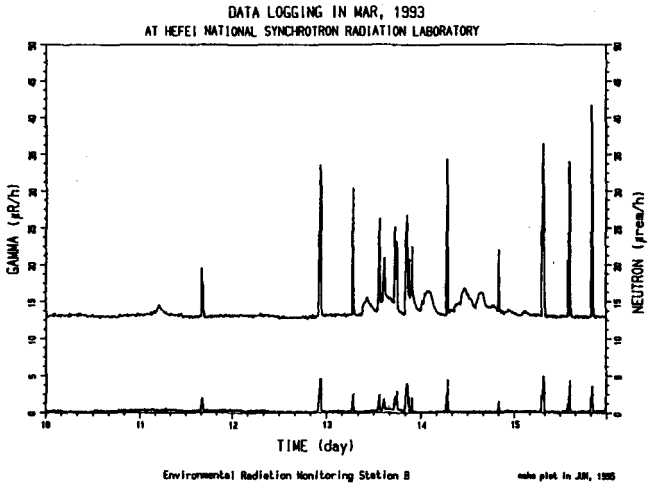
THE ANALYSIS FOR THE NSRL ENVIRONMENTAL STRAY RADIATION FIELD

By the environmental radiation monitoring system, we have gotten a series of monitoring results that reflect the field's characteristic and stray radiation level correctly. At the same time, some interesting things attract us very much and make us to think deeper about certain phenomenon. For instance, we have new idea about the relationship between rain and gamma natural background. Fig.2 includes two charts plotted by the system. They are gamma and neutron environmental radiation dose rate curves in the same six days. Here is the explanation:

1. The chart marked (a) is from monitoring station A and (b) is from station B. In every chart, the upper curve represents gamma dose rate and the lower one represents neutron. Their relative values are shown in the left and right of the chart respectively. The X axis is the date.
2. The increase of gamma dose rate is caused by the bremsstrahlung. Compare chart (a) with chart (b), we can see clearly that the gamma radiation level recorded by station B is much higher than that by station A during the injection period of the ring. This is because of the different location of these two stations. A is at the side of the injection direction while B is toward it. Strong directive is shown here. In the meantime, the neutron dose rate is isotropic. The difference is only because of the different distance between the stations and injection point.
3. On the gamma curve, all the peaks that no responses on the neutron curve are caused by the rain. It seems that the heavier the rain is, the higher the peak rises. The radiation level falls to background as soon as the rain stops. We do not think this phenomenon accords perfectly with the traditional theory about the background increase caused by the rain. So a further research is going on.
4. Neutron radiation level is not affected by rain. Its total dose equivalent is much less than that produced by gamma. It increases only during the injection period. So we can take it as a symbol to recognize if the gamma increase is caused by the rain.
5. From the charts we can see the additional dose is limited. Sometimes the gamma increase caused by rain is higher than that by machine operation. The natural background dose is about 1000 μSv per year, the recorded increase dose caused by the machine operation is usually much lower than 50 μSv even if at the station B.



(a)



(b)

Fig.2

CONCLUSION

We began to cumulate natural background data in 1987. Since then, the environmental radiation monitoring system has worked well for eight years. It helps us and other researchers a lot in many aspects. It is not only a guarantee for the radiation safety, but also a useful tool in the scientific research. The data acquisition device and processing software are improving so that we can study the field in a better way.

THE ARTIFICIAL RADIOACTIVITY IN BECHET-DABULENI AREA, FIVE YEARS OF SURVEY

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INTRODUCTION

The Bechet-Dabuleni area is placed in the southern part of Romania, in the Danube zone, near the Nuclear Power Plant from Kozlodui (Bulgaria). Here there is a very important agricultural area, with small and also big farms where for irrigation the Danube water is used.

The purpose of the study was the long term surveillance of the artificial radioactivity for the main foodstuffs. It has been investigated the radioactive content of the following foodstuffs from this area : vegetables (potatoes, carrots, onion, etc.), fruits (apples, pears, apricots, peaches) and cereals (wheat, barley). It has also been determinated the radioactive content of the irrigating water from the Danube River.

MATERIAL AND METHOD

Global beta-measurements were performed in order to obtain initial data on the contamination level of the samples. The determination of the Cs-137 and Sr-90 content from the water samples was done by coprecipitation from 50 l water followed by the chemical separation. Cs-137 was concentrated on APM and measured with a beta counter as Cs-hexacloroplatinate and Sr-90 was precipitated as carbonate and after the separation was measured as yttrium oxalate after the radioactive equilibrium.

The beta-measurements were done with a low level anticoincidence beta counter with 14% counting eficiency for Sr-90+Y-90 standard source. For the determination of the specific Cs-137 content from vegetables were used the ashes obtained in a thermoregulating furnace at the temperature of 450°C. The measurements were performed using a multichannel analyser system (4096 channels) type Canberra 40, with a high resolution Ge(Li) detector and with a multichannel analyser system (1024 channels) with a NaI(Tl) detector. The calibration of the equipment was made using internal standards, volume standards from IFA-Magurele and using also the intercomparison samples from activities organized by IAEA-Vienna.

RESULTS

The data obtained during 1989-1993 are synthesized in this study. The results pointed out the low levels of the artificial radioactivity in this area in comparison with the rest of Romania. The values for the radioactive content in the irrigating Danube water are presented in the figure 1.

The Cs-137 content was in the range 0.0013-0.0045 Bq/l and the Sr-90 content in the range 0.0066-0.0121 Bq/l, for the whole period.

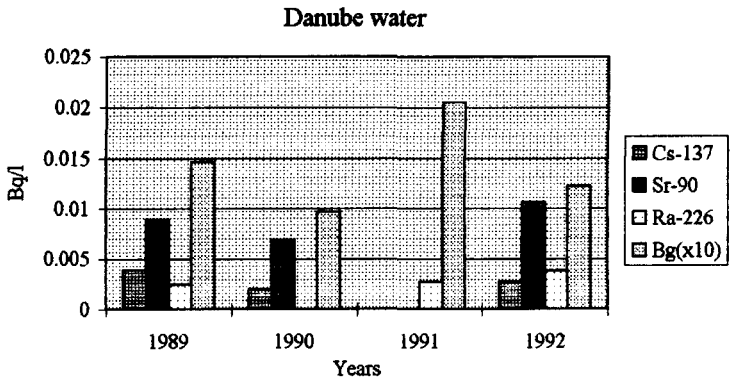

Figure 1. The radioactive content in the Danube water during 1989-1992.
Bg - global beta

In general the radioactive content for fresh vegetables was in the range 0.2-0.6 Bq/kg in Cs-137 and in the range 0.05-1.1 Bq/kg in Sr-90. Particularly Cs-137 had the maximum value (0.6 Bq/kg) in potatoes and minimum value (0.2 Bq/kg) for rest of vegetables (bean,onion,fruits, etc.). For Sr-90 high values were in bean and carrots and low values in potatoes.

The results obtained for vegetables are presented in the figure 2.

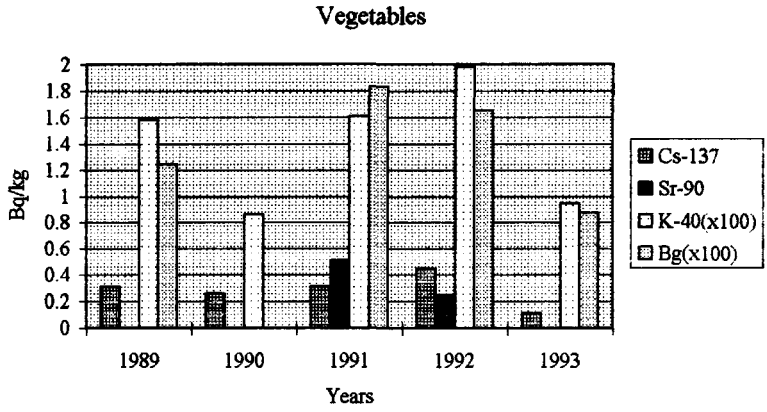

Figure 2. The radioactive content in vegetables.
Bg - global beta

The data obtained for fruits and cereals were frequently near the detection limit.

CONCLUSIONS

The artificial radioactivity content in the foodstuffs analysed from the investigated area (Bechet-Dabuleni) was at a low level and it has had an unsignificant variation in time.

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ON-LINE LOW CONCENTRATION STACK ACTIVITY RELEASE MONITORING SYSTEM FOR NUCLEAR REACTORS

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INTRODUCTION

Monitoring of radioactive effluents at Nuclear Installations is extensively reported, detailing aims & designs of Health-Safety Program[1].The references[2,3] describe techniques of Nuclear Data Processing with computer[4] based On-Line monitors.Using these as guidelines and with state-of-the-art technology,a monitoring system is successfully implemented which provides greater functionality,reliability,operator-acceptance and meets the stringent requirements of Health-Physicist & Regulatory Board.Although sophisticated techniques of Nuclear Data Processing & Analysis are used in laboratory instruments,porting these to field instruments calls for special expertise.

SYSTEM CONFIGURATION AND HARDWARE FEATURES

The system consists of main channel and hot standby channel.Each channel comprises of one Air Particulate Beta monitor,one Iodine-131 monitor and one Gross Gamma (mainly Ar^{41}) activity monitor as shown in fig 1. Activity released is measured through detector assemblies consisting of photomultiplier tube and scintillator (plastic for Beta Particulate monitor, NaI(Tl) for Iodine-131 & Gross Gamma monitor) and are shielded by lead rings. Beta and Iodine-131 activity is accumulated on easy to remove and replace filter paper[a] and charcoal cartridge[b] respectively,whereas two litres of chamber surrounding the detector assembly is used for Gross Gamma activity monitor.The direction of flow and position of filter is carefully selected to improve detection sensitivity. The hardware is identical in three monitors. Iodine-131 monitor functions as 128 channnel,16 bit wide Multichannel Analyser performing energy-spectrum processing, while only gross counting is performed in other two monitors.The design of microcomputer is based on popular,8 bit microprocessor with standard VME back plane (single height,six layer,five slots) and plug in EURO boards facilitating ease of testing, and maintenance.A hand held terminal provides user friendly man-machine interface for setting up the important system parameters. Liquid Crystal Display essentially displays the integral release,release rate of the activity in Engineering units.Pulse height spectrum is continuously displayed on CRO, in XY mode,enabling easy calibration and precise marking of Iodine-131 window. Comprehensive report of activity released is made available through printer and the system also annunciates alarms in case the integral release,release rate exceed the set alarm limits. Logarithmic analogue output is provided for remote logging.

SYSTEM SOFTWARE DESIGN FEATURES

System software is designed and developed adopting modular approach for ease of testing and modification. It offers adequate degree of freedom for incorporating advanced statistical & analytical techniques to improve the performance of the system. Integral release and Release rates are computed on the basis of gross counts obtained within a set period of time in Beta Particulate and Gross Gamma monitors, while in Iodine-131 monitor,net peak area under the Iodine-131 peak is used. Intrinsic interference due to the background is minimised in Iodine-131 monitor by incorporating spectrum analysis technique,wherein Trapezoidal Compton background area is determined employing End Point Averaging Technique (EPAT) .Software critically balances the contradictory requirements of fast response time and acceptably good statistical stability along with accurate determination of lower detectable limit of 10^{-11} µCi / ml by automatic selection of optimum acquisition time interval.Additional feature of the software common to all the three monitors is the sliding averaging technique used for providing stable display information of activity release with acceptable time constant and update time. System software also supports interfacing of the monitors to IBM compatible PC,where energy spectrum can be analysed by powerful spectrum analysis techniqes for radioactive isotopes other than Iodine-131.

NET PEAK AREA COMPUTATION

From figure 2, Total peak area and its standard deviation using simple Total Peak Area (TPA) method[5] are computed as

$$\text{Total Peak Area (TPA)} = \sum_{i=K1}^{i=K2} C_i - \{ [(K2 - K1 + 1) / 2] * [C_{K1} + C_{K2}] \} \qquad (1)$$

$$\sigma TPA = \left\{ \sum_{i=K1}^{i=K2} C_i + [(K2 - K1 + 1) / 2]^2 * [C_{K1} + C_{K2}] \right\}^{1/2} \quad (2)$$

Net peak area and its standard deviation using EPAT[3] are computed as

$$\text{Net Peak Area (NPA)} = \sum_{i=K1}^{i=K2} C_i - \{ [(K2 - K1 + 1) / 2] * [(B1 / N1) + (B2 / N2)] \} \quad (3)$$

$$\sigma NPA = \left\{ \sum_{i=K1}^{i=K2} C_i + [(K2 - K1 + 1) / 2]^2 * [(B1 / N1^2) + (B2 / N2^2)] \right\}^{1/2} \quad (4)$$

Where C_i is the counts of i^{th} channel

K1, K2 are cursor locations at left and right valley point of Iodine-131 peak, respectively

$\sum_{i=K1}^{i=K2} C_i$ is the Gross Area (GA) under the Iodine-131 peak

$[(K2 - K1 + 1) / 2] * [C_{K1} + C_{K2}]$ is the Trapezoidal Compton background Area (TA)

$[(K2 - K1 + 1) / 2] * [(B1 / N1) + (B2 / N2)]$ is TA using EPAT

B1 is the sum of counts over N1 channels including K1 and B2 is the sum of counts over N2 channels including K2. As seen from equations (2) and (4) EPAT gives considerable improvement in standard deviation than simple TPA method. An improvement of 22 % in standrad deviation was observed in the actual system by keeping N1 = 2 and N2 = 4 while the maximum difference in the value of net peak area, computed by two methods, was 6% only, which is acceptable. As EPAT and TPA methods assume linear background under the Iodine-131 peak, it may introduce error due to non linearity of compton background caused by presence of high energy isotopes. This error may be reduced by using Step Pedestal[4] or Quadratic Baseline interploation method for estimating the compton background under such conditions.

AUTOMATIC SELECTION OF OPTIMUM ACQUISITION TIME INTERVAL

Criteria[6] used for selecting new acquistion time interval aims at maintaining the Fractional Standard Deviation (FSD) constant throughout the monitoring range of the system. New acqusition time interval is estimated as : Let PA1 be the current peak area obtained after subtracting Trapezoidal Compton background Area (TA1) from Gross Area (GA1) under the Iodine-131 peak with acquisition time interval of T1. Let T2 be the optimum acquisition time interval for achieving the desired FSD_d, then peak area PA2 and its Variance will be as follows :

$PA2 = GA2 - TA2$ and $\sigma^2 PA2 = GA2 + [(K2 - K1 + 1) / 2]^2 * [(B1' / N1^2) + (B2' / N2^2)]$

But $GA2 = (T2 / T1) * GA1$, $B1' = (T2 / T1) * B1$, $B2' = (T2 / T1) * B2$ & $PA2 = (T2 / T1) * (GA1 - TA1)$

$\therefore \sigma^2 PA2 = \{ T2 / T1 \} * \{ GA1 + [(K2 - K1 + 1) / 2]^2 * [(B1 / N1^2) + (B2 / N2^2)] \}$

And $\sigma^2 PA2 / PA2^2 = (FSD_d)^2 = (T1 / T2) * (FSD_o)^2$ Where $(FSD_o)^2 = \sigma^2 PA1 / PA1^2$

i.e $(FSD_o)^2 = \{ GA1 + [(K2 - K1 + 1) / 2]^2 * [(B1 / N1^2) + (B2 / N2^2)] \} / \{ GA1 - TA1 \}^2$

$$\text{OR} \quad T2 = (FSD_o / FSD_d)^2 * T1 \quad (5)$$

Equation (5) gives the optimum acquisition time interval for the FSD desired. Figure 3 shows the estimated absolute fluctuations in computed activity release through stack as a function of activity accumulated on charcoal filter, for fixed FSD of 1 % and variable acquisition time interval as predicted by equation (5). The graph indicates that though response time improves with increase in accumulated activity , the absolute statistical fluctuations in estimated integral release also increase, thereby worsening the minimum detectable limit. Thus in the adopted fixed filter scheme, either for loaded or fresh filter paper / charcoal cartridge, sufficiently large counting times are required. Based on these findings, system was tested with various large counting time intervals, results of which are plotted in figure 3. The results clearly point out that counting time of 60 minutes as the optimum fixed counting time resulting in acceptable statistical performance for measuring low concentration release rates (50μCi / day or 10^{-11}μCi / ml). Large counting time is also necessary for applying decay correction.

SYSTEM CALIBRATION

Iodine-131 deposited on the charcoal layer closest to the detector, makes maximum contribution to the counts accumulated, while contribution due to subsequent layers of charcoal goes on reducing. Since effective depth of

charcoal for accumulating Iodine-131 is estimated to be 5 mm, a calibrated Iodine-131 source on filter paper was moved away from detector surface by 5 mm in steps of 1 mm and the average efficiency computed was found to be 10.32 %. Another important parameter is the effect of interference due to Gamma background.This was estimated by placing a Co^{60} source of 47 nCi (corresponding to $Argon^{41}$ release rate of 700 Ci / day) along with 3.75 nCi of Iodine-131 source and error due to interfering background was observed to be of the order of 20 %.System minimum detectable limit was determined using Iodine-131 source of 40 pCi under the clean background conditions with acquisition time interval of 60 minutes and it was found to be 60 µCi / day. Similar calibration of Air Particulate Beta monitor was performed using $Strontium^{90}$ & other Beta sources,deposited on filter paper,giving an average efiiciency of 20 %.The detector sensitivity of Gross Gamma monitor was determined by measuring the countrate from chamber filled with calibrated Ar^{41} sample and was found to be 1cps / pCi / ml.

CONCLUSION

Low concentration stack activity releases of 50 µCi / day were measured with the system installed at nuclear power plants in India.For composite Beta-Gamma detection using phoswich detector and simultaneous processing of multiple peaks,the eight bit microprocessor design can be easily upgraded to advanced 16 / 32 bit processor design.A networking configuration using network controller board on the same VME back plane can be provided.

FIGURES

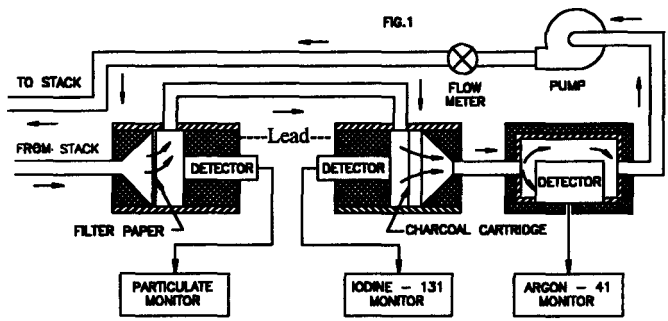

Figure 1

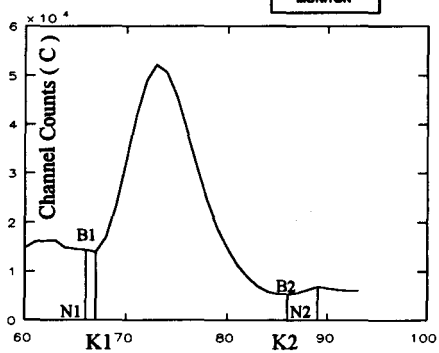

Figure 2 Channel Number (K)

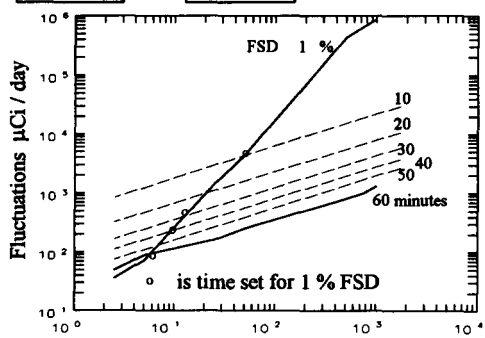

Figure 3 Filter Activity nCi

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a *A 50 mm diameter,1 micron pore size,Millipore cellulose fibre fixed filter*

b *12217 filter cartridge of Merlin Gerin ,57.7mm diameter, 26.8mm height consists of vegetable charcoal (coconut shell) impregnated with TEDA (Triethylene diamine) encapsulated in thermoplastic material*

IRPA9
1996 International Congress on Radiation Protection
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Abstract No. 90323 Eingang-TAK
Receipt
Author 20266 27. Juli 1995
Acceptance
Mini-Presentation

PAPER TITLE Environmental Monitoring in Qinshan NPP

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) Radiation Protection in the Environment

ABSTRACT (See instructions overleaf)

This paper presents the environmental monitoring in Qinshan Nuclear Power Plant, the first one in mainland China.

According to the ALARA principle of the ICRP, the environmental monitoring program is optimised during the operation of NPP. Using estimated dose equivalent received by the public with mathematical model and results of environmental monitoring, the quality of environment around Qinshan NPP is evaluated. Continuous effort is made to mitigate the environmental impact due to the operation of NPP.

ENVIRONMENTAL RADIOACTIVITY SURVEILLANCE OF THE ROMANIAN CANDU REACTOR OF CERNAVODA; NORMAL OPERATION AND ACCIDENT SCENARIOS

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INTRODUCTION

On behalf of the Ministry of Waters, Forests and Environmental Protection and according to the Romanian legislation, the environmental radioactivity surveillance in the Cernavoda area is performed by the Environmental Radioactivity Laboratory (ERL) of the Institute for Environmental Research and Engineering.

ERL is the scientific supervisor of the National Environmental Radioactivity Surveillance Network (NERSN). NERSN comprises 46 manned stations located in Romania's major cities. These stations perform gamma dose rate and global beta measurements. Routine global beta measurements are performed upon air, atmospheric deposition, water, soil and vegetation samples. Four of these stations also perform gamma spectroscopy on environmental samples.

Cernavoda Environmental Radioactivity Station is part of NERSN. Beside the standard gamma dose rate and global beta measurements equipment common to NERSN stations, Cernavoda Station is equipped with HPGe gamma spectrometer, Tritium (H-3) in air monitor, Iodine monitor, noble gases monitor and liquid scintillation analyzer. Global beta and gamma spectrometric measurements were performed in the Cernavoda area by ERL and Cernavoda Station starting with the 80's **(Figure 1, 2)**

In the framework of emergency planning and intervention ERL is the coordinator of the dose assessment group, one of the various groups acting as part of the Central Commission for Nuclear Accident (CCANCOC).

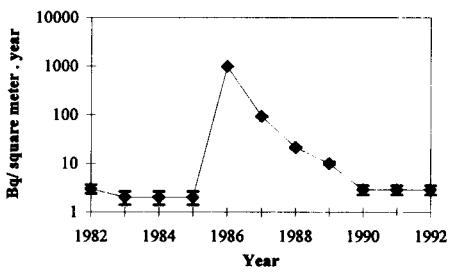

Figure 1. Cs-137 in atmospheric deposition at Cernavoda .

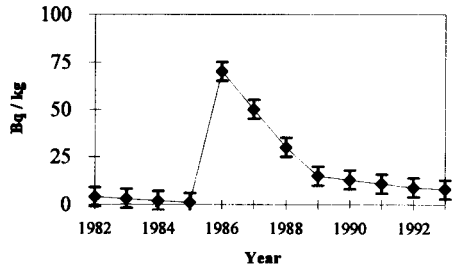

Figure 2. Cs-137 in surface soil at Cernavoda.

ENVIRONMENTAL RADIOACTIVITY SURVEILLANCE PROGRAM

The first unit (operational later in 1996) of the Cernavoda Nuclear Power Plant (CNPP) is powered by a CANDU 600 reactor. The town of Cernavoda (22043 people of which 8125 children) is 2 km away from the site of the reactor. Except of the flat, 20 km wide, island of the Danube to the West , the terrain around the CNPP is rough with hills up to 70 m. The first unit will have the liquid discharge in to the Danube - Black Sea navigation channel.

Gaseous effluents

Normal operation source term is: $3.4\ 10^{14}$ MeV Bq / year Noble gases, $9.4\ 10^{7}$ Bq / year Iodine, $9.4\ 10^{8}$ Bq / year Particulates, $5.7\ 10^{12}$ Bq / year C-14 and $3.5\ 10^{14}$ Bq / year H-3. The stack is 50.3 m height and

2.34 m in diameter. In between 3.9 m^3/s in winter time and 11 m^3/s in summer time is evacuated from the stack at 40– C. Particle size is a mean 0.3 – m with a maximum 1.0 – m.

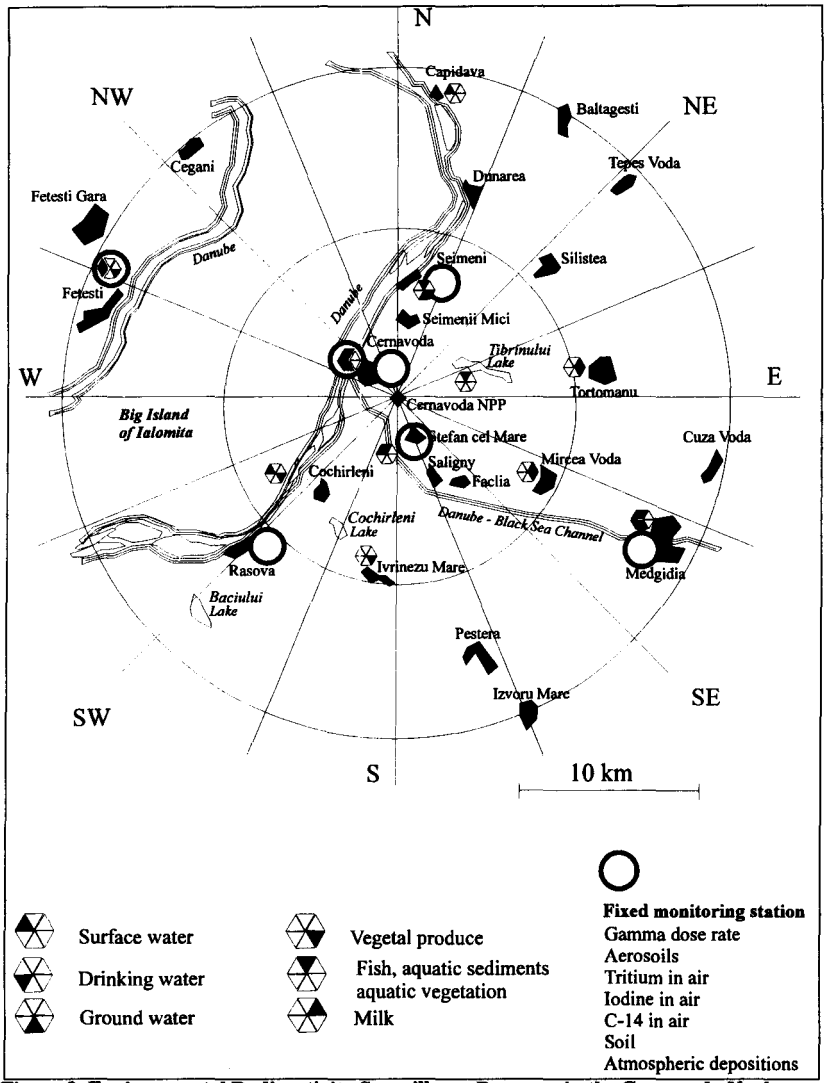

Figure 3. Environmental Radioactivity Surveillance Program in the Cernavoda Nuclear Power Plant Area.

Liquid discharge

Medium global activity of liquid discharge in the Danube - Black Sea navigation channel is 7.7 10^6 Bq/m^3 half of which consists of I-131. Tritium (H-3) will constantly be discharged at the rate of 1.23 10^{10} Bq/m^3. Other nuclides such as Co-60, Mo-99, Ce-144, Cr-51, Cs-134, Cs-137, will be discharged at rates representing 3 to 15% of the total global activity.

Location of the monitoring stations

Population distribution in the area and food habits, wind pattern and meteorological conditions, source term data were employed to establish the type, frequency and location of monitoring measurements to be performed in the environmental radioactivity surveillance program. Calculations using *CAP 88 PC* code (USA EPA Office of Radiation Programs) point the relevant wind sectors affected by continuous emission occurring in normal

operation of CNPP. Last, practical reasons such as security and access have also been considered in positioning the fixed off-site monitoring stations. In the fixed monitoring stations gamma dose rates are measured by RADOS intelligent detectors and radio transmitted to the central computer of Cernavoda Environmental Radioactivity Station. Gamma spectroscopy and global beta measurements are weekly performed upon atmospheric aerosoils, soil and atmospheric depositions samples. In the sampling points presented in **Figure 3**, milk samples, surface, drinking and ground water are weekly sampled and subjected to gamma and liquid scintillation analyses while fish, aquatic sediments and aquatic vegetation, vegetal produce are sampled and analyzed quarterly or according to the season.

External exposure and inhalation is a critical pathway for the people in the town of Cernavoda. Water from the Danube - Black Sea navigation channel is used for irrigation and drinking purposes so water ingestion is a critical pathway for people in Constanta (250 000 inhabitants). In order to control these two pathways emphasis is given to monitor H-3 concentration in air at Cernavoda (Berthold Tritium in air monitor) and H-3 concentration in the Danube - Black Sea navigation channel (H-3 content of water samples from the channel to be determined by means of liquid scintillation spectroscopy).

ACCIDENT SCENARIOS

In case of emergencies CNPP applies detailed procedures. Notification of competent authorities is common to all the procedures. These notifications contain source term, site meteorological conditions and projected doses for the population which may be affected if the releases would be done at various moments of time and in the given meteorological conditions. The computer code *Cernavoda Emergency Response Projection* (*CERP*) developed by CNPP mainly considers small or large loose of cooling (*LOCA*) accident type with or without maintaining the intervention capability of the emergency cooling systems. *CERP* also considers various situations of the reactor containment. According to *CERP* a small *LOCA* accident with intact containment (a release time of one hour starting the moment the accident is produced) will have a released activity of 7.5 10^{10} Bq Iodines (I-131) and 2.3 10^{12} Bq noble gases (Xe-133). A large *LOCA* accident with failure of emergency cooling systems and intact containment (a release time of one hour starting the moment the accident is produced) will have a released activity of 4.7 10^{14} Bq Iodines (I-131) and 1.6 10^{16} Bq noble gases (Xe-133).

Using these information provided to CCANCOC by CNPP, ERL performs first evaluation of population dose. Field dosimetry is employed and, with confirmed wind pattern the source term is modified and dose evaluation is again performed. *PC COSYMA* and modified *RASCAL* computer codes are used by ERL for accident consequences assessment. While *COSYMA* code may best provide countermeasures and long term effects, simple Gaussian models are best suited for the initial stages of the emergency response. There is an accepted agreement (*i.e.* same order of magnitude for thyroid dose) in between the computer codes used by CNPP and the computer codes used by ERL in accident consequences assessment.

CONCLUSIONS

For more than one decade the Environmental Radioactivity Laboratory (ERL) has been performing environmental radioactivity measurements in the Cernavoda area. This constitutes a reference base line for environmental radioactivity surveillance of the future to be operational CANDU 600 reactor of Cernavoda Nuclear Power Plant (CNPP).

An environmental radioactivity surveillance program has been designed and will be implemented before the commissioning of the CNPP.

Emergency preparedness for CNPP is developed in the legal national framework and ERL plays a key role here as coordinator of the dose assessment and monitoring group. There is an accepted agreement in between the computer codes used by CNPP and the computer codes used by ERL in accident consequences assessment.

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EMERGENCY PLANNING FOR THE ROMANIAN CANDU REACTOR OF CERNAVODA; THE ENVIRONMENTAL APPROACH

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INTRODUCTION

Cernavoda Nuclear Power Plant (CNPP) emergency exercise "AXIOPOLIS" was carried out in August 1995. Environmental Radioactivity Laboratory (ERL) of the Institute of Environmental Research and Engineering is responsible for the dose calculations and monitoring of the radioactivity in the environment following an off- site event at CNPP. The response actions were performed at local and national level by various Civil Protection structures coordinated by the Central Commission for Nuclear Accident (CCANCOC). CCANCOC is scientifically advised by groups of experts such as the dose assessment and radioactivity monitoring group where ERL plays a key role.

SCENARIO OVERVIEW

The scenario was designed by CNPP Emergency Planning Section. It is considered that the reactor has been operated at full power for about 10 months and the pre-equilibrium core conditions allow for maximum fission product released to containment. Reactor safety systems are poised and no maintenance is underway. Tritium levels in the reactor building are higher than normal and pressure heat transfer pump alarms for very high vibrations.

Vibrations cause a damage in the heat transfer balance piping which separates. A large loss of coolant (large LOCA) accident takes place. Shut down systems 1 and 2 trip. Containment is sealed. Emergency cooling systems fires. Dousing commence as a result of the containment pressure rise.

Scenario accident is built to test various emergency procedures both on-site and off-site. There is a contaminated casualty in the plant and the rescue and medical teams have to deal accordingly. The Main Control Room licensed staff demonstrated their ability to correctly evaluate the radiation incident, the emergency condition of the plant and the amount of radioactivity in the containment. Emergency Operation Center and the Command Unit were activated by the Shift Supervisor in a timely manner. For the off-site response of the authorities up to the phase of sheltering some area residents, the accident source term as given by *CERP* (*Cernavoda Emergency Response Projection*) code had to be modified.

INTERVENTION LEVELS AND CALCULATIONS

Intervention levels for sheltering as recommended by the regulatory body (National Commission for Nuclear Activities) are of in between 3 to 30 mSv for whole body and/or 30 to 300 mSv for thyroid, lung, skin where these values represent the external plus the committed dose through intake during the first 24 hours. The source term provided by CNPP and the given dispersion conditions did not led to calculated values within the above ranges. However for the need to check the response of local authorities sheltering was recommended and simulated in the town of Cernavoda. Calculations were performed by ERL representatives in CCANCOC using *PC COSYMA*, *RASCAL* and Gaussian dispersion models. **Figure 1** presents a *RASCAL* evaluation in the conservative condition of fast (10 minutes) release.

COMMUNICATIONS AND FIELD DATA

Communications from CNPP to CCANCOC were performed via facsimile and telex lines. Source term, meteorological site conditions and *CERP* projected doses, as given by CNPP, are presented as "fill in", previously prepared, forms. Field radiation monitoring is performed by teams of Cernavoda Environmental Radioactivity Station (CERS) and of Constanta County Civil Defense. The teams run a previous established route taking gamma dose rate readings in several locations focused on the centerline of the radioactive plume. The teams send the information to CERS by radio. CERS sends the information by public telephone to ERL. Environmental sampling is performed by the survey teams at request from ERL and gamma spectroscopy is performed on location at CERS in Cernavoda.

Cernavoda Environmental Radioactivity Station (CERS) plays an important role in the off-site response. When activated by Constanta County Civil Defense or ERL, the station switches on emergency program. Atmospheric aerosoils are sampled for 10 minutes, three times per hour and inspected for global beta activity. Frequent readings of the gamma dose rate meters are performed. The Iodine monitor and the Noble Gases

monitor are powered. Gamma spectroscopy is performed upon air, atmospheric depositions, soil samples collected by the field teams. All the above data are sent to ERL.

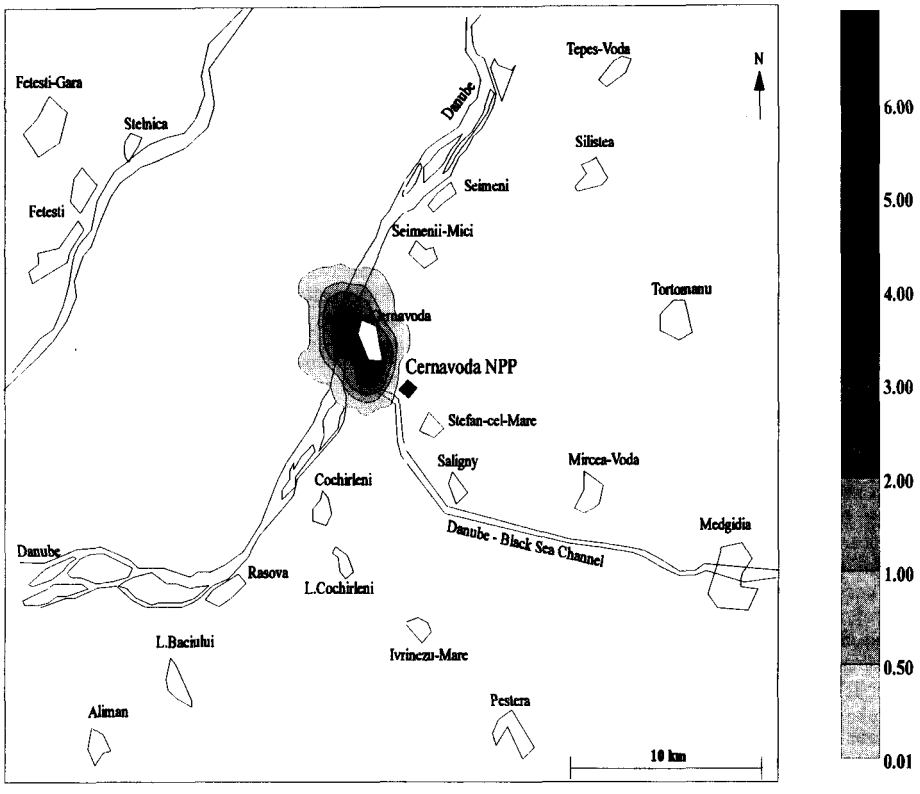

Figure 1.
Thyroide dose (mSv) 10 minutes after the start of the stack level emission.
Source term: I-131: 50 TBq, Xe-133: 200 TBq, release time: 10 minutes.
Meteorological data: 8 m/s wind blowing from SE toward NW , stability class D.

CONCLUSIONS

"AXIOPOLIS" 1995 tested emergency procedures both on-site and off-site CNPP. The capacity of the NERSN stations and field teams to carry out the emergency environmental radioactivity survey was tested. Dose predictions were performed by ERL within CCANCOC and as a back up, at the laboratory location. Countermeasures were issued by the dose assessment group coordinated by ERL within CCANCOC.

While radioactivity measurement equipment of NERSN stations (Ministry of Waters, Forests and Environmental Protection) generally complies with the requirements, there is a need for more consistent meteorological field data in the area. Communications is to be improved at all levels as reliability of public telephone lines is always low. The exercise has to be prepared and attended to each year as it was proved that only the players who practiced the procedures before, could give a satisfactory response.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Tritium in Danube water around the Romanian CANDU reactor of Cernavoda

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MAJOR SCIENTIFIC TOPIC NUMBER ..4.. (see page 7)

ABSTRACT (See instructions overleaf)

Liquid scintillation Tritium measurements were performed by the Environmental Radioactivity Laboratory in the Cernavoda area starting with 1994. The sampling points were upstream and downstream Cernavoda on the Danube and downstream Cernavoda on the Danube - Black Sea navigation channel where the nuclear power plant discharge will take place. Measurements will continue after the commissioning of the Cernavoda nuclear power plant in the second half of 1995.

Danube and Danube - Black Sea navigation channel water are used as irrigation and drinking water sources. Based on liquid scintillation measurements and aquatic dispersion modeling, the paper presents dose values and dose predictions for the Tritiated water consumption in the influence area of the Romanian CANDU reactor of Cernavoda.

Radioactive liquid discharges in relation with the phosphogypsum industry in European Countries

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I - The phosphated fertilizers industry

Phosphate contained in phosphated rocks being not directly assimilated by vegetal plants, a chemical treatment is needed to produce phosphate fertilizers. Most often, the ore is reacted with sulfuric acid to produce phosphoric acid. The by-product of the reaction is a calcium sulfate (phosphogypsum) which is separated by precipitation. Typically, the production of 1 tonne of phosphoric acid needs the extraction of 3 tonnes of ore and creates 4 to 5 tonnes of phosphogypsum. Most of the uranium contained in the raw material is associated to the phosphoric acid and the main part of radium remain in the phosphogypsum. In the average, phosphogypsum contains 14 % of the ^{238}U contained in the ore, 30 % of the ^{232}Th and 80 % of the ^{226}Ra.
If compatible with economical constraints, phosphogypsum is used in building materials or back-fill and road-base materials, but, most often, it is discharged to sea or disposed on tailing piles. Globally, 14 % of phosphogypsum produced is used, 58 % is stored on tailing piles and 28 % is discharged to sea.
The annual world production of phosphate ore turns around 130 Mtonnes. Ore importations by european countries has been divided by a factor 2 between 1980 and 1994 balanced by phosphoric acid importation. It was about 8 Mtonnes in 1994 (about one half from Morocco) [1]. It is obvious that the phosphoric acid and phosphogypsum productions have the same evolution. Producers expect that this trend will continue in the future (a 10 % decrease in the european countries phosphated fertilizer production is expected between 1993 and 1997).

II - The radioactive ore content

Natural radionuclides concentration in ore depends on its origin: minerals extracted from sedimentary soils have the highest uranium content (1,500 $Bq.kg^{-1}$ is a typical value). This kind of soil produces about 80 % of the world production. When extracted from volcanic areas, its uranium content is about 70 $Bq.kg^{-1}$. Table 1 gives natural radionuclides content for the main producing countries.

Table 1 : Natural radionuclides concentrations for different ores ($Bq.kg^{-1}$) according to Baetsle [2]

	^{238}U	^{226}Ra	^{232}Th	^{40}K
Morocco	1,500 - 1,700	1,500 - 1,700	20 - 30	10 - 200
United-States				
Florida	1,500 - 1,900	1,600 - 2,100	16 - 59	
Idaho	1,850	1,800	30	
Wyoming	2,300	2,300	10	
Togo	1300	1,200	110	<100
Tunisia	590	520	92	
Israel	1,500 - 1,700			
Jordan	1,300 - 1,850			
former USSR	44 - 90	30 - 70	78	

Nevertheless, some ores have higher concentrations: the content of ores extracted from South Carolina and from Tanzania is about 5,000 $Bq.kg^{-1}$ for ^{238}U and ^{226}Ra. At the opposite Chilian ore is particularly poor in natural radionuclides: 40 $Bq.kg^{-1}$ for ^{238}U and ^{226}Ra, 30 $Bq.kg^{-1}$ for ^{232}Th. The ^{232}Th concentrations also vary, from 10 $Bq.kg^{-1}$ (Wyoming, USA) to 110 $Bq.kg^{-1}$ (Togo). The 30 $Bq.kg^{-1}$ value is representative of concentrations in Moroccan ore.

Globally, we can conclude that concentrations in phosphogypsum are around 210 Bq.kg^{-1} for ^{238}U, 1200 Bq.kg^{-1} for ^{226}Ra and 10 Bq.kg^{-1} for ^{232}Th, so that a plant having a 300,000 t.y^{-1} capacity produces each year residus containing 10 GBq of ^{238}U, 300 GBq of ^{226}Ra and 1 GBq of ^{232}Th.

III - Discharges from european countries and environmental concentrations

In 1984, the industry of the european countries produced 17 Mt phosphogypsum, about 7 Mt of which was dumped. According to the mean concentrations, the annual discharge was 8.4 TBq for ^{226}Ra, 1.5 TBq for ^{238}U, 0.07 TBq for ^{232}Th. The decrease of the production by itself leads to an annual discharge about two times lower in 1993. In fact the waste management has been changed so that the decrease of dumping is now much higher. Globally, the european countries discharges in 1993 were 2,1 TBq for ^{226}Ra, 0.5 TBq for ^{238}U, 2.2 TBq for ^{210}Po. Phosphogypsum discharges in european waters are mainly due to industrial practices in the UK, in France (these two countries have ceased the water dumping in 1992) and in The Netherlands.

III.1 - France

About 3 Mt has been dumped in the Seine estuary, but the corresponding radiation exposures have not been estimated [3]. Several measurement campaigns were carried out along the Channel coast, showing ^{210}Po concentrations in the range 130 - 770 Bq.kg$^{-1}_{dry}$ in mussels, 0.2 - 2.7 Bq.m^{-3} in filtered water, 20 - 130 Bq.kg^{-1} in suspended matters and 10 - 120 Bq.kg$^{-1}_{dry}$ in sediments. The highest concentrations are observed in the Mont Saint Michel bay. The ^{210}Pb concentrations in sediments are in the same order of magnitude as the ^{210}Po ones. They are much lower in mussels (maximum: 17 Bq.kg$^{-1}_{dry}$).

III.2 - United Kingdom

From 1954 to 1992, the Whitehaven (Cumbria) phosphoric acid production plant dumped phosphogypsum in the Solway Firth. Since 1993, operations have been limited to the purification of imported phosphoric acid and an effluent processing plant is operating. The evolution of annual discharges is given in table 2.

Table 2 : Evolution of the mean annual discharge (TBq.y^{-1}) from UK

	^{238}U or ^{234}U	^{234}Th	^{230}Th	^{226}Ra	^{210}Pb	^{210}Po
from 1972 to 1992	0.42	0.60	0.31	0.57	0.36	0.36
1993	0.13	0.13	0.026	0.0011	0.01	0.0027

The ^{226}Ra global inventory in the area under control decreases exponentially (half-life: 70 days) from 1.46 GBq in 1992 to 0.06 GBq in December 1993 [4]. The ^{210}Po concentration in marine products strongly varies with marine species: it is about 260 Bq.kg^{-1} in winkles and 70 Bq.kg^{-1} in lobsters. These concentrations decrease with distance from the source point by about a factor 10 within 30 km. The ^{210}Pb concentration in the Eastern Irish Sea sediments varies from 5 to 14 kBq.m^{-2}, the mean deposition rate has been estimated by McCartney et al. about 200 Bq.m^{-2}.y^{-1}.

Before 1992, annual dose received by critical group (high consumption of marine products caught in the Saltom bay) due to phosphogypsum discharges was 0.27 mSv for adults and 0.38 mSv for children. Since 1993, it is estimated to be 0.05 mSv.y^{-1} (the natural background is around 0.3 mSv.y^{-1}) [5].

III.3 - The Netherlands

Three ore processing plants are still operating in The Netherlands, dumping the phosphogypsum produced in the Rhine estuary. Up to 1993, the annual ^{210}Po and ^{226}Ra discharges (about 2 Mt) were about 1 TBq [6] [7]. Since 1993, a 20 % reduction has been observed, due to a change of the ore origin.

Discharges are responsible for a 100 Bq.kg^{-1} increase in the edible part of mussels and shrimps in a 50 - 100 km radius around the pipe. Table 3 gives water and sediment concentrations along the North Sea coast, and the total sediment activity in a 2,500 km^2 area and a 5 cm deep layer [8]:

Table 3 : Natural radionuclides concentrations in water and sediments in the North Sea, along the Dutch coast

	Water concent. (Bq.m^{-3})	Sediment concent. (Bq.kg^{-1})	Global activity in sediments (TBq)
^{226}Ra	1.5	20	3.5
^{210}Pb	1.0	20	3.5
^{210}Po	1.0	20	3.5
^{231}Pa	0.001	1	0.17

According to the radiological half-life, Van der Heijde et al. consider that the equilibrium concentration for ^{210}Pb and ^{210}Po will be about 15 TBq after several decades with discharges (assumed to be constant). Locally much higher concentration have been measured [9] (see table 4) :

Table 4 : Natural radionuclides concentrations as a function of distance from the source

	water concentration ($Bq.m^{-3}$)			concentration in suspended matter ($Bq.m^{-3}$)		
	^{210}Po	^{210}Pb	^{226}Ra	^{210}Po	^{210}Pb	^{226}Ra
North Sea (30 km outside)	0.7	0.8	5.3	86	119	
Nieuwe Waterweg	3.3	10	19	478	444	300
Schelde estuary	6.1	6.2	68	258	218	145

For these three radionuclides, concentrations in Nieuwe Waterweg sediments are usually above 150 $Bq.kg^{-1}$ as far as 5 km from the pipe. Concentrations of 800 $Bq.kg^{-1}$ for ^{210}Po, 1000 $Bq.kg^{-1}$ for ^{210}Pb and 600 $Bq.kg^{-1}$ for ^{226}Ra are measured in harbours. The ^{210}Po concentration in mussels is in the range 20 - 50 $Bq.kg^{-1}$ (2 to 3 times higher than in non-affected areas). There is no significant increase of the ^{226}Ra in fishes, molluscs and shell-fishes.

Effective dose received by the Dutch population due to consumption of marine products caught in the concerned area is about 20 $\mu Sv.y^{-1}$ and can reach 200 - 300 $\mu Sv.y^{-1}$ for the critical group, mainly (90 %) due to ^{210}Po [9]. Phosphogypsum discharges contribute for about 50 % of this dose. Annual collective dose for the Dutch population due to marine products consumption is 170 man.Sv, phosphogypsum discharges contributing for less than 1 %.

Conclusion

Since 1980, phosphogypsum discharges in North European waters are considerably decreasing. Natural radionuclides concentrations are significantly enhanced 30 km around the pipe. Effective dose received by people consuming marine products near operating plants are usually 20 $\mu Sv.y^{-1}$, but it can be 10 times higher for some people. This dose is mainly due to ^{210}Po.

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CHANGES IN LIQUID RADIOACTIVE WASTE DISCHARGES FROM SELLAFIELD TO THE IRISH SEA: MONITORING OF THE ENVIRONMENTAL CONSEQUENCES AND RADIOLOGICAL IMPLICATIONS

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INTRODUCTION

In January 1994, British Nuclear Fuels plc (BNFL) were granted revised authorisations for disposal of radioactive wastes from their Sellafield site in Cumbria, UK, including the discharge of liquid effluent to the Irish Sea. The revisions took account of a continuing Government commitment to review authorisations regularly, as well as of new Sellafield plants and changes to existing operations. The new plants of prime importance were the Thermal Oxide Reprocessing Plant (THORP) for the reprocessing of oxide fuels, and the Enhanced Actinide Removal Plant (EARP) for improved treatment mainly of waste streams from the older, Magnox, plant. Revision of the liquid authorisation is described in reference 1. Many of the radionuclide discharge limits reduced significantly, notably for Cs-137, Ru-106 and the actinides, to reflect operation of EARP. Increases in limits, of relatively low radiological significance, were granted for tritium and I-129 which would be produced by operation of THORP. An increase for C-14 covered diversion to sea of Magnox gaseous scrubber liquors, so as to reduce the overall radiological impact of these waste streams. Increases for Sr-90 and especially Tc-99 were to allow discharge, after phased processing through EARP, of liquors which have been stored on site for an optimised period to allow radioactive decay. Tc-99 is not treated by EARP, and Sr-90 is treated less effectively than the actinides or radiocaesium.

DISCHARGES IN 1994

At the time of writing this paper, discharge data and results of environmental monitoring are not complete for 1995, and comment is restricted to 1994. The trends in discharges from 1990-1994 are shown as the histograms in Figures 1-5. Following granting of the revised authorisation, active commissioning of THORP began but no active chemical separation took place in 1994. Diversion of the Magnox scrubber streams resulted in increased discharges of C-14 from a pre-existing level of about 2 TBq y^{-1} to about 8 TBq in 1994. EARP began active commissioning, and discharges of alpha emitters, radiocaesium and Ru-106 decreased as a result. Feeds of decay-stored liquors were also processed in EARP, and increases were observed in discharges of Sr-90 and especially Tc-99.

ENVIRONMENTAL MONITORING

MAFF carry out a comprehensive marine monitoring programme in the UK which is kept under review to take account of any changes in such factors as the discharges themselves, the habits of critical groups, and environmental conditions. The results of this monitoring are published annually, most recently for 1994 (2). Selected results of this monitoring are presented in Figures 1-6 to indicate trends in data as a result of changes in Sellafield discharges. In addition, significant monitoring of radionuclides in sea water, especially for Tc-99, has been carried out and the results will be published in due course.

Figure 1 presents concentrations of Ru-106 observed in winkles (*Littorina littorea*) from Nethertown, 5km north of Sellafield, which are of significance for the critical group of seafood consumers. Reference 2 considers sampling and analytical uncertainties; for the purposes of this paper representative ±10% error bars have been ascribed to concentration data. Figure 1 shows that Ru-106 concentrations in recent years have fluctuated mostly in line with the discharges, and the reduced discharges of 1994 have been reflected in lower concentrations.

Figures 2 and 3 present data for Pu-239/240 and Am-241 in winkles from Nethertown. Decreases in concentrations were observed in 1994 in response to reduced discharges. This would appear to indicate a component of uptake due to current discharges, in addition to the influence of the historic inventory in the Sellafield environment.

Figure 4 presents averaged data for C-14 in cod (*Gadus morhua*) and plaice (*Pleuronectes platessa*) sampled from near Sellafield. Concentrations increased in 1994 by about a factor of 2 compared with pre-existing levels. The higher levels of discharges (about a factor of 4) had probably not continued long enough in 1994 for their full effect on fish uptake to be observed.

For Sr-90, (Figure 5), discharges increased from pre-existing levels in 1993 because of changes in operation of the Sellafield site ion-exchange effluent plant, SIXEP, as well as in 1994 because of disposals of decay-stored liquors. An increase in concentrations of Sr-90 in fish, however, was not observed until 1994, indicating a slower rate of uptake than for C-14.

Rates of discharge of Tc-99 (Figure 6) changed rapidly in 1994 due to processing of decay-stored liquors, and discharges are plotted on a monthly basis. Quarterly data for concentrations observed in lobsters (*Homarus gammarus*) near Sellafield are illustrated, again because of radiological significance, and for the seaweed *Fucus vesiculosus* because of its strong uptake of Tc-99 and indicator potential. This seaweed is not eaten but it can be used as a fertiliser. Near Sellafield, the effects of the period of release of Tc-99 in March/April 1994, whilst observed in both species, were not as significant as for the period of September/October 1994. This could have been due to hydrographic factors, because both releases were observed in a more additive fashion in *Fucus* at St Bees, 10km north of Sellafield.

RADIOLOGICAL SIGNIFICANCE

Changes in plant operations at Sellafield have resulted in decreases in discharges of some radionuclides (mainly the actinides, radiocaesium and Ru-106) and increases in others (C-14, Sr-90 and Tc-99). Assessments of doses to potential critical groups, however, have shown continuing reductions (2). The critical group near Sellafield of high-rate fish and shellfish consumers received 0.08 mSv in 1994 compared with 0.10 mSv in 1993 (and higher doses in previous years) using ICRP-60 data. This confirms the minor radiological significance of those nuclides whose discharges have increased. Nevertheless, careful surveillance is continuing to ensure that this remains the case under the developing operational conditions.

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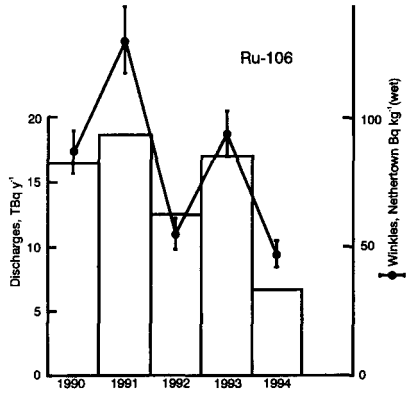


Figure 1.

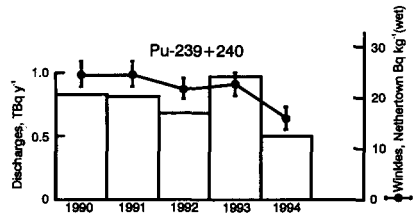
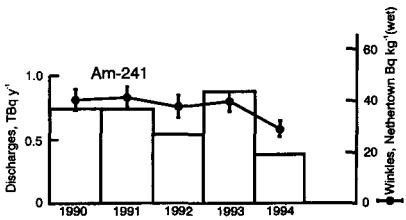


Figure 2.



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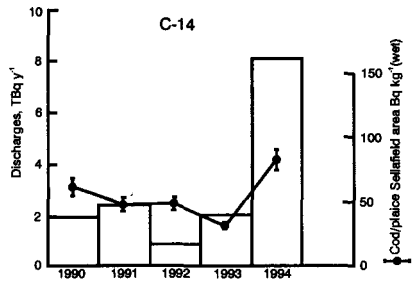

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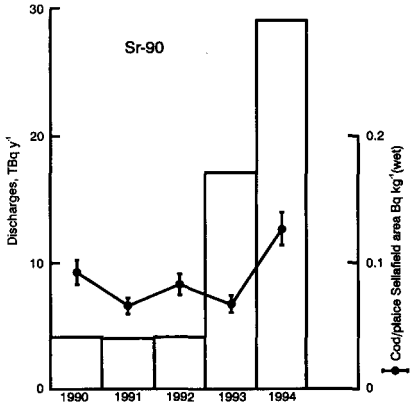

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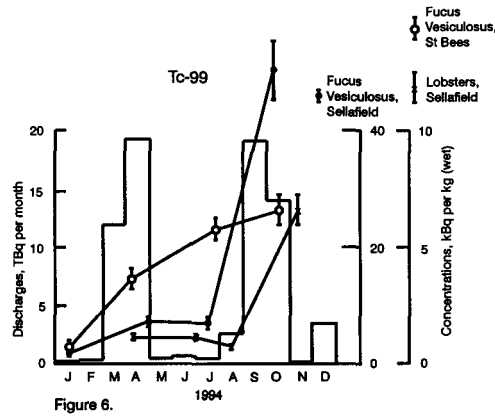

Figure 6.

^{137}Cs IN SEDIMENTS OF NORTHERN AND MIDDLE ADRIATIC SEA

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ABSTRACT

The activity of ^{137}Cs in relatively shallow northern and middle Adriatic open sea bottom sediments was obtained on the basis of gamma-spectrometry measurement results of 33 sediment box cores. Cores were sampled during the ASCOP 16 (Adriatic Scientific CO-operative Program) cruise in summer 1990 along seven different transects stretching between the borders of Croatian and Italian territorial waters. ^{137}Cs was determined in the recent sea bottom sediments (0-3 cm) as well as in older, 12-15 cm deep sediment cores. It was found that the lowest caesium concentrations correspond to sands which are spread along the Croatian coast mainly. Parallel to the Italian coast, ^{137}Cs concentrations are the highest in pellets, mud, silt and clay composed bottom. It seems that the influence of Po River is significant for ^{137}Cs activities in recent marine open sea bottom sediments along Italian coast south of Po River delta. Significantly higher ^{137}Cs activities in 0-3 cm vs. 12-15 cm open sea bottom sediment layer can be attributed to the deposition caused by Chernobyl accident. The absence of ^{137}Cs in few middle Adriatic 12-15 cm deep core indicates (inspite bioturbation that caused sediment mixing) significantly lower sedimentation rate in open sea of middle Adriatic vs. northern Adriatic and main deposition area of suspended matter carried by Po River.

INTRODUCTION

The shallowness of the continental shelf, which hardly overcomes 60 meters among Trieste, Venezia, Ancona, Zadar and Pula, and the great quantity of waters flowing from Po and other minor Italian and Croatian rivers, strongly determine the grain size distribution and sedimentation rate in this semi-closed system. The chemistry of northern and middle Adriatic as well as the cycling of nutrients and the fate of pollutants within the ecosystem are influenced by sedimentation characteristics. The countries surrounding northern Adriatic are reasonably contaminated by ^{137}Cs during Chernobyl accident (1-4).

The distribution of mean grain sizes, sediment types and main sediment characteristics in the open northern and central Adriatic sea are generally known (5). Spatial sediment distribution shows that the finest sediments are deposited along the Italian coast (south of the Po River delta) and in the central Adriatic, while sands are spread along the Croatian coast, as well as at stations 1, 4 and 32. The research of ^{137}Cs distribution in sediments of Grado and Marano lagoons (6) showed the influence of river input and grain-size distribution on caesium activities in recent marine sediments relatively near shore. The similar observation results are found in the broader Rhone mouth area (7). Concerning caesium solubility in sea water and significant input of terrestrial material (carried by rivers), the amounts and caesium distribution into recent marine sediments of open northern Adriatic are of special interest.

MATERIALS AND METHODS

The cores investigated in this work were sampled as part of a broader international and multidisciplinary research programme ASCOP, during the ASCOP 16 cruise in the summer 1990. Sediment samples were taken at 33 stations along seven different transects stretching between the borders of the Croatian and Italian territorial waters in the northern and central Adriatic Sea. The sediments were collected by box corer. After sampling, the sediments were sliced, frozen at -18 °C and kept until further use. Surface sediment (0-3 cm) and 12-15 cm-deep sediment samples collected along sampled transects in the northern and central Adriatic sea were prepared for gamma-spectrometric analyses.

Before the analyses, the samples were thawed at room temperature and dried at 106 °C to the constant weight. Dried samples were stored in 125 cm^3 counting vessels. The activity of ^{137}Cs in each sample was determined by gamma-spectrometry method using calibrated low background HP Ge detector system coupled to a 4096 channel analyzer. Spectra were recorded 80 000 seconds. ^{137}Cs activities were calculated from the 661.6 keV peak and the data were recalculated at May 1st, 1990.

RESULTS AND DISCUSSION

The distribution of the material, carried by the Po River and other Italian and Croatian rivers, on the sea bottom of open northern and middle Adriatic sea is the result of the sedimentation processes and the effect of the dominant sea currents. That fact is evident in ^{137}Cs activities in measured sediment cores. ^{137}Cs was detected in all surface sediment samples (0-3 cm) as well as in majority 12-15 cm-deep sediment samples.

The highest caesium activities are found in surface sediments taken at stations located in the Po River prodeltaic area (25.6 ± 0.8 Bq kg^{-1} dry weight at station 9 and 10.9 ± 0.3 Bq kg^{-1} dry weight at station 5). Along Italian coast south of Po River delta, the measured caesium activities in surface sediments are high too, about 10 Bq kg^{-1} dry weight generally. These two areas correspond with main sedimentation area of material carried by Po River. The bottom sediment composition in these two areas is made mostly by clayey sand, clayey silt, silt and clay (5) and has higher organic matter content (8). Relatively high caesium activities in surface sediments are found in the identical composed sediments of the central Adriatic at stations 28, 29, 31 and 33.

The lowest caesium activities measured in surface sediments (less than 2.0 Bq kg^{-1} dry weight generally) are found in samples collected at stations along Croatian coast as well as at stations located in middle part and in north-east part of Corsini Port-Pula and Riccione-Porer Island transects. In this area, the organic matter content in the bottom sediment, composed of sand mainly, is the lowest (5).

The similar spatial distribution of caesium activities was found in the 12-15 cm-deep sediment samples, but measured values were significantly lower generally. The exceptions are stations 9 and 10 with 10.2 ± 0.5 Bq kg^{-1} dry weight and 9.6 ± 0.4 Bq kg^{-1} dry weight respectively. Relatively very high caesium activities in 12-15 cm-deep cores at locations 9 and 10 indicates high sedimentation rate in this area. The lowest caesium activities in the 12-15 cm-deep cores are found in samples collected at stations located in middle part and in north-east part of sampled transects of northern Adriatic. In central part of middle Adriatic at stations 27, 28, 29, 32 and 33, caesium was not found in the 12-15 cm-deep sediment samples.

Inspite sediment mixing, which is caused by bioturbation, the absence of ^{137}Cs in few middle Adriatic 12-15 cm deep cores as well as very low caesium activities in same cores of middle and north-east part of northern Adriatic indicates significantly lower sedimentation rate in central part of middle Adriatic and middle and north-east part of northern Adriatic vs. in west and south-west Adriatic. Significantly higher ^{137}Cs activities in 0-3 cm vs. 12-15 cm-deep open sea bottom sediment layer can be attributed to the deposition caused by Chernobyl accident. Activities of ^{137}Cs in the first 15 cm of open northern and middle Adriatic sea bottom sediment (fig. 1) were estimated on the basis of measured activities in recent sea bottom sediment (0-3 cm) and older, 12-15 cm-deep sediment cores. Evidently northern and middle Adriatic marine sediments with high caesium content correspond with the area of the highest sedimentation rate. That area is the main deposition area of suspended matter carried by Po River.

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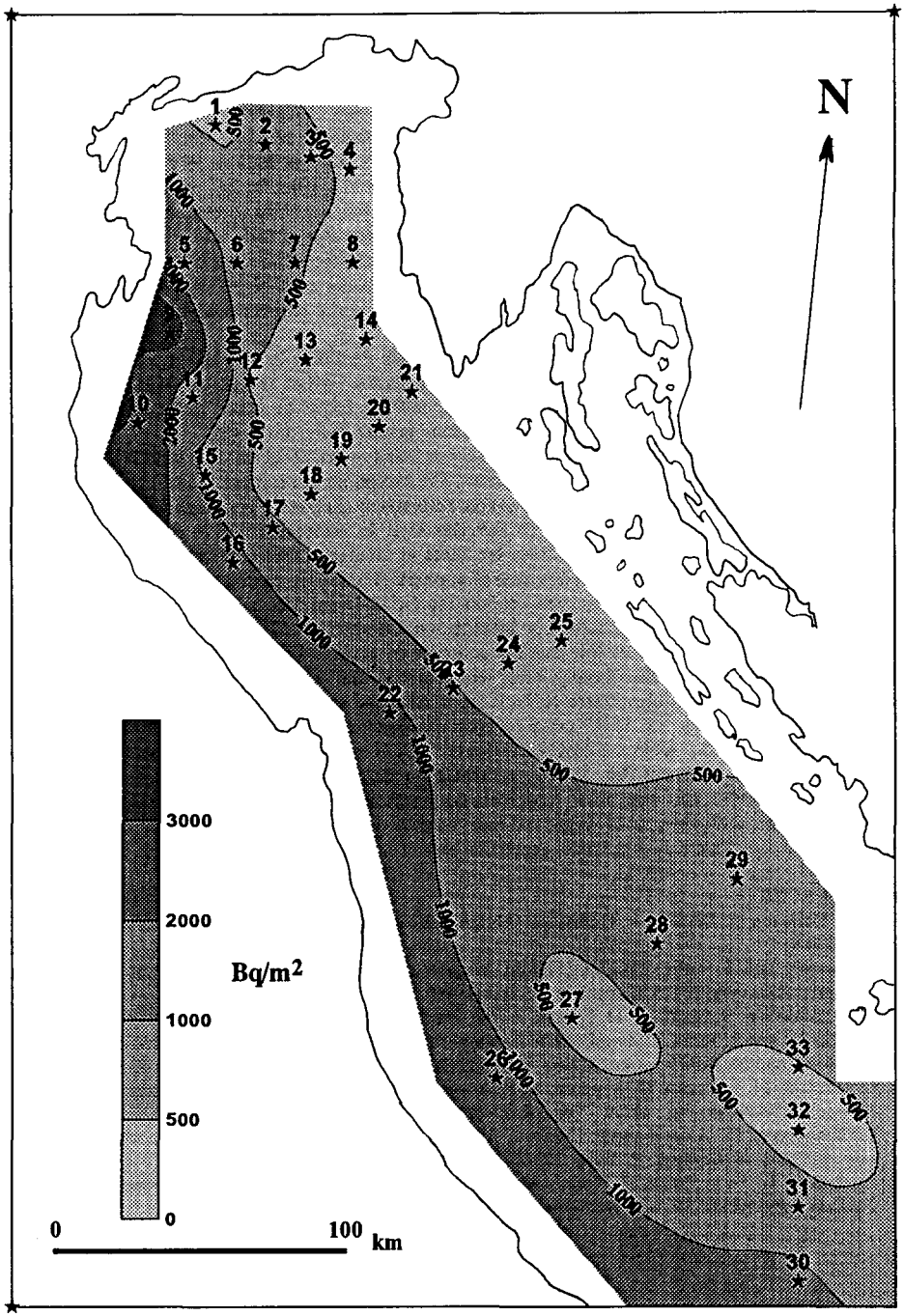

Fig. 1. Activities of ^{137}Cs in the first 15 cm of open northern and middle Adriatic sea bottom sediment

PLUTONIUM AND AMERICIUM IN THE RHÔNE SEDIMENTS

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INTRODUCTION

$^{238,239,240}Pu$ and ^{241}Am are measured in sediments from river ecosystems. These artificial radioisotopes are related to military and civilian activities. Small concentrations require an original radiochemical process, followed by alpha spectrometry. 104 analysis on sediment from the Rhône have been taken since 1989.

THE ORIGIN OF TRANSURANIC ELEMENTS IN THE RHÔNE

The transuranic elements present in sediment from the Rhône originate principally from the atmospheric nuclear tests carried out between 1945 and 1975, fallout from which spread some 1.2×10^{16} Bq of $^{239,240}Pu$ and 2.9×10^{13} Bq of ^{238}Pu into the atmosphere, 80% of which was in the Northern Hemisphere (1). Various accidents have also released transuranic elements into the environment. Such was the case with the SNAP-9A satellite, which burned up in April 1964 and released 5.1×10^{14} Bq of ^{238}Pu, 40% of that in the Northern Hemisphere (2). Fallout from the Chernobyl accident containing α emitters was negligible in France. Irradiated-fuel reprocessing plants (La Hague and Marcoule) are authorized to release α emitters in their liquid effluent. The Marcoule plant is authorized to release 150 GBq per year into the Rhône. In 1991, the Marcoule facility renovated its liquid effluent treatment station. The activities released diminished considerably, as is shown in the following table:

Years	$^{239+240}Pu$ (MBq)	$^{241}Am + {}^{238}Pu$ (MBq)
1990	2.11×10^4	2.07×10^4
1991	1.26×10^4	1.36×10^4
1992	0.33×10^4	0.45×10^4
1993	9.54×10^2	2.18×10^3

Nuclear power generation stations are not authorized to release α emitters in their liquid effluent.

PLUTONIUM AND AMERICIUM DETERMINATION PROTOCOLS

The sediment sample is autoclaved at 110°C, then sifted at 2 mm. A 200-g sample is calcined at 550°C in order to destroy any organic material present. After cooling, the sample receives 20 mBq of ^{242}Pu and ^{243}Am respectively. These isotopes, not present in the sample, make it possible to determine the yield of the various radiochemical operations after the final counting by α spectrometry.

The sediment then undergoes a series of oxidizing acid attacks in order to render the transuranic elements soluble. The solution thus obtained is made up essentially of the stable major elements (Ca, K, Na, Fe) comprising the matrix and the radioisotopes of the artificial elements (Pu, Am, Cm) as ultra-traces. The problem of the analysis is to separate and purify the plutonium and americium using a protocol allowing progressive elimination of the totality of the stable major elements and all of the natural α-emitting radioisotopes. The proportioning protocol uses a radiochemical process whose successive stages make use of coprecipitation, ion-exchange chromatography, and extraction chromatography techniques. Through a first series of coprecipitation of plutonium and americium with calcium oxalate at pH=1.5, the alkalines are eliminated while the iron is maintained in solution through formation of complexes with the oxalate ion. The isolated and dried calcium oxalate precipitate is transformed into calcium carbonate through calcination. The latter is dissolved in a hydrochloric acid solution, diluted with distilled water, and brought to pH=8.00 in order to obtain precipitation of the last traces of iron in the form of ferric hydroxide. Quantitative coprecipitation of the plutonium with a few milligrams of iron makes it possible to eliminate almost all of the alkaline earths. The small volume of solution obtained after dissolving the precipitate in an 8M nitric medium is injected onto an AGIX8 anionic resin, which allows separation of the plutonium from the elements Th, U, Am, and Cm and the rare earths and other metals not eliminated during the coprecipitations. Separation of the americium from the elements Fe, Cu, Ni, and the rare earths requires two further extractions. The first consists of an extraction chromatography done on TRU-spec resin, and the second uses ion-exchange chromatography on AGIX4 resin.

Sequential separation of the elements Pu and Am is done through two electrodepositions, each carried out on a previously electropolished stainless-steel disk. Each electrolytic deposit is then counted in a chamber maintained under vacuum and equipped with an inserted passivated junction silicium detector having a useful surface equivalent to 280 mm^2. Duration of the count can be up to one week. The detection limit made possible both by equipment performance (a detection yield of between 0.25 and 0.30 - an electronic background noise level of < 2 cps/24 hours) and by that of the radiochemistry used (weight of the sample being processed, tracer purity, separation performance, peak resolution) varies between 1 and 10 mBq/kg for each radioisotope proportioned.

Pu AND Am CONCENTRATIONS IN SEDIMENTS

Distribution of transuranic elements in sediment is not uniform throughout the length of the Rhône. The river may be separated into three zones according to the potential source terms:

Sediments sampled upstream from Creys-Malville.

This part of the Rhône is not subject to the influence of any nuclear installation. The concentrations observed are, respectively: 0.017 to 0.086 Bq.kg $^{-1}$ dry for ^{239,240}Pu and < 0.001 to 0.0044 Bq.kg $^{-1}$ dry for ^{238}Pu. ^{241}Am was not measured.

The ^{238}Pu/ ^{239,240}Pu isotopic ratio of 0.043 ± 0.038 (3 significant values) is characteristic of fallout from atmospheric tests (values between 0.02 and 0.08) (3).

Sediments sampled from Creys-Malville to a point upstream from the Marcoule fuel reprocessing plant.

The sediments show concentrations of 0.030 to 0.201 Bq.kg $^{-1}$ dry for ^{239,240}Pu, from <0.001 to 0.017 Bq.kg $^{-1}$ dry for ^{238}Pu, and from <0.005 to 0.076 Bq.kg $^{-1}$ dry for ^{241}Am.

The ^{238}Pu/^{239,240} Pu isotopic ratio is 0.046 ± 0.010 (15 significant values). As in the case of the previous zone, it shows that the plutonium measured is from atmospheric testing fallout. Higher plutonium concentrations were measured downstream from the Creys-Malville plant than upstream in Rhône sediments collected in 1991. But all the isotopic ratios are characteristic of atmospheric fallout. The differences observed are due only to granulometric differences (4). In fact, plutonium radioisotopes have a great affinity for finer particles.

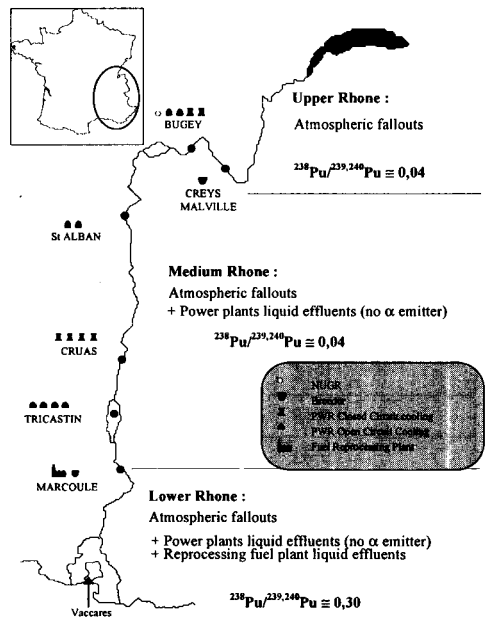

Figure 1. Source terms and Pu ratio in the Rhone river sediments

If the plutonium isotopic ratio departs from this value, this can be a sign that the plutonium is from another source. Thus in the Loire, Martin and Thomas (3) found a ratio of 0.15 to 0.40 downstream from the C.N.P.E. facility at St-Laurent-des-Eaux following an incident in 1982 that led to the melting of two fuel elements in reactor 2. The ratio was less than 0.04 in 1994 (5). Similarly, downstream from the C.N.P.E facility at Bugey, sediment collected in July 1989 had an isotopic ratio of 1.66 (6). This could be explained by an incident at the site's graphite-gas reactor. A second sediment sample collected in 1992 showed a ratio of 0.05, which is consistent with the characteristic value for atmospheric test fallout.

Sediments sampled downstream from the Marcoule fuel reprocessing plant.

In the sediments from this area, concentrations vary between 0.2 and 7.8 Bq.kg^{-1} dry for ^{239,240}Pu, between 0.17 and 2.1 Bq.kg^{-1} dry for ^{238}Pu, and between 0.4 and 5.3 Bq.kg^{-1} dry for ^{241}Am. The plutonium isotopic ratio is 0.280 ± 0.023 (56 significant values), thus greater than that found farther upstream (7). It is characteristic of the plant's liquid effluent (8). The startup of the liquid effluent treatment station in 1991 caused a decrease in the concentrations observed in sediments collected between Marcoule and the Camargue:

Dates	^{239,240}Pu (Bq.kg^{-1} dry)	^{238}Pu (Bq.kg^{-1} dry)	^{241}Am (Bq.kg^{-1} dry)
September 1990	0.77 to 5.49	0.54 to 1.82	< 0.48 to 4.46
September 1991	0.27 to 4.39	< 0.1 to 1.42	< 0.26 to 4.12
September 1992	0.26 to 2.58	0.06 to 0.72	0.13 to 4.14
1993 *	0.07 to 2.59	0.07 to 0.60	0.04 to 2.18

A sediment sample was taken in Lake Vaccarès (in the Camargue) in September 1992. The measurements resulted in the following values: 0.058 ± 0.006 Bq.kg^{-1} dry for ^{241}Am, 0.107 ± 0.010 Bq.kg^{-1} dry for ^{239,240}Pu, and 0.0038 ± 0.0010 Bq.kg^{-1} dry for ^{238}Pu (9). The ^{238}Pu/ ^{239,240}Pu isotopic ratio of 0.035 indicates that the point on the lake sampled is not influenced by the liquid effluent from the Marcoule plant.

CONCLUSION

Transuranic elements measured in sediments from the Rhône are in weak concentrations. In the entire part of the river not affected by effluent from Marcoule, the ^{238}Pu/^{239,240}Pu isotopic ratio is in the area of 0.04, which is characteristic of fallout from atmospheric testing.

Downstream of the Marcoule plant, the ratio is 0.3, which is characteristic of the liquid effluent from the irradiated fuel reprocessing plant. Since 1991, concentrations of transuranics in the sediments have decreased as a result of the startup of the liquid effluent treatment station.

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* The level of the Rhône rose greatly in October 1993 and January 1994, flooding a small part of the Camargue region. In the flooded areas, plutonium radioisotopes from sediments from the Rhône were found in the soil of the rice fields. Their isotopic ratio was 0.2, characteristic of Marcoule. Agricultural work reduced the plutonium concentrations by a factor of 3 (0.03 Bq.kg^{-1} for ^{238}Pu and 0.19 Bq.kg^{-1} for ^{239,240}Pu). No significant activity could be measured in the rice.

INDUCTIVELY COUPLED PLASMA SPECTROMETRY AND NEUTRON ACTIVATION ANALYSIS OF RARE EARTHS IN CACHOEIRA RIVER

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ABSTRACT

Inductively Coupled Plasma Spectrometry (ICP-MS) and Neutron Activation (INAA) are used for determination of Rare Earths in the water and Sediments of Cachoeira River in State of Bahia - Brazil. Elemental concentration of 10 elements have been measured in 20 ground water and sediments samples collected to different parts of this river. There is a good agreement between the two analytical methods and the results are complementary.

INTRODUCTION

The concentration and physico-chemical State of trace elements in water and sediments is of considerable importance for their transport in nature and then influence on biological systems.
Several methods have been used for measuring the environnent pollution in water, sediments and aquatic plants. Neutron activaction analysis is a classical method for multi elemental determination which has a high sensitivity at trace level[1,2]. The concentration of rare earths is very reliable in environmental control[2]. This technique has been widely used for the analysis of water and sediment[3,4].
Inductive Coupled Plasma Mass Spectrometry may be used for trace analysis of environmental samples. This multi-elemental technique is accurate sensitive and fast[5], allowing a daily processing rate of up 100 samples or more and has recenthy been used to determine the composition of rivers water and for routine trace analysis [6,7] ICP-MS became a prefered method of elemental analysis with detection limits for most of elements of less than 0,1 µg/l [8].This work uses INAA and ICP-MS analysis methods to determine level concentration of rare earths.

EXPERIMENTAL

Water and sediment samples were collected of the approx. 30 km long the Cachoeira river. Water samples were collected near the river bank from a depth 15cm, placed in 125 ml Nalgene bottles and stored at 4°C tempertaure pH and electrical conductivity were measured by each sites.
We had not detected significative variations between the different areas. The average values of these parametres are: temperature 30°C ; pH = 7,2 ; eletrical conductivy = 630 µS.
The sample were taken in duplicate and half of them were acidified on site with 5 drops ultrapur acid nitric (pH =2). This acidification is believed to reduce any absorption on the walls of the bottles[9]. The sediments samples were collected in 20ml polyethylene tube and also stored at temperature of 4°C. They were dried for one day at temperature of 90°C.
Prior to analysis one sediments sample was collected per water sample. The method of neutron activation analysis used in the present experiment consists in the pre-concentrated of water samples before irradiation, 10ml sample were evaporated in the ultrapur quartz ampoules (SUPRASIL) 80°C for during 24h and heat - sealed. The samples and standards were irradiated in Saclay's OSIRIS nuclear reactor at a neutron flux of 1,4 X 10^{14} $n.cm^{-2}.s^{-1}$ for 17h. The Gama rays subsequently limitted were counted for 8 hours, after a decay time of 10 a 20 days, with 100 cm^3 coaxial HPGe PGT detector coupled to a 4096 pulse height analyser. The computational Ko method [9] was used to determine the concentration of elements of interrest.
The sediment samples were irradiated in the OSIRIS for 2h. To enable the techniques of INAA and ICP-MS to be compared, 20ml of 3:1 mixture of HCl and HNO_3 was added to the irradiated sediments samples, the mixture evaporated at 80°C diluted with 20ml of 5% nitric acid and centrifuged. The liquid samples were counted for 4 hours.

The ICP-MS mesurements were corried out using the multi-elemental modes. An analytical program was stablished for both calibration and routine analysis. This program uses a series of automated operations to align the optics, select the analytical wavelenght for the peacks and position the source on the entrenance slit in order to optimize the sinal. The selected analytical wavelenghts are the characteristics lines of the elements which are free from spectral interferance and this avoids the necessites of the corrections at concentration level of interest. The ICP-MS used in this study was a PQ 2 - PQ Vision software. The mesurements were taken in duplicate, using the following operating conditions: - ICP-MS r.f. power: 1,35 Kv; - Coolant Argon flow: 14 l/min; - Nebulizer Argon flow: 0,8 l/min; - Auxiliary Argon flow: 0,8 l/min; - Sample uptake rate: 0,6 - 1 ml/min. The instrument was calibrated with a commercial solutions (SPEX) which contained standards prepared out of 10 ppb multielementary and 1000 ppb Ca solutions. Blanks were also used in order to enable an accurance of 2-3%. The internal standards of samples were 10 ppb solution of In, Re, Be. The relative standard deviation was typically 5 -10%.

RESULTATS AND DISCUSSION

The samples analysed were collected from four diferents areas to Cachoeira river. The resultats (Table 1 and 2) indicate that the rare earths concentration in water and sediments sample is present the maximum concentration in the area « P. do Pitu ». The separation of suspend matter by filtration process permit to distinguish particulate material in suspension from the elements in solution. This process presents the risk of contamination. The centrifugation process reveals a good behavior the effectiveness of the centrifugation process is a function of the centrifuge speed, time and particle density. A centrifugation at 4300 rpm during 10 minutes is convenient.

Table 1 - Concentration of Rare Earths in water samples (ppb)

Areas	Itapé		Ferradas		Itabuna		B.Vitória		Ilhéus	
Elements	ICP-MS	INAA	ICP-MS	INAA	ICP-MS	INAA	ICP-MS	INAA	ICP-MS	INAA
La	0.37	0.33	0.59	0.56	2.1	2.0	1.4	1.2	0.2	0.18
Ce	0.71	0.68	1.2	1.1	4.0	3.8	2.8	2.7	0.48	0.46
Nd	0.35		0.58		1.80		1.3		0.21	
Pr	0.08		0.11		0.46		0.34		0.04	
Sm	0.07	0.04	0.12	0.13	0.3	0.27	0.24	0.26	0.06	
Eu	0.13	0.11	0.05		0.23		0.22		0.01	
Gd	0.04		0.08		0.3		0.22		0.04	
Dy	0.06		0.12		0.28		0.20		0.02	
Er	0.02		0.03		0.14		0.10		0.01	
Yb	0.03		0.02		0.11	0.10	0.08		0.01	

Table 2 - Concentration of Rare Earths in sediments samples(ppm)

Areas	Itape		Ferradas		P. do Pitu		Ilhéus	
Elements	ICP-MS	INAA	ICP-MS	INAA	ICP-MS	INAA	ICP-MS	INAA
La	8.0	7.2	5.3	5.0	18	19	11	10.5
Ce	17	16.5	11.5	10.8	35	34.5	21	20
Nd	6.9		4.8		13		9.6	
Pr	1.8		1.4		3.8		2.8	
Sm	1.3	1.3	0.78	0.73	1.5	1.4	1.6	1.5
Gd	1.5		1.5		3.1		2.0	
Eu	0.26	0.22	0.14	0.12	0.35	0.30	0.12	0.10
Tb	0.13		0.17		0.28		0.19	
Dy	0.9		0.4		1.3		1.1	
Ho	0.04		0.01		0.16		0.07	
Er	0.31		0.28		0.54		0.28	
Yb	0.19	0.17	0.18	0.16	0.26	0.28	0.23	0.22

CONCLUSION

The asssociation of INAA and ICP-MS, two metods of multi-elementary analysis permits to investigate the rares earth elements in water and sediments samples.
The two analytical methods are complementary and in general the results showed a good agreement.

ACKNOWLEDGEMENTS

This paper was supported by CAPES (Ministério da Educação do Governo do Brasil) CNPq (Ministério da Ciência e Tecnologia do Governo do Brasil) and CNRS (Centre National de la Recherche Scientifique, France).

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MEASURED TRITIUM IN GROUNDWATER RELATED TO ATMOSPHERIC RELEASES FROM THE MARCOULE NUCLEAR SITE

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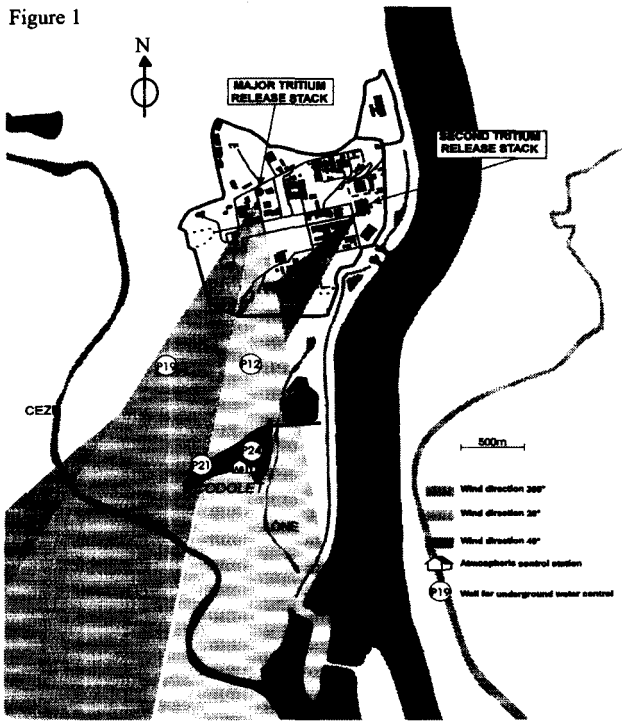
1 - Introduction

Tritium is released into the atmosphere during normal operation from the industrial facilities operated by COGEMA at Marcoule; over a 15-year period covered by this study (1979-1994) the quantities ranged from 4940 to 520 TBq·yr^{-1}. Atmospheric release in rainy weather results in tritium migration into the groundwater by a series of mechanisms associated with the water cycle. COGEMA monitors the groundwater by means of boreholes, four of which are shown here (Figure 1). Atmospheric monitoring is also routinely performed; data on the tritium activity concentration in the air and rainwater are available for the same time period. A simplified observation suggests a relation between the atmospheric tritium release and the groundwater radioactivity .In 1994, the activity ranged from 100 to 200 Bq·l^{-1} in the boreholes located 1 km and 2 km downwind from the point of release, diminishing with the distance to less than 20 Bq·l^{-1} at about 3 km.

Figure 1



The authors attempted to model two types of transfers: atmospheric transfer from the release chimney to the borehole, and transfer in the alluvial groundwater. The aquifer comprises the alluvial deposits forming the Codolet plain extending to the south of Marcoule, downwind from the point of atmospheric tritium release. The hydrogeology of the entire Marcoule site has been described in previous studies by the French bureau of geological and mineralogical research (BRGM) and ANTEA.

2 - Modeling Atmospheric Transfer

COGEMA possesses a site model for Marcoule based on a wind-tunnel mockup representing the local relief, and used to calculate the Atmospheric Transfer Coefficient (ATC). Given the known annual tritium releases, weighted by the frequency of tritiated vapor (HTO) conditions, the ATC values are used to calculate the activity concentration in the air in rainy weather, and to determine the concentration in the rainwater:

$$Ar = Aa.\frac{1}{\rho}.f$$

where Ar : radioactivity in rainwater (Bq·l^{-1})

Aa : radioactivity concentration in air (Bq·m^{-3})

ρ : specific activity of water vapor in rainy weather (12.10^{-3} kg·m^{-3})

f : factor characterizing tritiated water vapor ⇄ rainwater exchange[1] depending notably on distance and intensity of rainfall.

The authors determined an experimental value for f using the weekly measured values in the rainwater and air at the atmospheric monitoring station (AS1): $f = 0.15$. The measured rainwater activity at AS1 correlates satisfactorily with the value calculated by the model (Figure 2). The model may therefore be used to calculate the rainwater activity contributing to the groundwater activity at any point on the plain.

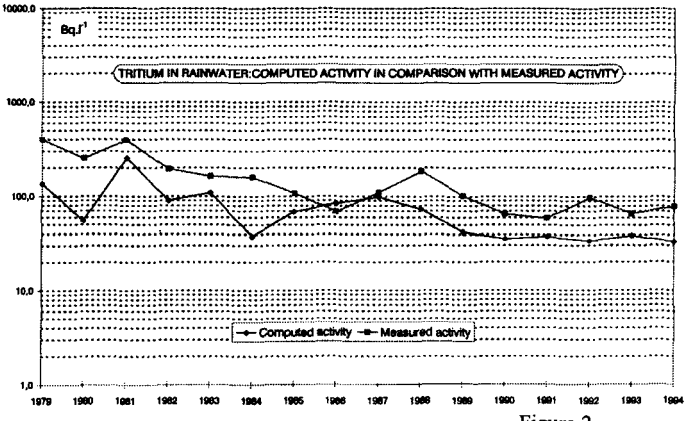


Figure 2

3 - Modeling Rainwater – Groundwater Transfers

When tritium-laden rainwater reaches the ground, the complexity of the subsequent pathways depends on the nature of the soil and its moisture content:

- A negligible fraction enters the surface hydrographic system directly in lakes or waterways.

- Runoff water contributes to the surface water flow.

- Another fraction infiltrates directly into the soil and may enter the evapotranspiration cycle, constituting an available water supply in unsaturated zones, or may slowly percolate down to replenish the groundwater. The rainfall replenishment fraction is subject to specific hydrodynamic mechanisms in the aquifer, and migrates at a rate depending on the hydraulic gradient, the permeability and the storage coefficient in the case of transient variations.

Tritiated water migration in the aquifer is the result of three kinematic phenomena:

- convection, corresponding to the mean fluid displacement;

- kinematic dispersion, due to the heterogeneity of the microscopic flow rates;

- molecular diffusion through Brownian movement of water molecules (this phenomenon is negligible in a permeable aquifer).

3.1 -Calculating the Rainfall Replenishment

Climatological data on rainfall, temperature and insolation obtained from the French Weather Service for the period from 1979 to 1994 were used to determine the evapotranspiration potential (ETP) from the Turc and Penman formulas. Groundwater replenishment by rainfall was estimated using the GARDENIA hydrological model developed by the BRGM, in which a series of reservoirs simulate the principal mechanisms of the water cycle: rainfall, evapotranspiration, infiltration and flow. Transfers from one reservoir to another are governed by simple laws and are adjustable for each reservoir. The model calculates the theoretical water table from rainfall records and from the ETP. The model flow parameters are adjusted by comparison with measured groundwater levels.

The mean annual replenishment calculated in this way over the time period of this study was 145 mm for a mean annual rainfall of 740 mm. The ratio of the two terms, approximately 20%, is consistent with the values commonly observed for this type of aquifer.

3.2 - Hydrodynamic Flow Model

A hydrodynamic model is used to quantify the groundwater flows based on the calculated rainfall replenishment, the aquifer geometry and hydrogeological parameters (permeability, storage, etc.) and boundary conditions. The MARTHE code (BRGM) solves the diffusivity equation using a finite-differences algorithm. In the hydrogeological context of the site, a single-layer regular square mesh was used.

The model was validated for the Codolet plain by comparison with reference data representative of intermediate water conditions from a synchronous piezometric survey conducted in October 1995; under these conditions, the groundwater drains into the Lône and the Cèze.

3.3 - Transport Model

The objective is to account for the tritium activity values measured by piezometers in the Codolet plain based on the various groundwater migration mechanisms using the MARTHE code, which also integrates specific transport equations.

Tritium influx to the groundwater was estimated over monthly time steps from the calculated rainfall replenishment and from the radioactivity measured in the rain by COGEMA. These values were used as input for the model in the form of a mass infiltrated over the entire surface of the Codolet plain.

The model uses the hydrodispersive parameters (kinemattic porosity, longitudinal and transverse dispersion) estimated for the aquifer after several groundwater tracing campaigns carried out earlier on the Marcoule site. The simulations covered a 15-year period from 1979 to 1994. The model calculates the radioactivity of each grid square for each monthly time step to account for the measured variations and the iso-activity maps.

4 - Conclusion

The results show satisfactory agreement between the radioactivity values measured in the boreholes and the calculated values (Figure3). It appears that the tritium activity observed on the Codolet plain may therefore be explained by the contribution of rainfall, based on a deterministic approach taking into account all the physical phenomena involved in the transfer.

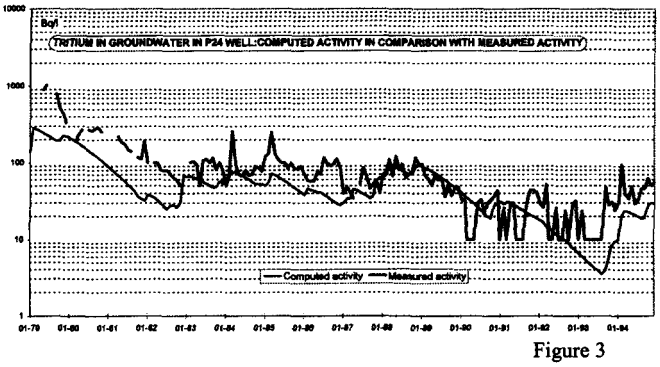


Figure 3

The combined use of several models makes it possible to predict the groundwater radioactivity according to the released activity and the average weather conditions.

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RADIOISOTOPE AND STABLE ELEMENT PARTITIONING COEFFICIENTS IN THE DANUBE RIVER

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ABSTRACT

The analytical measurement results obtained during the last decade for the radionuclide, trace element and heavy metal concentrations in sediment, periphyton and water samples collected from the Danube River were explained with respect to the partitioning coefficients. The periphyton concentration ratios (CR·s) for each isotope were correlated with the sediment distribution coefficients (K_d·s) for that isotope. The K_d-CR relationships obtained were defined by the equation $\lg K_d = -0.70 \lg CR + 3.16$. The K_d-CR relationship determined in this work was compared with the K_d-CR relationships proposed by Baes and Shepard.

INTRODUCTION

The concern for the appropriate partitioning coefficient (distribution coefficient and/or concentration ratio) determination is important for the modelling of aquatic systems and the engineered and/or the natural radwaste barrier systems (1,2,3,4,5,6,7). One of difficulties associated with measurement of partitioning coefficients either in laboratory or in environment is to ensure that the element concentration in sediments/biota is in equilibrium with the element concentration in water (2). The great number of factors-physicochemical form of element, presence of competing ion, colloid formation, co-precipitation, etc. - which affect element assimilation and/or retention account in part for the variability of partitioning coefficient values reported in the literature (3,4,5,6,7). It appears, also, that the applicability of laboratory partitioning coefficients and/or in-situ partitioning coefficients depends on the specific problem studied.

MATERIALS AND METHODS

Sample collection and preparation.

Three locations along the Danube River, from Visnica, 1162 km upstream, to Ostrovo, 849 km upstream, were chosen for samples collection and for the routine control of the river pollution for the last 10 years. Sediments were collected by a grab sampler and periphyton samples were collected manually. Water samples were collected with plastic bottles and acidified with 0.1N HNO_3. Sediments and periphyton were dried in an oven at 105°C to the constant weight. Water samples were filtered and evaporated under infrared lamps.

Measurement procedure.

Energy dispersive x-ray fluorescence spectrometry (XRF) was applied for stable elements concentration measurement in the all collected samples. Excitation with gamma rays

from ^{109}Cd of 370 mBq and ^{241}Am of 3700 mBq was employed. The detector Si(Li) had a resolution of 250 eV FWHM at 5900 eV. From the known intensity ratios of k_α/k_β, the overlapping lines of the neighbouring elements were subtracted. The detector efficiency correction curve was determined by the measurement of standard sample matrix (8).

The radioisotopes in periphyton were analysed on a HP Germanium (ORTEC) well-type detector and those in sediments and water on a HP Germanium (ORTEC) coaxial detector. The same geometry was used for each periphyton and sediment and water sample. The counting times were 85,000 s. The accuracy of the measurement was checked by counting NBS Standard Reference Material-River sediment No 4350 (9).

RESULTS AND DISCUSSION

The partitioning coefficients such as the periphyton concentration ratios (CR·s) and the sediment distribution coefficients (K_d·s) were assumed as indicative of elements bioaccumulation, transport and retardation in the Danube River ecosystem. The determination of the partitioning coefficients was based on the mass balances. Thus the distribution coefficient (K_d) is the ratio of the element concentration in the sediment to the concentration of the same element in the water, $K_d = C_s/C_w$. The K_d·s were expressed in ml g^{-1}. The CR is the ratio of the element concentration in the periphyton to the sum of concentrations of the same element in sediments and water, $CR = C_p/(C_s+C_w)$. The CR·s were expressed in mg g^{-1}. The K_d-CR relationships calculated for the Danube River are shown graphically in Figure 1. Each point in the figure is accompanied with the average standard deviation of all data ($\sigma_{lg\ CR}$ and $\sigma_{lg\ Kd}$). The K_d-CR relationship obtained for the Danube River is defined by the following equation $lg\ K_d = -0.70\ lg\ CR + 3.16$. The significant correlation was -0.75 at $P < 0.05$. The 95% confidence curves and the regression line are shown in the same figure. The K_d-CR relationships proposed by Baes (4) and Shepard (5) were expressed by equations $lg\ K_d = -0.85\ lg\ CR + 1.31$ and $lg\ K_d = -1.12\ lg\ CR + 2.08$ and presented in the Figure 1. Our K_d-CR relationship was used to calculate the missing K_d·s or CR·s for a wide range of radionuclides and stable elements (10) and to compare these values with those found by Baes, Shepard and the other workers (6,7,8). From the other author·s findings and from this work it is evident that it is not possible to establish a single K_d/CR value for each isotope.

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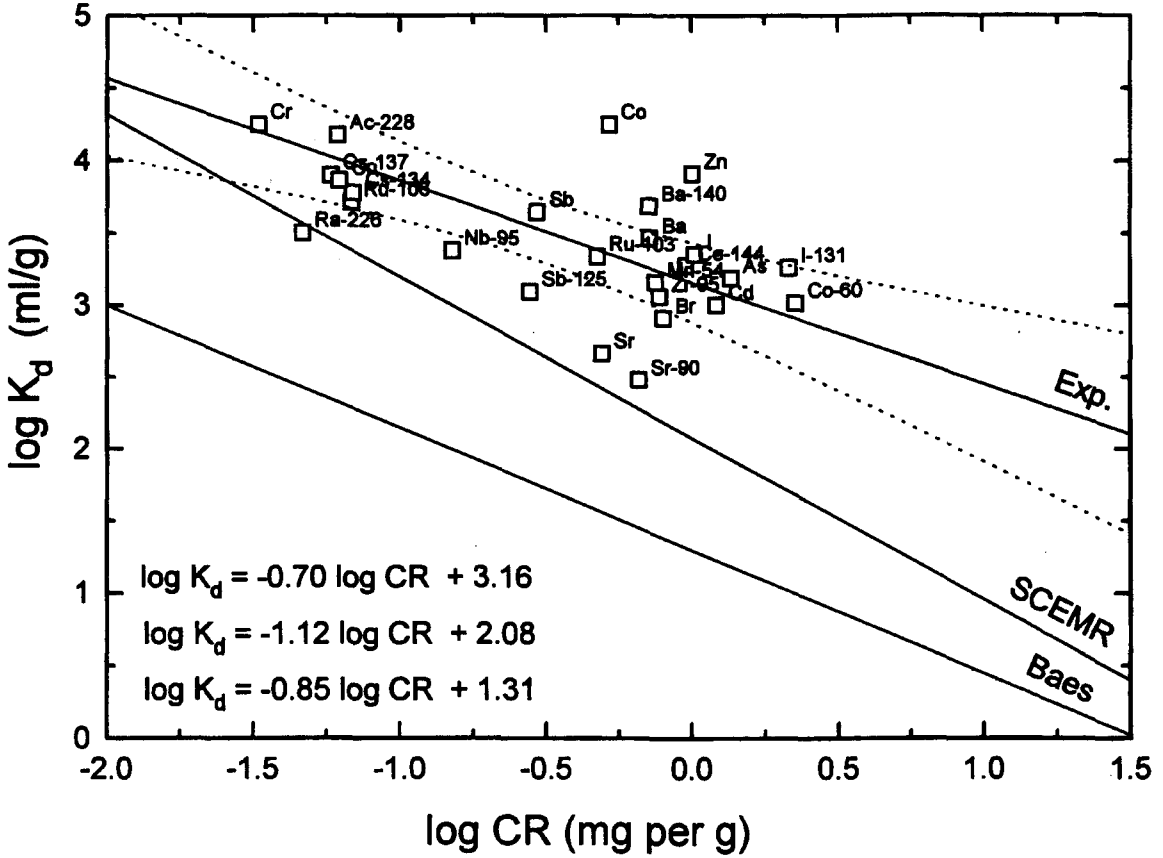

Figure 1. DISTRIBUTION COEFFICIENT VS CONCENTRATION RATIO

IRPA9
1966 International Congress on Radiation Protection
April 14-19, 1966
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PAPER TITLE **^{238}U, ^{232}Th, ^{226}Ra and ^{40}K Radioactivity Levels in Different Types of Fertilizers**

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MAJOR SCIENTIFIC TOPIC NUMBER (see page 7) Environmental Radioactivity
ABSTRACT (See instructions overleaf)

^{238}U, ^{232}Th, ^{226}Ra and ^{40}K Radioactivity Levels in Different Types of Fertilizers

Samir Abdul-Majid, Ibrahim Kutbi, Waleed Abulfaraj and Majid BaSabrian

Saudi Arabia import large quantity of inorganic fertilizers each year, from Europe, Midle East and U.S.A. Analysis were made for natural radionuclides in 42 samples of different types, by gamma spectrometry to assess the extend their application to soil would affect the soil radioactivity levels. Out of the 10 phosphate fertilizers samples analyzed, 5 showed very high ^{226}Ra concentrations whose mean value was about 620 Bq/kg, more than one order of magnitude that of soil while the rest 5 showed very low mean values of about 24 Bq/kg. The ^{238}U concentration were high in all the 10 samples and had a mean value of about 2600 Bq/kg. The ratio of $^{238}U/^{226}Ra$ revealed that the two radionuclides were far from being in equilibrium. The ^{40}K was very low of about 90 Bq/kg, about one order of magnitude less than ^{40}K in soil, that of ^{232}Th was about 30 Bq/kg.

The analysis of 32 NPK and potassium types showed very low ^{226}Ra and ^{232}Th concentrations of only few Bq/kg, while the mean ^{40}K was about 1300 Bq/kg. That of trace elements fertilizers showed both low ^{226}Ra and ^{232}Th concentration of few Bq/kg and low ^{40}K level of about 65 Bq/kg mean values.

URANIUM MINING AND MILLING SITES IN ARGENTINA: ENVIRONMENTAL RADIOLOGICAL MONITORING (1981-1994)

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ABSTRACT

Environmental radiological monitoring in the vicinity of Argentinean uranium mining and milling plants is performed on a routine basis, in order to assess the possibility of a significant environmental contamination due to uranium mill wastes or by mill tailings for the plants still operating or by those plants where the exploitation was concluded.

Dissolved ^{226}Ra and natural uranium concentrations in surface water are measured in samples taken at selected points upstream and downstream from rivers, in the area of influence of the mills.

In the present paper the environmental radiological monitoring program results obtained for the 1981-1994 period are shown, and from the analysis of these data it can be concluded that no remarkable exposure occurs for the population living in the vicinity of the analysed areas due to the uranium mining and milling plants operation or their wastes.

INTRODUCTION

In figure 1 are shown the uranium mining and milling areas included in the environmental radiological monitoring program, their geographical location and the type of exploitation. The uranium industry began its development in Argentina in the '50 and, at present time only San Rafael and Los Colorados plants are still operating.

As a results of plant operation, two types of wastes are obtained, acid liquid wastes that are neutralised before being piped into large evaporation ponds and ore tailings, solid wastes that result after the acid-leach process.

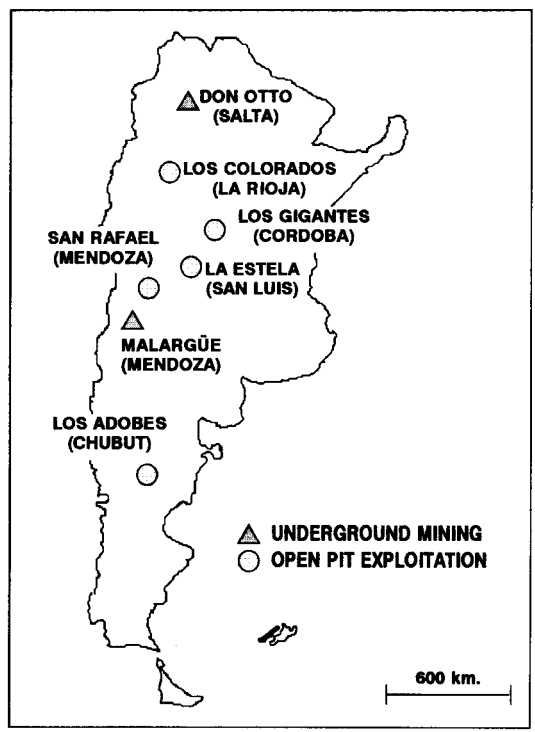

Figure 1: Uranium mining and milling areas included in the environmental radiological monitoring program.

MATERIALS AND METHODS

Surface water samples are taken from river locations upstream and downstream from uranium mills, according to a special monitoring plan set up for each facility (1), in order to allow the estimation of an eventual water contamination.

Dissolved ^{226}Ra in surface water samples were analysed by the classical radon emanation Rushing technique with Lucas cells (2). From 1988 till 1993, gross alpha counting using ZnS (Ag) as a detector was performed on water samples. Only those samples exceeding 70 mBq/L of gross alpha activity were selected for ^{226}Ra analysis. Nowadays, the ^{226}Ra concentration is measured by the ^{222}Rn emanation technique in sealed vials with toluene scintillation cocktail (3). The method consists in a ^{226}Ra co-precipitation with $BaSO_4$ and an alkaline EDTA dissolution before the liquid scintillation counting .

Natural uranium concentration is determined by a direct fluorometric measurement or after a co-precipitation with $CaHPO_4$ as a carrier, in order to enhance the detection limit.

^{222}Rn emanation rate from ore tailings is determined by adsorption on activated charcoal and a gamma measurement of the ^{214}Bi (609 Kev) in equilibrium.

RESULTS AND CONCLUSIONS

Figures 2 and 3 show the range of values of ^{226}Ra activities and the maximum natural uranium concentrations measured in surface water samples taken from the Diamante river and the El Tigre stream, from sampling points located in the surrounding of the San Rafael plant, from 1981 to 1994.

The average ^{226}Ra concentration was 4mBq/L either for samples taken from points < 10 km upstream or those samples taken < 50 km downstream, from the San Rafael mining and milling plant, with a range of values of 0.7-19 mBq/L and 0.7-13 mBq/L, respectively.

For the same period of time considered, the average natural uranium concentration measured was 0.02 mg/L, with a range of values of 0.0009-0.1 mg/L either in samples taken above or below the plant discharge point.

The application of the statistical test of Wilcoxon (4) demonstrated that no significant differences in the ^{226}Ra and natural uranium concentrations were found between the surface water samples from river locations above and below the milling plant of San Rafael, as well as for the rest of mining and milling areas of Argentina, where the environmental radiological monitoring program was performed.

Figure 4 and 5 show the corresponding ranges of ^{226}Ra concentrations and the maximum natural uranium concentrations, obtained for all the uranium sites, for the last year assessed. An important variability was observed in the concentrations of the radionuclides of interest, due to geographical and seasonal causes, but all the results obtained were below the derived limits for drinking water, that taking into account the biokinetic model (5) and dose conversion factors (6) were estimated as 180 mBq/L for ^{226}Ra and 0.12 mg/L for natural uranium.

^{222}Rn emanation rates were measured at San Rafael (1992) and Malargüe (1993) ore tailing sites, and the range of values obtained were 9-11 $Bq/m^2.s$ and 10-17 $Bq/m^2.s$, respectively. The contribution of ^{222}Rn at the natural background concentration, due to the ore tailings, is not significant at a few hundred meters from the emanation points.

Taking into account all the measurements of the environmental radiological monitoring program for the last 14 years, it may be concluded that no remarkable exposure occurs for the population living in the vicinity of the uranium mining and milling sites in Argentina due to their operation or their wastes.

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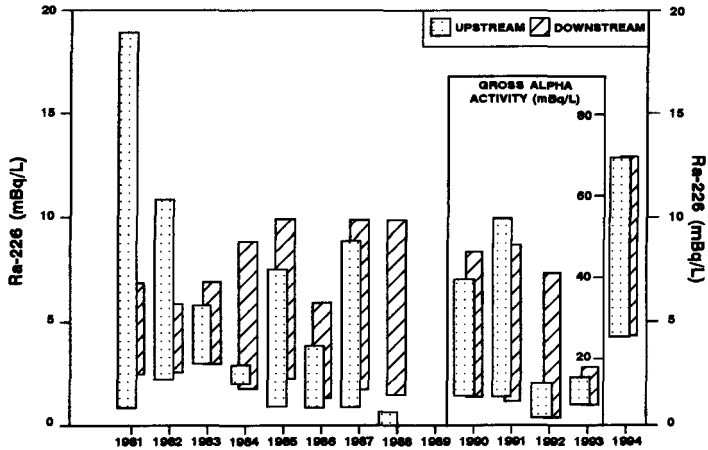

Figure 2: Range of ^{226}Ra concentrations and gross alpha activities measured in surface waters, < 10 km upstream and < 50 km downstream from the San Rafael plant (Mendoza), 1981-1994.

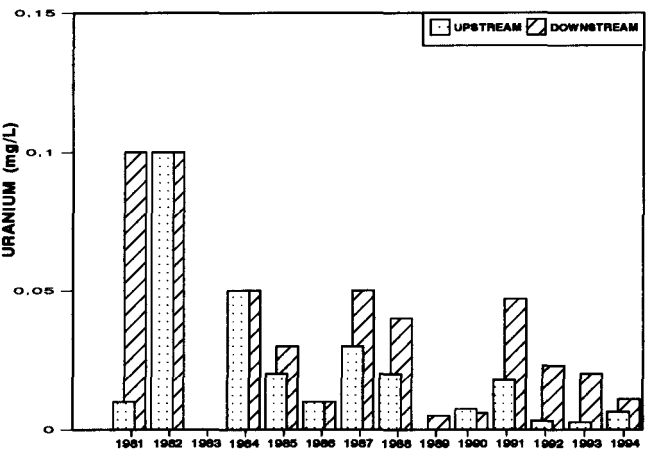

Figure 3: Maximum natural uranium concentrations measured in surface waters, < 10 km upstream and < 50 km downstream from the San Rafael plant (Mendoza), 1981-1994.

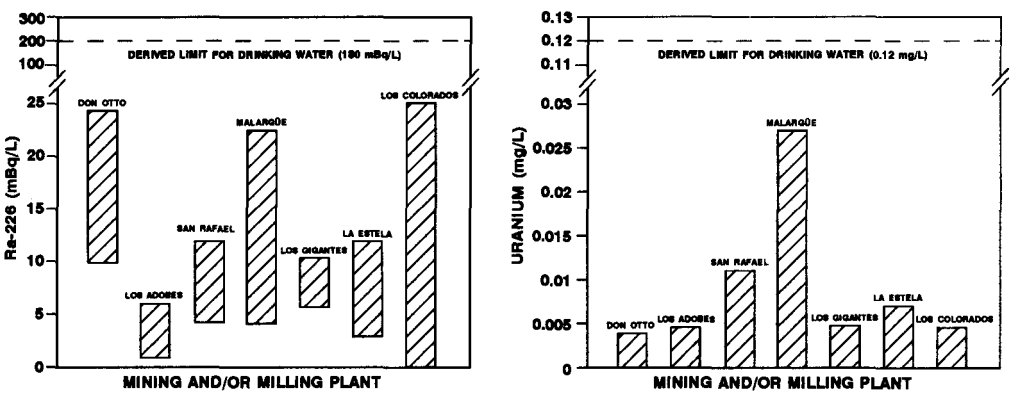

Figure 4 and 5 : Range of ^{226}Ra concentrations and maximum natural uranium concentrations measured downstream, < 50 km from the plants, for the last year assessed.

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PAPER TITLE SHORT-TERM ENVIRONMENTAL MEASUREMENTS WITH THE HIGH SENSITIVE THERMOLUMINESCENT DOSEMETERS

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MAJOR SCIENTIFIC TOPIC NUMBER 4.1. (see page 7)

ABSTRACT (See instructions overleaf)

Considering that average environmental daily doses are in the range of 1-5 uGy, it was performed an experiment to check the real capability of the ultra sensitive TL phosphors to detect extremly low doses. The results obtained with four types of TL dosemeters such as LiF:Mg,Cu,P (GR-200A), solid sintered $CaSO_4$:Dy (TLD-2000H), $CaSO_4$:Tm (TLD-2001H), and Al_2O_3:C (TLD-500K), as well as two types of electronic dosemeters, have been studied, taking high pressure ionisation chamber as the reference instrument. The evaluation were performed using different types of TLD readers.

The experiments were also performed in order to study the additivity, stability and thepossibility to detect small variations in daily doses for different exposure periods, from one week to 3 months period.

The dosemeters were exposed to environmental radiation at the selected locations: inside the Institute Vinča, on the site of the Accelerator Installation (in construction) to establish the zero point levels with respect to dose rates, two years before starting, and at the standard environmental measuring location, 1.5km far from the Institute.

It was found that all the dosemeters were capable of reliably measuring these very low doses after an exposure of 1 day, without performing special measurement procedures.

SYSTEMATIC APPROACH TO THE ORGANIZATION OF MEASUREMENT ASSURANCE IN RADIOECOLOGY

I.M.Khaikovich (A.A.Logachyov Institute, VIRG-Rudgeophisika)

A.M.Blumentsev (VNIIGeosystem)

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1. Radiation conditions of the environment are characterized by a great number of various parameters which, relative to radiation, describe different objects and for the measurement of which the use is made of various analytical methods in field and laboratory conditions. To ensure the required level of reliability of the results of radioecological studies on large territories, it is suggested to set the list of parameters to be measured depending upon the problem to be solved and the object under study, as well as to establish the requirements for the accuracy of determination of those parameters.

2. All radiometrical methods used in radioecology can be divided into three groups according to the measurement conditions. The first group includes measurements directly in natural beddings, i.e. in blast-holes/wells and on the day surface. Parameters: fractions of total mass and specific effective activity of natural radionuclides (NRN), specific and surface activities of radionuclides of artificial origin (ARN) and value of their reserve per unit of surface, air kerma (dose equivalent) rate of gamma-radiation.

The second group contains radon measurements in the earth (soil) and free air. Parameters: radon concentration, volume effective equilibrium activity of radon and daughter products of its decay.

The third group consists of methods of lithochemical sampling of water and bottom sediments. Parameters: NRN and ARN concentration in water, NRN and ARN content in bottom sediment samples.

3. Parameters characterizing radiation sources in every group are mutually connected. Thus, to provide for the measurement assurance, it is necessary to correlate parameters of standard measuring means (MM) applied for organizing the measurement assurance (MA), by choosing a metrological model reflecting adequately the conditions of measurement of the physical parameters in every group and of certification of the standard MM's by the set of parameters to be measured. The metrological model describing the measurement of parameters with a guaranted uncertainty, must be chosen by the following scheme:

physical object))→ physical model → mathematical model → metrological model

4. Physico-mathematical analysis carried out for the first-group methods, establishes the correlation between the parameters of this group. For this group there are stated requirements to the unified measurement chain for the MA of radiometers applied in complex radioecological studies. Such a chain should have at least three levels.

The first level of standard reference MM's (SRMM) - standards includes:

a) Set of standard reference materials (SRM) of the NRN and ARN content and the ARN surface activity to maintain and reproduce the units of fractions of total mass of NRN's and of specific activities of ARN's in the model "homogenious space" or "homogenious half-space", as well as those of surface activity of ARN's of the

specified content in the model "thin film". SRM dimensions must ensure "saturation by gamma-radiation" in the measurement condition of "difference effect": for 4 geometry - 140 cm in diameter and 150 cm in height with the bulk density 1.5 g/cm^3; for 2 geometry - 100 cm in diameter and 50 cm in height for the model "homogenious half-space" and 30 - 50 cm in diameter for the model "thin film".

b) Working standard to maintain and reproduce the unit of air kerma (equivalent dose) rate of gamma-radiation.

c) Standard multichannel gamma-radiometer (gamma-spectrometer) to control stability of the parameters of the SRM set and to transfer the units to standard and working MM's.

The second level consists of standard MM's. To measure in 4 geometry, standard MM's are built as SRMM's. For ground-based and aerogamma- surveys it is recommended to use as standard MM's the SRM's in the form of testing grounds certified by transferring the unit from SRMM's.

The third level includes working MM's.

5. The suggested measurement chain ensures for working MM's the estimation of parameters by the first-group methods with the confidence basic uncertainty less than 10 % when measuring in 4 geometry. The confidence uncertainty won't be higher than 20 and 30 % (0.95) when measuring the surface activity and air kerma (equivalent dose) rate of gamma-radiation by the methods of ground-based and aerogamma-surveys, respectively.

RADIOACTIVITY CONCENTRATIONS IN SOIL IN THE WESTERN PROVINCE OF SAUDI ARABIA

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INTRODUCTION

Radioactivity in soil was measured in the seven Western Province cities of Saudi Arabia in order to establish radiation data baseline of natural as well as man-made radionuclides. These cities were: Makkah, Taif, Baha,Abha, Madeena, Yanbu and Tabuk, whose locations is shown in Fig. 1.

Worldwide concern about the chernobyl accident caused interest in measuring radioactive contamination. Concern was raised on whether radioactive contamination had reached Saudi Arabia, particularly that few models have predicted that small amount of radioactivity may have reached the country (1,2). Only long lived radionuclides, such as ^{137}Cs, of 30 years half-life would remain after long time. Its concentration would give a good indication of the degree of contamination. The other long lived radionuclide is ^{90}Sr of 28.5 y half-life, but this radionuclide is less volatile than Cs. It is, therefore, expected to travel much less than Cs. The reported $^{90}Sr/^{137}Cs$ is Europe (3) was only few percent. The ratio in Saudi Arabia is expected to be less due to the larger distance from the reactor accident location.

MATERIALS AND METHODS

Seven soil samples each about 2 kg were collected from approximately top 5-8 cm, of undisturbed soil by excavation or construction, from each city. The detector used was a vertical type of 10% relative efficiency HPGe whose FWHM was about 2 keV at 1.332 MeV. It was shielded by lead blocks of 30 cm thickness with additional cooper shielding of 5 mm around the detector, inside the lead shield. Standard sources were prepared from known radioactivity ^{152}Eu and ^{137}Cs sources. The ^{232}Th concentration was found out from its daughters ^{228}Ac, ^{212}Bi, ^{212}Pb and ^{208}Tl, that of ^{226}Ra from ^{214}Bi and ^{214}Pb. The counting time for each sample was 72000 s.

RESULTS AND CONCLUSION

The mean concentration values of ^{232}Th, ^{226}Ra, ^{40}K and ^{137}Cs are shown in Figs.2-4. The concentration values of ^{226}Ra was higher in all cities except in Tabuk. The distribution of samples concentrations of ^{226}Ra, ^{232}Th and ^{40}K belonging to any city was not much different from that of all samples from all cities. Most values of ^{226}Ra and ^{232}Th were between 10-25 Bq/kg and that of ^{40}K between 400-700 Bq/kg.

The concentration of ^{137}Cs is shown in Fig. 4. In the cities of Makkah, Yanbu, Madeena and Tabuk, its values did not exceed 5 Bq/kg, with most values were around 2 Bq/kg, expected to be coming, mainly, from fallout of nuclear explosions above ground. The concentration values in high elevation high rainfall cities of Abha, Baha and Taif was clearly higher. Relatively wider variation in concentration values were observed among the samples. The nonuniform higher concentration is in these cities expected to be due to Chernobyl reactor accident, which may have precipitated by rainfall. The distribution of concentration values of Makka is shown in Fig. 5, and that of Abha in Fig 6, clearly indicating the difference in the concentration distribution fashions

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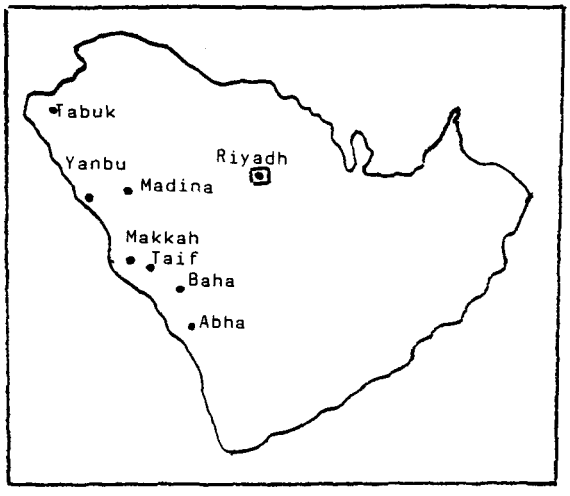

Fig. 1. Location of cities from which soil samples were collected.

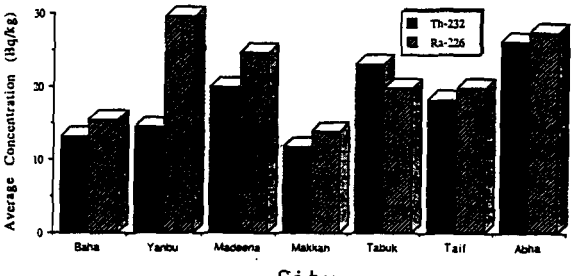

Fig. 2. Average Th-232 and Ra-226 concentrations in soil from seven cities.

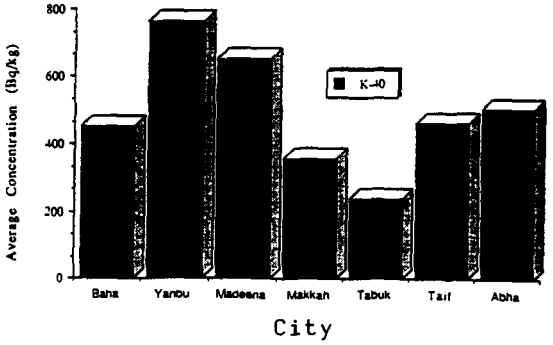

Fig. 3. Average K-40 concentration in soil from seven cities.

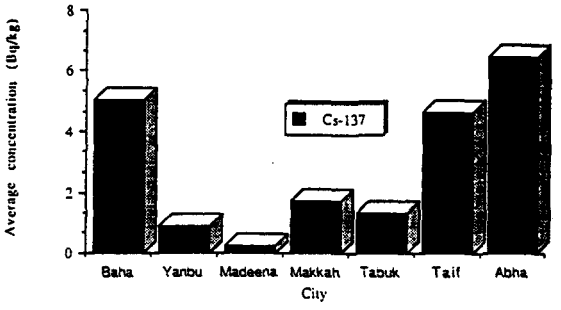

Fig. 4. Average Cs-137 concentration in soil from seven cities.

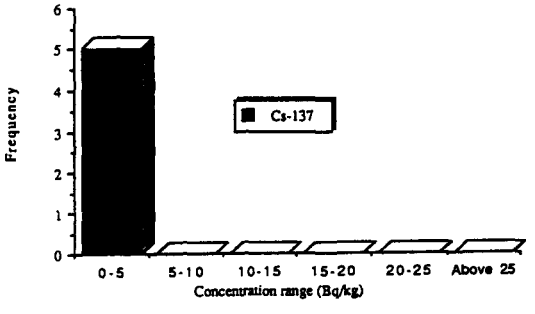

Fig. 5. Distribution of Cs-137 concentration in Makkah soil.

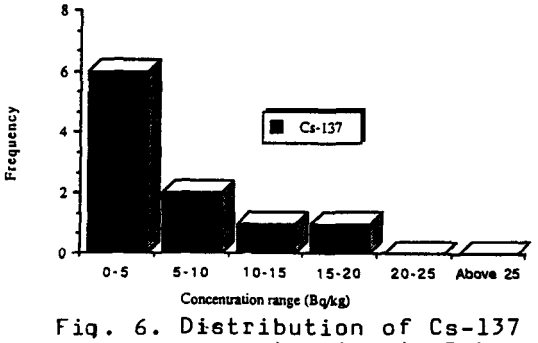

Fig. 6. Distribution of Cs-137 concentration in Baha soil.

DECONTAMINATION OF RADIOACTIVE PHOSPHOGYPSUM

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INTRODUCTION

Phosphate fertilizer industry is a source of concern for environment mostly by phosphogypsum resulted as waste in the sulfuric acid attack of phosphate rock. Phosphogypsum is obtained in large amounts and is found in deposits located around fertilizer plants. There are four plants in Romania and the same time four phosphogypsum deposits of approximately 8 millions tons resulted in the dihydrate process of sulfuric acid attack of phosphate rock. The phosphogypsum acumulated all over the world is very large perhaps more than 1 billion tons.

The main problem consists of phosphogypsum radioactivity where all ^{226}Ra from natural sedimentary rock is present being a serious hazard radioactivity source. The radioactivity of romanian phosphogypsum fluctuate between 7 and 30 pCi/g ^{226}Ra.

Due to its radioactivity, restrictions in phosphogypsum use in construction industry or in agriculture have been imposed in many countries. The environmental contamination by phosphogypsum was in attention of many scientists and various studies have been carried out in view of eliminating ^{226}Ra and creating the conditions of its use in various fields (1-4). This study is an atempt to solve this problem.

EXPERIMENTAL

In this work three hypothesis were taken into consideration:

- radium removal from phosphogypsum by treatment with ammonium salt solutions which increase radium sulfate solubility;
- physical concentration of radium with hydrocyclones since experimental data have shown that ^{226}Ra is concentrated in fine grains of phosphogypsum;
- phosphogypsum (calcium sulfate) conversion to sodium or ammonium salts by treatment with sodium hydroxide, sodium carbonate or ammonium carbonate and radium retention on ion exchangers from the solution resulted by dissolution in hydrochloric acid of phosphogypsum conversion residue.

The research work was carried out at laboratory scale and in pilot plant. Phosphogypsum was obtained from SOFERT S.A. Bacau and had an average radioactivity of 17 pCi/g. Technical reagents have been used. Laboratory experiments were carried out in usual glass equipment. The experiments from pilot plant were made in suitable equipment acid and base resistant. ^{226}Ra determination was carried out by emanometric method.

RESULTS AND DISCUSSION

Phosphogypsum treatment by ammonium salts solutions gave no significant results, and the radium rate removal from phosphogypsum was very low. The requirement of unitary washing operations being over hundred.

The hydrocyclone separation using a unit of 100 mm diameter led to two granulometric fractions of phosphogypsum:+45 with radioactivity of 7.8 pCi/g and -45μ with radioactivity 27.9 pCi/g. The +45μ fraction, representing approx.40% weight may be used according to some literature data in the material of

construction industry.

In order to convert calcium sulfate (phoshogypsum) to sodium or ammonium sulphates, the chemical treatment is carried out and the conversion yields obtained are given:90-95% with 700 kg NaOH/t phosphogypsum or 600 kg ammonium carbonate/t phosphogypsum, consumption.

The hydrochloric solution containing radium and calcium, obtained from dissolution of residue resulted in the conversion process of phosphogypsum, by hydrochloric acid with dissolution yield of 98.9% and acid consumption of 600 kg/t phosphogypsum.

The efficiency of various ion exchangers for radium retention from hydrochloric solution was also studied. The best results were obtained with ion exchanger VIONIT CS-34S made by VIROMET S.A.-Victoria (Romania). The loading curve with ^{226}Ra of this resin is given in Fig.1, where experimental conditions are mentioned.

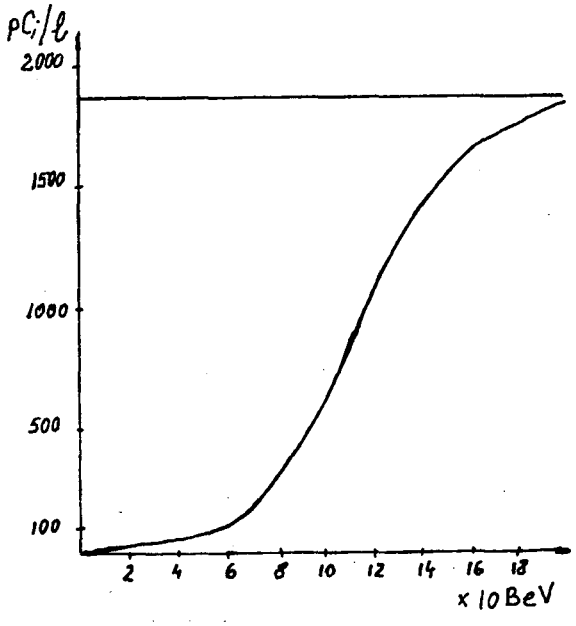

Fig.1. SORPTION ISOTHERM OF ^{226}Ra ON VIONIT CS-34S RESIN.

The pilot plant experimental data have confirmed the laboratory results and led to the flowsheet of the process for radioactive elimination and phosphogypsum conversion as given in Fig.2.

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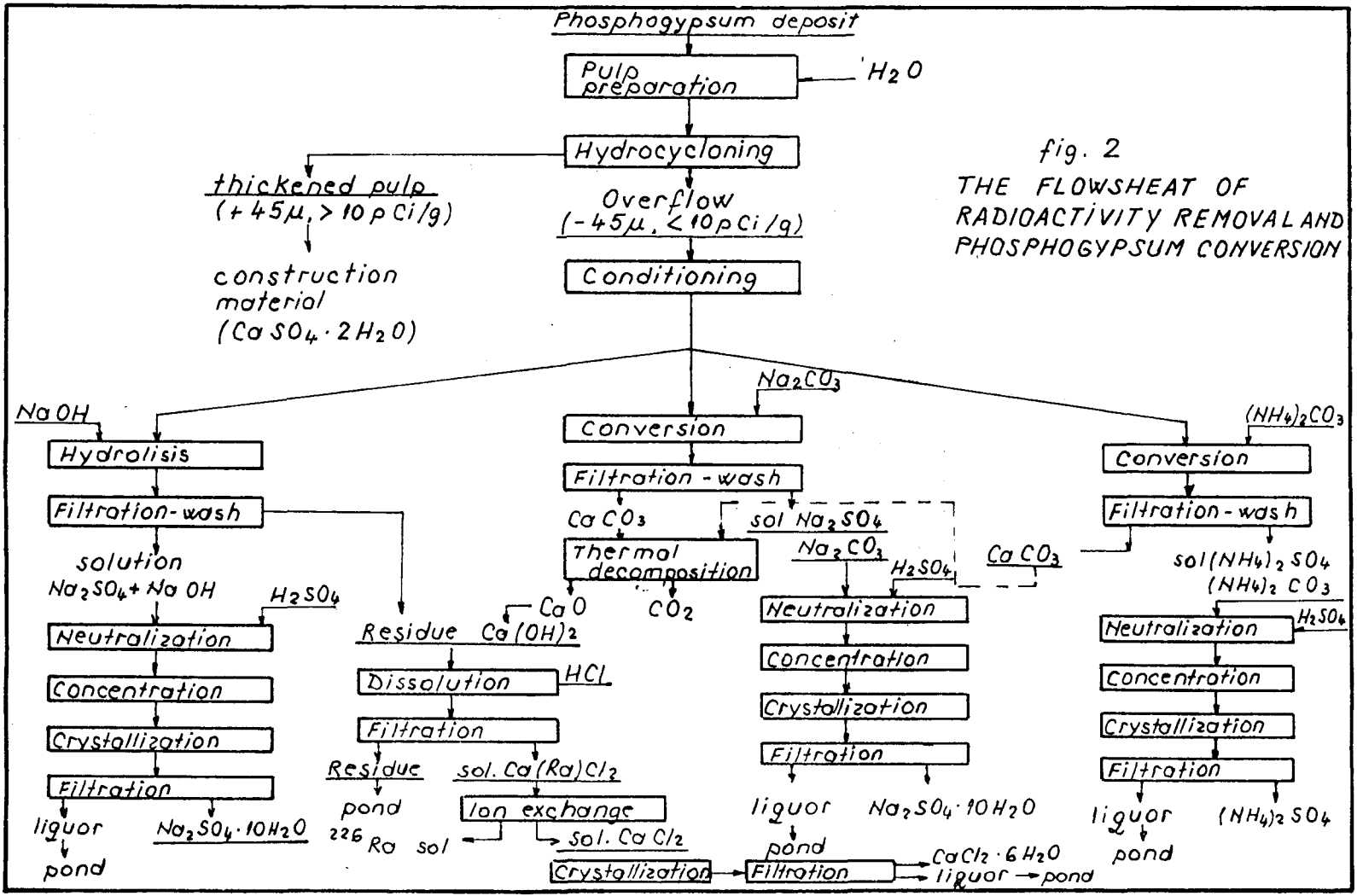

fig. 2
THE FLOWSHEAT OF RADIOACTIVITY REMOVAL AND PHOSPHOGYPSUM CONVERSION

INVESTIGATIONS OF THE BEHAVIOR OF ^{110m}Ag IN SOIL SYSTEMS

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INTRODUCTION

^{110m}Ag ($t_{1/2}$ 250.4 d) is a little-studied radionuclide that arises as an activation product in nuclear power plants. Although it has featured little in studies of environmental pollution from nuclear reactors, it is present in the low level liquid effluents released from some pressurized water reactors under normal operating conditions (1) and has been reported to be 10% of the total gamma radioactivity in some effluents (2). In 1988 760 MBq of ^{110m}Ag was released into the atmosphere from the Hungarian pressurized water reactor at Paks (3). Its presence in the plume from Chernobyl has been noted (4) and in West Cumbria, in May 1986, monitoring studies revealed that ^{110m}Ag from the Chernobyl plume was 3 - 5% of the ^{137}Cs found on vegetation (5). These releases have prompted us to consider the extent to which radiosilver may move through the environment (6).

The sorption processes of radiosilver on various soil were investigated by batch technique. The effect of the soil:solution ratio have been determined and the results expressed as the distribution coefficient, R_d.

MATERIALS AND METHODS

The soils used in this investigation were sampled from the upper 30 cm layer, stored moist at 4 °C, then air-dried and sieved to 200 μm. Some of their properties are given in Table 1. The isotope, ^{110m}Ag, was obtained from Amersham as the nitrate salt in 1M HNO_3 (44 MBq in 1 cm^3) and was used without further preparation.

Soil suspensions were prepared as follows: soil (1 g) was accurately weighed into polypropylene tubes, then 2-20 cm^3 solution containing 4 mg/dm^3 K^+ and 20 mg/dm^3 Ca^{2+} pipetted to give a solution ratio in the range 1:2 - 1:20 kg/dm^3. The activity of the solution was 10-20 kBq (0.54-1.09 x 10^{-6} g; 4.85-9.90 x 10^{-9} mol). All suspensions were prepared in triplicate. The suspensions were shaken in a reciprocation shaker at 20 °C. Aliquots (1 cm^3) were removed at 10, 30, 60, 120, 240, 1440 and 2880 min. for γ-counting. The γ-activity originating from the radionuclide was determined by counting 658 KeV photon emission of ^{110m}Ag on a γ-ray spectrometer NK-350 with a NaI(Tl) scintillator and an NZ-138 well-type detector shield (GAMMA, Hungary).

Table 1. Physical-chemical characteristics of soils used in the experiment

Soil type	leached Ramann forest soil	chernozem light sandy soil	alluvial soil	calcareous chernozem soil
Location	Gödöllõ	Õrbottyán	Foktõ	Nagyhörcsök
Soil code	1	2	3	4
pH (KCl)	5.0	7.6	7.5	7.8
Organic matter (%)	1.05	1.26	1.53	3.40
CEC (meq/100 g)	8.9	11.0	11.3	32.2
exchangeable K (meq/100 g)	0.35	0.12	0.37	0.28
exchangeable Ca (meq/100 g)	1.43	4.09	8.57	10.32
Coarse sand (%)[a]	32.9	11.9	0.4	0.8
Fine sand (%)[a]	41.0	79.4	50.0	15.7
Silt (%)[a]	17.8	4.9	43.1	60.4
Clay (%)[a]	8.3	3.8	6.5	23.1

[a] Percentage of mineral fraction only

Calculation of the distribution coefficient

The distribution coefficient (R_d) of solid phases was calculated from the counts of the solution taken as follows:

$$R_d = \frac{\text{radioactivity of Ag sorbed kg}^{-1}\text{ sorbent}}{\text{radioactivity of Ag in solution after sorption dm}^{-3}\text{ solution}}$$

RESULT AND DISCUSSION

Kinetic studies are a common prerequisite to equilibrium sorption experiments. By monitoring the distribution ratio (R_d) as a function of time a 'time invariant' value for R_d is obtained. The uptake of radiosilver from solution on different soils was studied for contact times varying from 10 min. to 2 days. From Fig. 1. it can be seen that 'equilibrium' is reached within 2 days in each case and that there are different sorption rates for all the soils. From the Fig. 1. a binding strength order for soils can be set up: $1 < 2 < 3 < 4$. From this order it could be concluded that the mobility of radiosilver in these soils is the reverse.

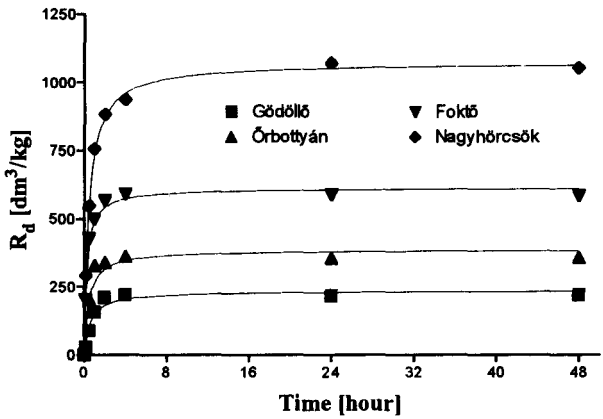

Figure 1. Sorption of ^{110m}Ag on soils vs. time (soil:solution ratio 1:20)

Staunton and Nye (7) used the variation of distribution coefficient with soil:solution ratio to extrapolate to the value of K_d on moist soils. They found good agreement between calculated and measured values of distribution coefficient when it was possible to extract solution by immiscible displacement (8). To use of the distribution coefficient measured in different soil:solution ratio to real natural moist soils we have investigated the effect of soil:solution ratio on the distribution coefficient (R_d). Fig 2. shows the effect of the soil:solution ratio, in the range 1:2 - 1:20 kg dm^{-3}, for each of the four soils. The effect of soil:solution ratio is complex: in general R_d increases as the ratio is decreased. However this is not always the case, for example maxima is observed for Nagyhörcsök. Other authors (9, 10) have got the same results investigating the effect of solid:solution ratio on the distribution coefficient. Unfortunately, adequate explanation for the effects has not been established.

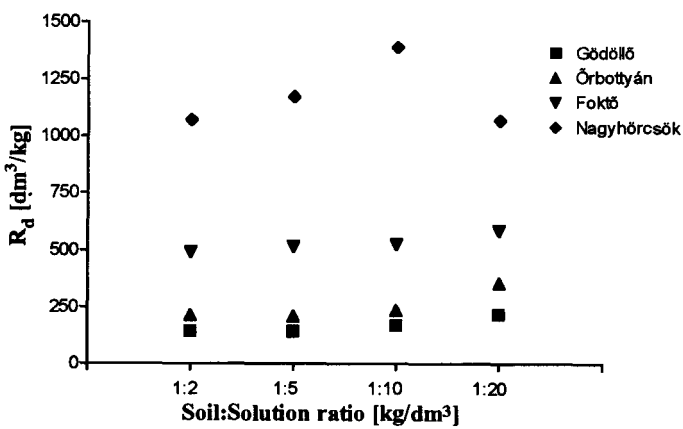

Figure 2. Effect of soil:solution ratio on the distribution coefficient of ^{110m}Ag in soils

Our previous investigation (6) shows that the soil humic matter the main sink for radiosilver in soils. From an investigation of characteristics of studied soils (see Table 1.) we have got positive correlation between radiosilver binding in soils and soil organic matter content (Fig. 3.). From Fig. 3. it is apparent that the R_d values increase monotonously as the humic content of soils increases . Other researchers (11,12) have also noted this positive correlation between Ag^+-binding in soils and soil organic matter.

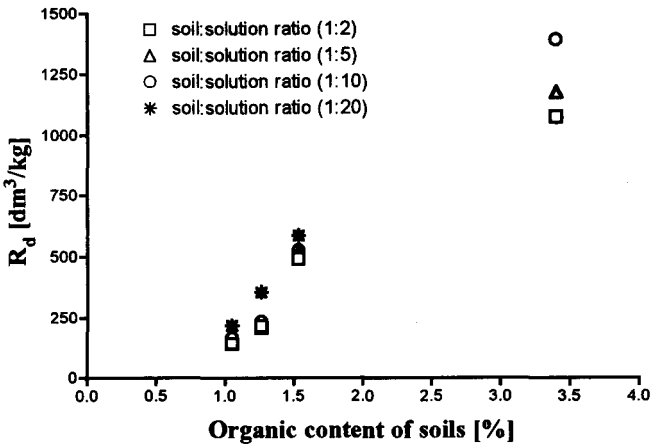

Figure. 3. Effect of soil humic matter content on the distribution coefficient of ^{110m}Ag

CONCLUSION

In conclusion, the distribution coefficient, R_d is a simple, convenient and useful measurement. However care must be taken both to avoid experimental artifacts and in the extrapolation of measurements to *in situ* conditions. In order to extrapolate to an *in situ* distribution, sorption must be studied from different point of view.

This study has shown that soils with different properties bind the radiosilver with different strengths. In addition it has been shown that the main sink for radiosilver in soils is the humic matter.

ACKNOWLEDGMENTS

This project was partially funded under the Hungarian OMFB Radiohygiene Protection Programme (Contract OMFB 94-97-47-0822). Part of the research described has been conducted under contract from the Commission of the European Communities Radiation Protection Programme, contract number ERB-CIPDCT930429F13P-CT920010.

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A METHOD FOR THE DETERMINATION OF VERTICAL DISTRIBUTION PROFILES OF RADIOACTIVE CONTAMINATION IN SOILS

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ABSTRACT

A method for precise determinations of vertical profiles of radioactive contamination in soils is described.

A polyethylene tube is driven into the ground and the tube, together with a soil core, withdrawn. The tube and the core are frozen and the whole sectioned with a diamond saw. The technique produces 'precision engineered' parallel-sided circular discs. Discs of less than 1 cm in thickness can be produced if desired, thus permitting a well resolved soil profile. The individual sections are dried, weighed and analysed.

The method was used in the city of Gävle in Sweden and showed that more than eight years after contamination by fallout from Chernobyl, the radiocaesium concentration in undisturbed soil (sandy loam) peaked at a point several centimetres below the ground surface. Trace amounts of radiocaesium were found to have migrated to a depth of at least 15 cm. The data obtained are suitable for testing models designed to represent downward migration of caesium fallout in soils and for calculating dose-rate over undisturbed soils contaminated with radiocaesium.

INTRODUCTION

Computer modelling, as well as direct dose rate measurements, has shown that when a housing environment has become contaminated by airborne radiopollutants (essentially ^{137}Cs) deposited after a nuclear accident, areas of soil in the environment are often the main contributors to the dose rate. This is the case immediately after the deposition, and as time goes by the relative dose rate contribution from soil areas will normally increase. The reason for this is the relatively strong fixation of radiocaesium in the upper soil layers. Many different decontamination or reclamation procedures have been suggested over the years to reduce the external radiation dose to populations by for instance burial deep in the vertical profile of a thin top soil layer containing almost all the radionuclides or by total removal of a thin top layer.

The crucial question is : how deep a layer should be removed ? This question can not be answered by the generally applicable computer models of radionuclide migration, which are not sufficiently case-specific. If too little is removed, the reduction in dose rate may not be worth the effort, whereas if too much is removed, areas with thin fertile layers (typical of e.g. large areas of the former Soviet Union) could be turned into desserts. It is therefore very important to know the vertical contaminant profile before any clean-up operation is initiated. Refined techniques have been developed, by which estimates of 'mean' penetration depths can be made from in situ measurement spectra obtained at the surface, but these assume an exponentially decreasing pollutant depth profile, and this assumption is not valid in reality. Soil profile sampling is therefore the only certain way to investigate soil radionuclide penetration.

METHODS

Cylindrical polyethylene tubes with an inner diameter of 81.0 mm and a wall thickness of 4.5 mm are driven into the ground using a rubber hammer. When the full length of the tube (about 20 cm) has penetrated the ground, it is withdrawn along with a soil core. The withdrawal is performed with special care to prevent loose material in the bottom from falling out. Therefore, the soil surrounding

the outer surface of the tube is first excavated. The sample cores (still in plastic tubes) are wrapped in a thin plastic film to prevent them from drying out and falling apart. When the sample cores have been transported to the laboratory, 200 ml of distilled water is added to each tube, which is subsequently transferred to a deep freezer (-20°C), where it is left for two days.

The tubes with the samples are then sliced with a diamond saw. The blade of the saw is 2.57 mm thick. Consequently, by this slicing procedure, a corresponding layer of the soil core is lost by each sectioning. A spacing device on the diamond saw makes it possible to accurately adjust the thickness of the slices. The saw blade usually has to be cooled with water, but at this point the samples contain sufficient amounts of water to grease/cool during the slicing process. As the samples are deep frozen the slices keep their parallel-side circular disc shape and even stones in the samples are cut through. The reproducibility of this 'precision engineered' technique has been evaluated through an examination of the sectioned core samples, which shows that the variations in thickness have a standard deviation of about 1 %.

Since the samples stay in the plastic tube during the slicing process, the sliced tube sections form rings around the sample slices. These rings support the samples during the different analysis sequences (drying, weighing and γ-measurement). Further, a removal of the whole soil core from the tube would introduce an unnecessary cross-contamination hazard.

The samples are then dried in 3 days at 60°C. After drying, the density of the samples is established through a weighing, where the mass of the plastic ring is subtracted. Stones and grass roots are not removed. This enabled a determination of the bulk soil density as a function of depth. The density of the sample is also used to allow for the attenuation of gamma-rays through the soil slice while measuring the gamma-activity.

The photopeak count rate of ^{137}Cs (661.6 keV) in each sample is measured in a lead shielded gamma spectrometer which includes a 15.6 % efficiency Ge(Li) detector. In this calibrated detector system the sample was placed directly on top of the upward facing detector end cap.

Finally, the photopeak net count rate is converted to a contamination level (Bq/cm^3 of ^{137}Cs) in the sample, taking into account differences in slice thickness and densities relative to the calibration standard.

RESULTS/DISCUSSION

The above described method was applied in situ in the Gävle area of Sweden in the summer of 1994. This area received high levels of contamination after the Chernobyl accident in 1986. Thus, soil core samples from the area contain sufficient amounts of radiocaesium to perform analyses with good accuracy. The soil profiles were sampled at five different locations in the area. At each location a series of 5 cores was sampled: one in each corner of a square with a side length of 1.40 m and one in the centre of the square.

Measurements of the soil density at the 5 different locations showed almost identical profiles. The bulk density of the surface layer (ca. 1cm) was found to range from 0.36 to 0.47 g/m^3. It was found that the bulk density increases almost linearly with depth through the top 10 cm layer. Below 10 cm, the soil has a nearly constant density, ranging from 1.2 to 1.5 g/cm^3. This is approximately the value that has been used as a default value for the whole profile in existing biospheric transfer models (1,2). Only the top layers of the beach samples were found to differ significantly from the rest. Here, the top layer density was found to be only 0.24 g/m^3, which can hardly surprise since this top layer was sand. The generally lower density of the top layers of the other samples must be attributed to a high content of organic matter, which has previously been recorded by textural analysis of soil from the area (3). This lower density in the top layers is expected to influence not only the shielding against radiation from penetrated radiocaesium, but also the downward migration of radiocaesium in the soil.

Figure 1 shows the recorded vertical soil profiles (averages over 5 cores). Although the total amounts of contamination were found to vary by as much as a factor of 1.5 between the individual cores, the shape of the profile was found to be fairly consistent. The counting uncertainties were less than 1 %. Samples were taken from the lawn in front of the city hall, a local surveyor's office in the city area (Land C.), a lawn in an industrial area (Gevalia), a lawn about 13 km to the north-east of the town

centre (cottage) and a beach area about 11 km to the north of the town centre. Also shown is an estimate made with the computer model MLSOIL (1) of the profile, assuming a k_d value of 37 ml/g, which is actually the minimum value that has been used so far by the model for uncertainty analysis.

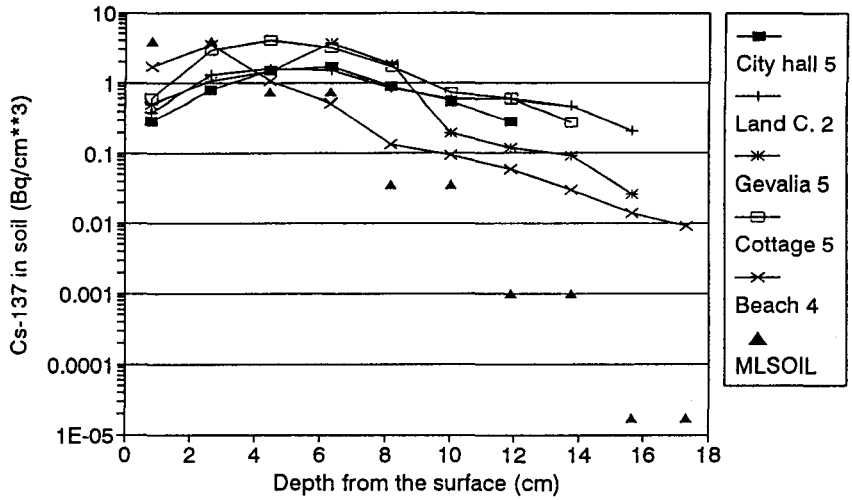

Figure 1. Vertical ^{137}Cs profiles in soil in 5 different lawns in the Gävle area. The sampling results are compared with estimates obtained with the MLSOIL model.

As could be expected from the density measurements, the higher content of organic matter and consequently higher macro-porosity and smaller fixation potential of the top soil layer is reflected in the radiocaesium profile. However, compared with previous investigations in the same area (the lawns termed 'Gevalia' and 'Cottage') in 1988 and 1990 (3), the average depth of the contamination has increased rather much. In 1990 the profile of the 'Gevalia' soil peaked at a depth of about 1 cm. Anyway, it has certainly been demonstrated that the assumption that soil contaminant content decreases exponentially with depth is not generally valid. After longer periods of time, the contaminant content in the uppermost layer can be depleted.

CONCLUSIONS

A specially adapted technique for soil sampling was established and tested in the field in the Gävle area of Sweden. By this method, the sectioning of soil cores can be performed with very good accuracy (ca. 1 %). The gamma analysis of trace amounts of radiocaesium in the soil slices from Gävle (initial contamination level ca. 200 kBq/m^2) was performed with an uncertainty of less than 1 %. Although the total contents of radiocaesium differed between the individual samples taken at a site, the shape of the vertical caesium profile was found to be relatively consistent. The radiocaesium concentration was found to peak at a depth of several centimetres in the investigated sandy loam type soil.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE The Technique of the I-129 Amount Measurement in the Soil Samples Using Radiochemical and Neutron Activation Analysis.

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MAJOR SCIENTIFIC TOPIC NUMBER 4.1 (see page 7)

ABSTRACT (See instructions overleaf) The original technique of the I-129 amount measurement in the soil samples using radiochemical and neutron activation methods was developed in the IPPE. Its main stages are the following :

- isolation of iodine from a sample by extraction with transfer of iodine into PbI_2;
- sediment irradiation in a reactor channel in a flow of thermal neutrons not less then 4×10^{13} $n/cm^2 \times s$ for transfer of I-129 in I-130;
- repeated precipitation of PbI_2 in CuI_2 after irradiation for decrease of interfering elements contents (Na, Br, Sb and others) up to a acceptable level;
- measurement of I-130 activity on a gamma-spectrometer device with Ge-Li detector;

The technique was tested on the soil samples which were taken from different regions of Russia contaminated as a result of Chernobyl accident (Cs-137 surface contamination ranges from 5 up to 18 Ci/km^2), and also in the vicinity of nuclear installations (near Obninsk).

The technique permits to determine the I-129 amount in quantities up to 3×10^{10} nuclee/kg at the I-129/I-127 ratio not more then 10^{-12}.

The limits of the technique sensitivity permit to use it not only for the restoration of "iodine impact" characteristics in the first period after nuclear power plant accident but also for the I-129 measurement in global fallout as a result of nuclear weapon tests and registration, control and reconstruction of iodine fallout near nuclear power plants.

Study on the Behavior of Technetium-99 in Soil Environment

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1.Introduction

Technetium-99 (Tc-99) is an important radionuclide for environmental assessment around nuclear fuel cycle facilities, because it has a long half-life and relatively high mobility in the environment. Its determination method, distribution and behavior in the environment have been studied.

While the method using an anti-coincidence type low background gas flow counter has been principally utilized as the conventional determination method for Tc-99, in recent years other determination methods such as neutron activation analysis and liquid scintillation counting method have also been considered. However, these methods are not effective to determine Tc-99 in environmental samples, because their detection limits are not low enough and measuring times are needed.

Inductively Coupled Plasma Mass Spectrometer (ICP-MS) is a quantitative and qualitative analysis apparatus having superior characteristics such as extremely low detection limit, high resolution, high precision, short measurement time and capability for simultaneous multi-elements analysis.

This study developed an analysis technique to acquire field data. As the results, several information regarding the distribution and behavior of Tc-99 in the surface soil environment was obtained.

2.Experimentation

The ICP-MS utilized was the PMS-2000 manufactured by Yokogawa Electric Corp. Japan. In order to enhance amount of sample feeding into plasma, the MISTRAL, a new type nebulizer made by ARL Corp. Switzerland, was connected with the sample feeding system. A schematic diagram of the ICP-MS is shown in Fig.1. The ICP-MS consisted of a nebulizer, ICP as an ion source, sampling interface, quadrupole mass filter, and detector. A solution sample was nebulized into the aerosol at the nebulizer and then ionized in the ICP. the ionized elements passing through the interface were focused by ion lens and then analyzed by the qua

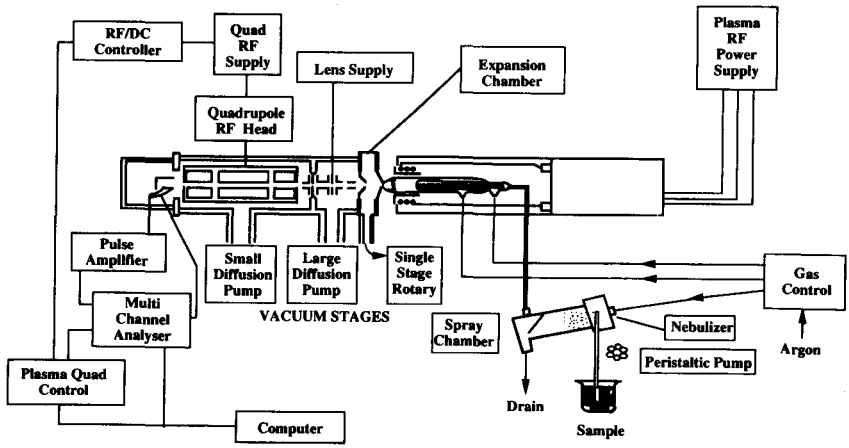

Fig.1 A schematic diagram of ICP-MS

3.Results and discussion

3.1 Development of high sensitivity measurement method for Tc-99

3.1.1 Ashing conditions

Because of high volatility of Tc-99, its volatilization loss might occur, if ashing is applied to dried samples without pretreatment. Therefore, a treatment method in which Tc-99 was changed to a nonvolatile chemical form before ashing with reductive agents such as hydrochloric acid and ammonia solution was investigated. As a result, ammonia solution pretreatment could improve the chemical recovery of Tc-99 to about 90%, which was 10% higher than that without the pretreatment.

3.1.2 Matrix effect on ICP-MS

If the concentration of matrix elements in a sample is high, the signal intensity of each peak in the ICP-MS is decreased due to plugging of the hole on the cone at the sampling interface. Therefore, to overcome this problem some countermeasures such as elimination of organic matrix by dry ashing and elimination of inorganic matrix by combined treatments of iron hydroxide coprecipitation, anion exchange and solvent extraction were carried out.

Table 1 Decontamination factors of Ru and recoveries of Tc for some separation methods

Separation method	D.F. of Ru	Recovery of Tc (%)
Chelate exchange	1.1×10^5	97
Solvent extraction		
TIOA-1M HNO_3	11	96
TBP-0.1M HNO_3	35	91
$C_6H_{10}O$-1M HNO_3	75	71
$C_6H_{10}O$-1M K_2CO_3	1.9×10^5	80
Coprecipitation (TPAC)	27	66

3.1.3 Elimination of the interference elements

If an isobar element is contained in a sample, it becomes impossible to determine the target radionuclide, because both the target radionuclide and its isobar have the same peak position in the mass spectrum. For example, ruthenium-99 (Ru-99) interferes with the determination of Tc-99. A countermeasure for this interference is to eliminate the isobar nuclide by using chemical separation treatments.

The results of decontamination factor tests for eliminating these interference elements using tracers showed that the chelate resin ion exchange method was useful for eliminating Ru-99 (Table 1).

3.1.4 Development of systematic analysis method

The results of the tracer tests indicated that the iron hydroxide coprecipitation method was suitable for separating Tc-99, and that the nitric acid form ion exchange method was appropriate for separating alpha emitting radionuclides such as Np-237 and Pu-239,240. The systematic analysis method developed in this study is shown in Fig.2.

The sensitivity of this method was 10 to 100,000 times higher, and the counting time was 300 to 100,000 times shorter than the conventional

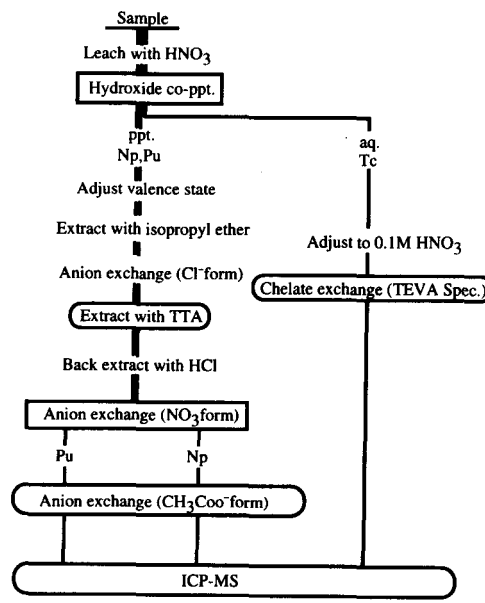

Fig.2 Systematic analysis method for Tc-99, Np-237 and Pu-239,240

radioanalytical methods.

3.2 Determination of environmental samples

3.2.1 Concentration levels of surface soil

The surface soil samples containing abundant amounts of organic materials were collected at nine points. The results for the surface layer soil samples are presented in Table 2. As the table indicates, the concentration levels of Tc-99 in Japan are found to be about 10^{-1} Bq/kg,dry.

3.2.2 Depth profile

Core soil samples were collected by an electric core sampler at one point. The core soil samples were divided into 10 layers of 5cm thickness. The depth profiles of Tc-99 and some long-lived radionuclides are presented in Fig.3. The concentration profiles show that more than 95% of each radionuclide is retained in the surface layer, up to 10cm in depth, which contains more organic materials. The results suggest that content of organic materials in soil is related to adsorption of radionuclides onto soil.

The retainment of Tc-99 onto surface layer appears to be contrary to the hypothesis that Tc-99 would have high mobility in soil, because Tc-99 forms an anion, i.e., TcO_4^- in oxidizable environment.

Table 2 Concentration of Tc-99 in surface soils

Sample	Concentrarion of Tc-99 (Bq/kg,dry)
Mito	$(1.8\pm0.06)\times10^{-1}$
Hitachinaka	$(1.0\pm0.03)\times10^{-1}$
Tokai	$(1.3\pm0.03)\times10^{-1}$
Rokkasyo	$(1.3\pm0.04)\times10^{-1}$
Rokkasyo	$(1.0\pm0.04)\times10^{-1}$
Rokkasyo	$(8.1\pm0.02)\times10^{-2}$
Okuetsu	$(5.8\pm0.09)\times10^{-1}$
Okuetsu	$(4.2\pm0.05)\times10^{-1}$
Fukui	$(1.9\pm0.09)\times10^{-1}$

Mito, Hitachinaka, Tokai; Middle of Japan
Rokkasyo; North of Japan
Okuetsu, Fukui; West of Japan

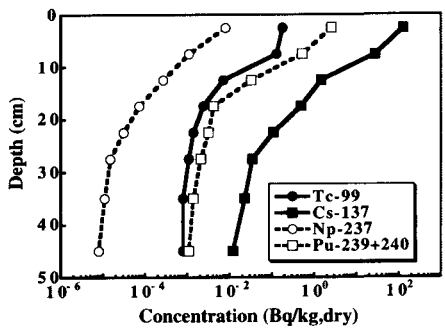

Fig.3 Depth profiles of Tc-99, Cs-137, Np-237 and Pu-239,240 in soil core

4. Conclusion

1. The development of the high-sensitivity determination method for Tc-99 and long-lived radionuclides using ICP-MS successfully improved detection limits as compared with conventional radioanalytical method, and the behavior Tc-99 which has not been elucidated so far to be analyzed from the field data.

2. The results of measurement of the environmental samples showed that the organic materials in the soil affected the adsorption of Tc-99.

3. Since Tc-99 is in an anionic chemical form under an oxidation atmosphere and transfers with water, its vertical transfer rate was considered to be fast, but the results obtained in the field were against this theory. This may stem from the formation of complexes with organic materials contained in the surface soil, reduction by organic acids and selective intake by microorganisms.

RADIOLOGICAL MONITORING IN SLOVENIA

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Abstract: In the light of recommendations provided by ICRP, IAEA and other international bodies, an up to date monitoring programme should provide relevant data for a realistic quantative assessment of radiation doses likely to be received by members of the public. Radiological monitoring programmes in Slovenia are explained and discussed.

INTRODUCTION

The form of a monitoring programme is determined by a number of factors: the type and amount of radionuclides likely to be found in the environment, the principle pathways of exposure, the way by which the environment is used by man and the characteristics of the population, its distribution, social activities, means of earning its livelihood, dietary habits and the origin of the food it consumes. Slovenian activities in this field are just one example of how these general issues are being resolved.

MONITORING PROGRAMMES IN SLOVENIA

The overall monitoring programme in Slovenia consists of three major parts:

i. a general monitoring programme of the radioactivity in the country,
ii. three specific monitoring programmes in the surroundings of the Krško NPP, research reactor and Mine at Žirovski vrh,
iii. emergency monitoring programmes.

Basic programme structure is shown in Fig. 1.

The aim of National (general) Monitoring Programme is to control the external radiation and the specific concentrations of radionuclides in the environment and foodstuffs (Fig.2). It defines sampling locations, sampling and measurement frequency and list of interested radionuclides. Yearly assessment consists of dose assessment to the population, environmental contamination assessment and some projections. Unusual results in regular monitoring trigger supplementary monitoring programme which is in fact just an extended regular monitoring programme with an objective to clarify unusual results.

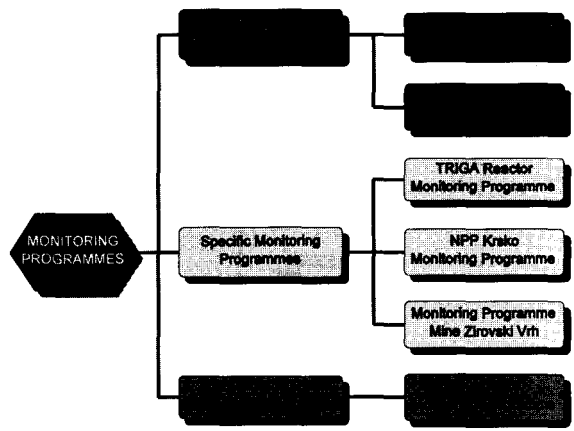

Figure 1. Monitoring programmes in Slovenia

Specific monitoring programmes are performed in the surroundings of nuclear facilities. The most comprehensive one is the monitoring programme around the Krško NPP (Fig. 3). It consists of three sub-programmes:

Programme A -

A-1 regular monitoring programme; it defines locations, methods and frequencies of monitoring of radioactive contamination of air, soil, rivers, precipitation and fallout, drinking water, human and animal feed;

A-2 extended monitoring programme; it defines increased sampling rates and additional sampling locations; it is triggered by unusual results of programme A-1 to clarify the

problem;

Programme B - the programme of effluent and intercomparison measurements;

Programme C - the emergency preparedness programme of the Ecological Laboratory with a Mobile Unit (ELMU).

While *Programme A* represents regular monitoring programme *Programme B* defines sampling locations, representative samples and frequencies of sampling for control measurements of liquid and

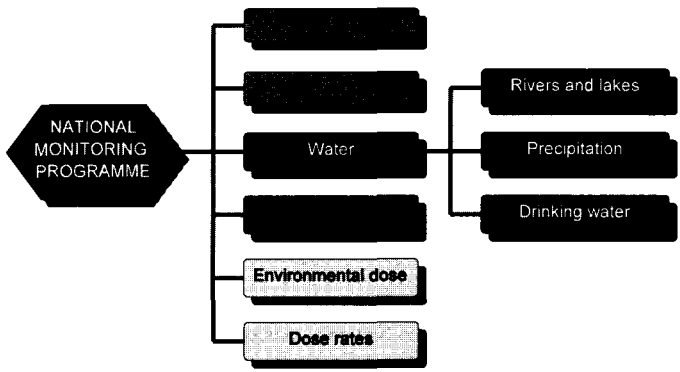

Figure 2. National monitoring programme

gaseous effluents and obvious intercomparison measurements. The aims of *Programme C* are intercomparison measurements between ELMU and Krško NPP, overall tests of emergency capabilities of ELMU, identification of any actual or potential weaknesses in equipment, procedures or team skills, team acquaintance with the surroundings of NPP, routes, measuring points and normal radiological parameters of the environment and finally improvements of the response plans, procedures and co-ordination. *Programme C* is for an adequate and efficient response in radiological emergencies of exceptional importance.

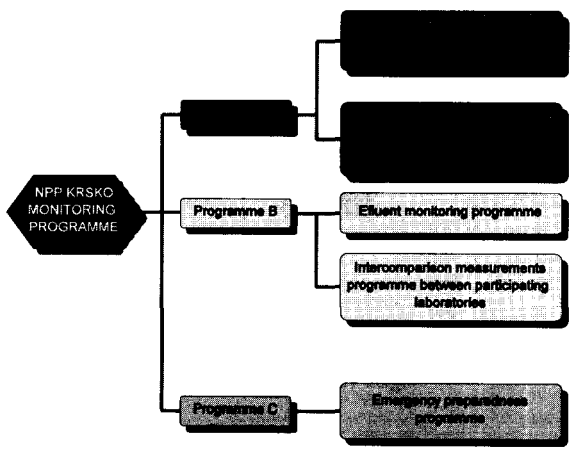

Figure 3. Structure of NPP Krško monitoring programme

Emergency monitoring (Fig. 4 and 5) consists of an early warning system and monitoring in EPZ. *Programme A* offers a sound basis for emergency monitoring. It supports a network of fixed measuring

points, a three circle network of TL dosimeters and a network of automatic meteorological stations with built-in gamma dose rate meters. The three basic networks are further supported by a network of accidental TL dosimeters, a network of fallout sample collection (vaseline plates) and a network of potential, predetermined measuring points.

Continuous monitors (MFM-202 of AMES Co.) with an automatically adjustable time constant supply data with a statistical uncertainty of $2\sigma = 5.4\%$. Thirteen such monitors around the NPP, which are on-line connected over MOBITEL to a central computer, together with 12 others installed at different meteorological stations distributed over Slovenia (so far operating autonomously) form the basis of an *early warning system* now under construction.

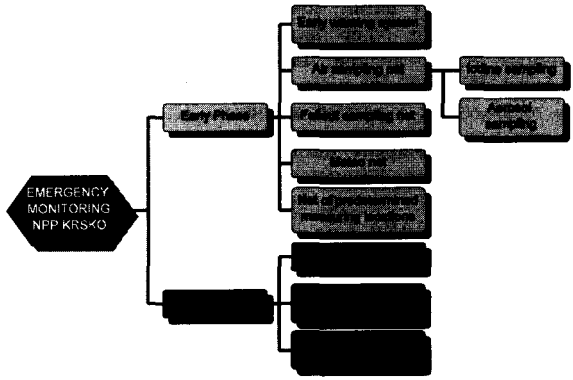

Figure 4. Off-site emergency monitoring programme structure

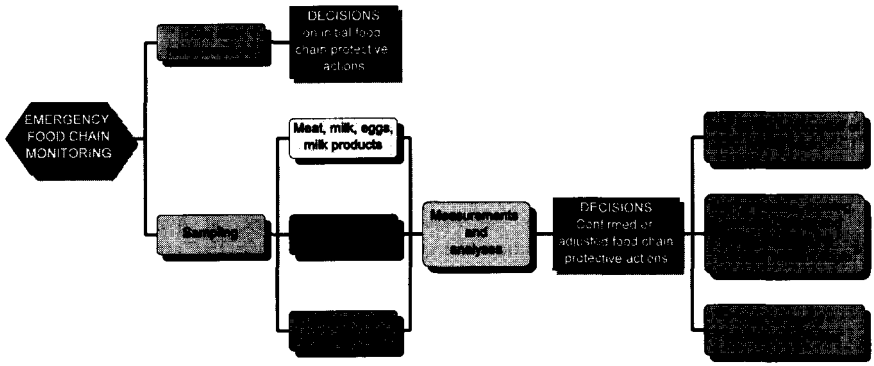

Figure 5. Emergency food chain monitoring

CONCLUSION

All monitored and quantitatively evaluated radiation burdens in the environment in the past 15 years have been in compliance with the licensed values, and well below the authorized limits specified by the competent authorities.

According to these data, the radiological impact on the environment of the Krško NPP for the last fourteen years of operation could be considered of minor significance and of a comparable order of magnitude to other radiological impacts in the Krško - Brežice region, caused by non-nuclear activities (paper - mill industry, hospitals, etc).

LONG-TIME ATMOSPHERIC ^{3}H AND ^{14}C RECORD IN CROATIA

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INTRODUCTION

The natural production of radioisotopes ^{14}C and ^{3}H in the atmosphere is in an equilibrium with their radioactive decay and their world-wide distribution has attained a steady state with subtle variations depending on the latitude, season and solar modulation. This steady state, however, was disturbed significantly during sixties, when the natural activities of ^{14}C and ^{3}H increased due to extensive nuclear bomb tests. ^{14}C and ^{3}H distribution in the atmosphere started to be monitored about at the same time, therefore, the peaks due to the anthropogenically produced ^{14}C and ^{3}H were recorded. Since then, atmospheric ^{14}C and ^{3}H activities are regularly measured as a part of a network (1, 2, 3) or local monitoring (4, 5).

In this paper we present the complete record of ^{14}C and ^{3}H data measured in Zagreb. We present an attempt of analytical representation of our data, and we compare our data with recently published ^{14}C records in Europe (1, 4) and in Canada (5), as well as with the tritium data in precipitation published by IAEA (3).

SAMPLING AND MEASUREMENTS

Measurements of atmospheric ^{3}H and ^{14}C activity have been performed in our laboratory since 1976 and 1978, respectively. Since then, tritium in monthly precipitation samples at Zagreb has been regularly measured, as well as precipitation at Ljubljana (Republic of Slovenia), and these results have been published in IAEA-WMO Technical Reports series (5). For shorter periods of time, tritium in monthly precipitation was measured at Plitvice National Park, SE from Zagreb, and in Rijeka, at the North Adriatic coast. Recently, we started to collect precipitation samples also on Mt. Medvednica, station Puntijarka, at ~1000 m a.s.l., nearby the city of Zagreb. We have been measuring also tritium activity in atmospheric water vapor since 1988 at the Institute and at several sampling sites in down-town Zagreb or in its surroundings, with the aim of studying possible local tritium contamination.

^{14}C in biosphere has been monitored through atmospheric CO_2, annual plants and leaves. Samples were collected at the sparsely populated area of Plitvice National Park and Mt. Medvednica, at the density populated area of Zagreb and in wider area of nuclear power plant Krško, which is located in Slovenia, ~20 km from the Croatian border. Measurements of ^{14}C activity of tree rings grown in the Plitvice National Park, at Mt. Matra (Hungary) and near Krško were used for reconstruction the past radiocarbon in the atmosphere.

Atmospheric water vapour were collected by absorption on silicagel. Samples of CO_2 were collected monthly by absorption on saturated NaOH solution. For ^{14}C measurement in tree rings cellulose samples of individual tree-rings were separated. All the samples for ^{3}H and ^{14}C measurements were converted to gas methane which is used as a counting gas in proportional counters.

RESULTS AND DISCUSSION

^{3}H in the atmosphere

^{3}H concentration in precipitation at three continental stations, Zagreb (1976-1995), Ljubljana (1981-1995) and Plitvice (1980-1982) closely follows each other showing seasonal changes typical for the temperature zone of the northern hemisphere with a decreasing yearly mean values. The yearly mean values in Zagreb and Ljubljana are closely correlated with those at Vienna station (3), as the closest continental station within the IAEA-WMO network. By using this

correlation we were able to reconstruct the past tritium activity in Zagreb (6). The ^3H measurements at Puntijarka station started in 1995. This station is assumed to be less affected by local contamination effects, such as fossil-fuel combustion or tritium release, than the nearby city of Zagreb. The first results show no significant difference in ^3H concentration between Zagreb and Puntijarka station. However, ^3H concentration at the coastal station Rijeka is, on the average, half of that at Zagreb, and is closely correlated to the nearest Mediterranean station Genoa.

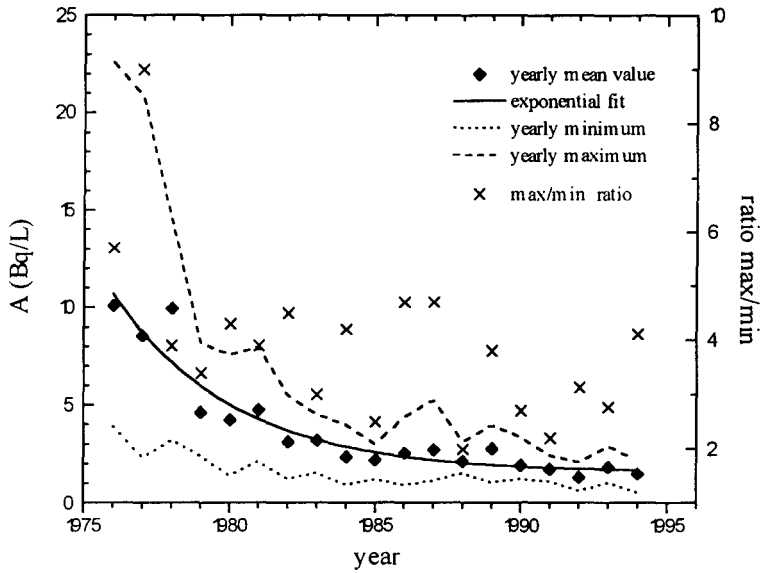

Figure 1. Statistically elaborated ^3H data in Zagreb precipitation.

The twenty year record of ^3H data in Zagreb precipitation is statistically elaborated in Fig. 1. Yearly mean values of Zagreb precipitation decrease from 10.1 Bq/L in 1976 to 0.5 Bq/L in 1994 and the decrease can be well described by the exponential decay curve (full line, Fig. 1). We show the maximal and the minimal activities in monthly precipitation during the year. Both values decrease following again approximately exponential decay. The lowest value, measured mainly in the winter time, in the last three years is 0.5 Bq/L, very close to the assumed natural pre-bomb tritium activities. The maximal value, measured mainly in summer period, are between 2 and 5 Bq/L. Similar activities were measured by Meijer *et al.* (4). We also present the behavior of the ratio of the maximal to the minimal yearly values. The values of max/min ratio fluctuate between 2 and 5 with an exemption in 1977 with a value of 9.

^3H concentration in atmospheric water vapour at Ruđer Bošković Institute was several times higher than that in precipitation from the same period. At all other investigated places the ^3H concentration in water vapour was very close to that in precipitation, so no local contamination similar to that at the Institute was observed. ^3H concentration in water vapour is much more locally affected than that in precipitation and it is a good indicator for local contamination.

^{14}C in the atmosphere

The summary of all ^{14}C environmental and atmospheric results is shown in Fig. 2. Values of atmospheric CO_2 are mean yearly values, those of plants are either single or average values of several plants, and "tree" stands for tree-rings. Curves fitted to the data of Plitvice tree,

period 1964-1986 (full line), as well as Zagreb atmospheric CO_2, period 1983-1995 (dotted line), are shown in Fig. 2. Data in Fig. 2 can be discussed in several time periods: 1) up to 1964, increase of $^{14}CO_2$ activity, 2) 1964-1969, period of fast decay of $^{14}CO_2$, app. 50-65‰ per year, 3) 1970-1979, lower decay rate, app. 20-30‰ per year, and 4) 1980-present, decay rate app. 10‰ per year. Assuming that no new sources will introduce new ^{14}C into the atmosphere and that the rate of fossil fuel combustion will remain constant, the extrapolation of the ^{14}C decay curve to the year 2000 gives $\Delta^{14}C$ value of 116‰ (fitted line, Plitvice tree) and 62‰ (fitted line, Zagreb CO_2), respectively. It means that atmospheric ^{14}C activity will approach 11% and 6%, respectively, above prebomb levels by the turn of the century. For comparison we used atmospheric $^{14}CO_2$ data from Germany (1), from the Netherlands (3) and from Canada (4). All data show the exponential decay of mean yearly ^{14}C activities, with half life of app. 8 years. Data from Canada (Fig. 2) agree very well with ^{14}C activities measured in Plitvice tree, resulting in similar extrapolated ^{14}C activity for 2000. Lower $^{14}CO_2$ activities in Zagreb indicate greater influence of fossil fuel CO_2.

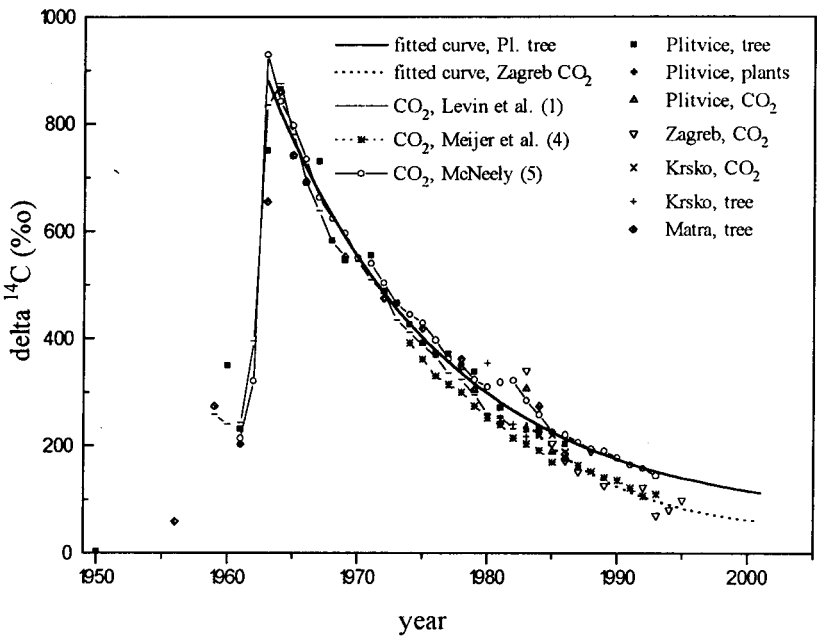

Figure 2. The ^{14}C activity of atmospheric CO_2 and tree rings. Fitted curve for Zagreb CO_2 and Plitvice tree extrapolated to 2000 and compared with the ^{14}C measurements in Germany (1), the Netherlands (4) and Canada (5).

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
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Abstract No.
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PAPER TITLE Determination of ^{90}Sr in natural samples

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MAJOR SCIENTIFIC TOPIC NUMBER 4 (see page 7)

ABSTRACT (See instructions overleaf)

Strontium isotope ^{90}Sr belongs to the group of isotopes which are very dangerous for human health. Therefore ^{90}Sr is regularly determined in natural samples such as soil, drinking water and food. When determining ^{90}Sr in natural samples it is necessary to isolate ^{90}Sr from the sample because ^{90}Sr is a pure β-emitter. As natural samples contain much more sodium and calcium than strontium, it is necessary to isolate a small quantity of strontium from a large sample with simultaneous separation of calcium and other interfering elements.For this purpose an elegant method based on ion exchange chromatography has been developed. Strontium is isolated from the natural sample and simultaneously separated from calcium, sodium, potassium, caesium, iron and some other elements on the chromatographic column filled with strong anion exchanger (type DOWEX and AMBERLITE) and alcoholic acid medium as eluent. Therefore, the paper will show the separation of strontium from caesium, potassium, sodium, calcium, barium and iron on two types of exchangers with different alcoholic acid eluents (ethanol + HNO_3, methanol + HNO_3). The application of the separation on fast ^{90}Sr isolation from a liquid sample will be presented and the possibility of fast and accurate determination of ^{90}Sr after isolation and separation from ^{90}Y on ion exchangers will be particularly considered.

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FORM FOR SUBMISSION OF ABSTRACTS
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Abstract No.

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Author

Acceptance

Mini-Presentation

PAPER TITLE On the radioactivity of the main components of the Romanian diets

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MAJOR SCIENTIFIC TOPIC NUMBER 4.1 (see page 7)

ABSTRACT (See instructions overleaf) In the paper there are presented the results of the radioactive measurements on some main food items like: meat and meat products milk and dairy products; fresh vegetables and fruit, wheat and wheat flour.

The foodstuffs are sampled from six areas:Bechet(plain, near Kozloduj NPP, Bulgaria) Bucharest (plain, the most populated urban area); Cernavodă (plain, future NPP); Neamtz (mountain); Piteşti (hill, nuclear facility); and Sibiu (hill). Those areas are representative both from geographical and dietary point of view.

The measurements have been performed by usual methods of low-level high-resolution γ- ray spectrometry and radiochemical analysis.

The main considered radionuclides were: Cs - 134; Cs - 137, I - 131, Sr - 90, from the contaminating ones and K - 40 and Ra - 226 from the natural ones in the midlle of the 1986 there were identified other radionuclides too.

From these measurements there is also presented the evolution of the radioactive ammounts in the specified foodstuffs for the last decade.

On these basis there is assessed the contribution of the food consumption to the equivalent dose (ingestion dose) for the mean characteristic Romanian diets.

INVESTIGATION OF THE CONTENTS OF NATURAL RADIONUCLIDES IN COAL AND ASHES FROM KOSOVIAN POWER PLANTS

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ABSTRACT

The basic aim of this work was investigation of the contents of natural radionuclides in the samples of coal and in the products of its burning in Kosovian power plants, as the beginning steps to estimate total radioactive effects on the surrordning, having in mind that coal power plants are one of the more influental source of the redistribution of natural radionuclides.

The investigation has been done by gamma spectroscopy analysis using HP Ge detector, with relative efficiency of 20% and resolution of 1,8 keV in energy of 1332 keV ^{60}Co. The activity concentation of natural radionuclides in measuered samples of coal, ashes and slag were from 3,1 to 67 Bq·kg^{-1} for ^{226}Ra; from 1,7 to 32 Bq·kg^{-1} for ^{232}Th; from 0,3 to 6,5 Bq·kg^{-1} for ^{235}U; from 6,5 to 138 Bq·kg^{-1} for ^{238}U and from 14 to 234 Bq·kg^{-1} for ^{40}K.

INTRODUCTION

In the frame of the Kosovian basen there are more than as much as 10 billion tons of coal. In order to use this natural resources have been built some of the Industrial plants such as are : Thermal power plants Kosovo A and Kosovo B with the total instaled capacity of 1470 MW; dry plant with the capacity of 1.220.000 tons, gas plant with the capacity of 48 x 10^7 m^3, fertilizer factory with the capacity of 360 000 tons per year of KAN-a and other plants in the connection with the coal production.

Kosovian coal belongs to the group of the typical lignite. It is known for its high content of ashes (over 15%) and low thermal capacity (from 5 400 kJ·kg^{-1} to 10 500 kJ·kg^{-1}). The largest amount of this coal are used as unrefined energetic solid fuel in Kosovian power plants. The firebox is in the process is deposition of the slag in the firebox what is the cause of the increasing of ashes and the emission floated ashes in atmosphere.

Power plant Kosovo A belong to the old type plant. It is consist of 5 blocks which by its chomneys emits in atmosphere about 15% of producing ashes, what amounts for 5 000 h of yearly working of the plants about 1200 000 t per year. The second power plant is Kosovo B it consists of two energetic blocks and belongs to the contemporary plants. Its emission of ashes is about 1 000 000 t per year.

The air of this work is the monitoring of the content of natural radionuclides in coal and the products of the combustion in the process of the work of Kosovian plants. The influence of chemical products polution like as ashes, dust and gases from the smoke of stacks are is much more than radionuclides do it. To the risk of the radionuclides, which release in the process of the work of the power plants intented for producing of electrical energy, unsatisfactory pays attention and the negative contribution of this sourse becomes significant for the environment. So as it is well known according to estimates that power plant of 1 GW yearly release the collective effective doze of 20 man Sv (1).

MATERIALS AND METHODS

The analysed samples of coal have been collected from the surface mines of Kosovian basen Belaćevac and Dobro Selo. The first patterns were by means of the furrows which are placed in the coal drawings of the mine. But of that manner of collecting the samples gave up because of some reasons: The floors were as high as about 10 m , steeps and incomprehensible and therefore very dangerous for the gatherers of the samples. In addition to these facts, it was umpossible to reach some floors because of enormous maisture of the soil and coal appeared by the underground waters. Taking into consideration that facts the samples of the coal were done by the method of the raudomly samples. The samples were taken from the given quantities of coal on the conveyer belt and the freight cars for definite timing intervals. The samples size of coal were of the piece, cube, nut, hazelnut and coal dust. The samples of coal dust have been taken on coal depos on given places and different depths. The samples of deposited ashes have been taken from all phases of the electriphylters and in the some way on depos and lagoons . The samples of the slags have been taken below of the bunker for the slag

(on that place the slag is in the heaty state) and from the places where are in teh coal state. Some of these samples have been taken from lagoons where ashes and the slag are in the mixture state.

All samples after complex guartering and dreying at 105 % grinded to the granulation of the fine powder and brig down to the some volume. After done of the homogenous of the samples the samples were measuered and filled in the standard Marinelli plates. The Marinelli plates are then welded by beeswax because of the airtight of the samples. Prepared is such way, they are left to repose 30 days (before of the begninung of the measuring) in order to esstablish the radioactive equilibrium. The measuring gamma activity of the radioactive meterials in the samples has done by multy shanel gamma analyzer ORTEC with HP Ge detector with the relative efficiency of 20% and resolution (FWHM) of 1,8 keV at 1332 keV ^{60}Co. The callibration of energy has been done by the standard series of points sources (Coffret d´etalon ECGS-2, Sacle) which contain the radionuclide ^{133}Ba, ^{60}Co, ^{137}Cs, ^{22}Na and ^{241}Am (activity of 10^3 Bq at day 29.11.1977.). The geometrical efficiency was given by the referent material-soil, contaminated by series of radionuclides ^{22}Na, ^{57}Co, ^{60}Co, ^{88}Y, ^{133}Ba and ^{137}Cs with the activities between 122-355 Bq/kg (total 1461 Bq/kg at the day 1.7.1991.) with the total uncertainity of 5% (National Office of Measures OMH, Budapest) (2).

The activity of ^{226}Ra and ^{232}Th is determined, using their descendants ^{214}Bi ,^{214}Pb and ^{228}Ac , ^{208}Tl , respectively. The activity of ^{235}U is determined on energy of 185,83 keV with correction on ^{226}Ra (186keV). The activity ^{238}U we computed using natural relate of the activity of ^{235}U/^{238}U which is 0,04604. The activity of ^{40}K we computed using its only tranzition an 1460,8 keV. The correction taking into consideration the autoapsorption was not done, because there were not significant difference among obtained concentration for ^{226}Ra and ^{232}Th in the selected spectrum lines (3). The time of collecting the pulses i.e. recording time of the spectrum, was from 10^5 to $3x10^5$ s depending of the activity of the samples.

RESULTS AND DISCUSSION

The results of the experimental measuring of specific activity of ^{226}Ra, ^{232}Th, ^{235}U, ^{238}U and ^{40}K in samples of coal are shown in Table 1. It descibes intervals of the concentration of the activity for monitored radionuclides for each kind of the coal samples for diferent radiuses.

In Table 1. has been seen that analysed samples of coal contents almost the same concentracion of the activity of monitored samples with the except the samples of the coal dust. The reason for enlarget concentration of the activity in the coal is affter all in the prensence of the floating ashes emited by the chimney of the power plants and it deposits in the depos. On the base of obtained results may be appraissed that the concentration of activity of the natural radionuclides in the monitoring samples of Kosovian lignite are relatively lower in comparison with the concentration of activity in other countryes in the world (4)

Tabele 1. The range specific activity of the natural radionuclides (Bq kg^{-1}) in the samples Kosovian lignite

Kind of the samples	^{226}Ra	^{232}Th	^{235}U	^{238}U	^{40}K
coal piece	3,1-4,9	1,7-4,2	0,29-0,42	6,5-9,1	14-87
coal cube	4,5-7,7	3,5-4,3	0,43-0,52	9,4-11	19-20
coal nut	3,5-5,8	4,1-4,9	0,56-0,71	12-15	14-25
coal hazelnut	4,2-14	4,8-8,6	0,47-1,1	10-23	25-32
coal dust	7,3-25	8,0-18	0,61-4,5	13-45	49-97

The experimental results of the measuring of the specific activity of ^{226}Ra, ^{232}Th, ^{235}U, ^{238}U and ^{40}K in samples of ashes and slag as the products of the combustion in Kosovian plant are shown in Table 2.

In analysed samples of ashes the biggest concentration of the majorits of the tested radio-nuclides is presented in the samples of ashes of the bottom of the chimney funnel. Because of that fact we my suppose that the content of U and Ra enlarges with the lowering the diameter of the particles (5). In Table 2. can be seen that the concentration of activity ^{40}K is the biggest in the samples of the slag. The reason of that facts we should to try in the physical-chemical features of potassium as and that facts that the slag is waste materials which consists of enough uncompletely burned organic material in coal.

Table 2. The range specific activity the natural radionuclides (Bq kg^{-1}) in the samples of ashes and slag because of burning of Kosovian lignite

Kind of the samples	^{226}Ra	^{232}Th	^{235}U	^{238}U	^{40}K
ashes from the bunker	12-20	14-20	1,9-2,2	43-48	150-199
ashes from the palettyser	15-67	16-31	2,1-4,5	45-138	179-219
ashes from the electrofilter	14-23	16-18	2,0-3,5	46-65	108-179
ashes from the chimney funnel	23-48	22-32	3,6-6,5	89-146	128-236
ashe from the substation	22-65	20-30	2,5-5,5	56-109	167-229
ashes from depos	29-44	17-25	3,0-4,8	65-104	170-210
ashes and slag from the lagun	10-18	24-30	3,0-5,0	91-109	211-239
slag from from the bunker	12-14	13-15	1,7-1,9	37-42	163-198
slag from the substation	14-20	18-23	2,1-2,9	63-77	184-243
slag from the depos	24-28	24-28	3,5-4,6	77-87	177-183

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CONTENTS OF RADIONUCLIDES IN SURFACE LAYER OF UNCULTIVATED SOILS OF A MOUNTAIOUS REGION

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ABSTRACT

The results of natural (K-40, Ra-226, Th-232, U-235) and fallout radionuclides (Cs-134, Cs-137) activity estimation in mat (surface layers of uncultivated soil) of a mountain region are presented. The samples were collected on locations with limestone, shale or mixed type soil. The activity of the radionuclides was determined on a HPGe detector by standard gamma spectrometry.

Mats of shale soils have higher content of Ra-226 and Th-232 than those of limestone, while for K-40 and U-235 there are no significant differences due to the type of soil. Compared to the samples taken from the depth of 5-15cm, the content of Ra-226, Th-232 and U-235 is lower in the surface layers; for K-40 there are exceptions, mainly on limestone soils. The content of Cs-137 and Cs-134 in mat is much higher than in deeper layers of soil, indicating slow migration, except on the riverside.

INTRODUCTION

The long-term radioactivity investigations in a mountainous part of central Serbia (rich in honey-bee pastures and meadows) indicated two radioecological areas: one with the lower gamma field intensities, mainly on limestone and the other with higher exposure rates, on shale soils (1,3). The exposures were supposed to be mainly due to natural radionuclides in soil (4), but Chernobyl radionuclides (Cs-134, Cs-137) could not be neglected (3,5). These investigations are a part of a project dealing with determination of natural radionuclides and Cs-134 and Cs-137 in meadow flora, soil and honey in the region.

MATERIALS AND METHOD

The samples of mat (surface layers of soil with roots of meadow flowers, 25x25cm) were randomly collected on eight locations in a mountain region (300-1000m above sea level), in summer 1991. The soils differ in physical and chemical

characteristics: limestone is predominant (locations II, IV, VI, VIII, partialy III), but shale soils can be found, too (locations I, V, VIII). The samples were dried on the room temperature, grinded, weighted in standard Marinelli beakers (0.5 l) that were sealed by a honey-wax film and left for 4 weeks to reach the radioactive equilibrium.

The activity of the radionuclides was determined on a HPGe detector(relative efficiency 20%) by standard gamma spectrometry. Geometric efficiency was determined by a reference soil standard (National Office of Measures,Budapest) spiked with a series of radionuclides (Na-22, Co-57,60, Y-88, Ba-133, Cs-137; activities 122-135 Bg/kg on 1.7.91, overall uncertainty 5%).

Counting intervals ranged from 150.000 - 300.000s and the background radiation was checked regulary after or before the sample. Total standard error of the method was 15%.

RESULTS AND DISCUSSION

The activities of natural and fallout radionuclides (Bq/kg dry weight) ± total error of the method are presented in Tables 1 and 2, respectively. On locations where two samples of mat had been taken, both results were presented. LLD denotes lower limit of detection.

Table 1. Activity (Bq/kg) of natural radionuclides in mat

Location	Ra-226	Th-232	U-235	K-40
I1	27 ± 4	36 ± 5	3.0 ± 0.4	378 ± 52
I2	26 ± 4	38 ± 6	2.6 ± 0.4	396 ± 59
II	21 ± 3	35 ± 5	3.1 ± 0.5	528 ± 79
III	38 ± 8	50 ± 8	3.6 ± 0.5	446 ± 67
IV	17 ± 3	22 ± 3	5.1 ± 0.4	307 ± 46
V	25 ± 4	27 ± 4	2.7 ± 0.3	436 ± 65
VI	6 ± 1	5 ± 1	LLD	219 ± 33
VII	50 ± 8	39 ± 6	4.3 ± 0.6	554 ± 83
VIII1	44 ± 7	54 ± 9	4.5 ± 0.7	827 ± 124
VIII2	38 ± 8	49 ± 8	5.2 ± 0.4	731 ± 109

Generally, mats on shaleous soils have higher contents of Ra-226 and Th-232 than on the limestone, while for K-40 and U-235 there are no significant differences due to the type of soil. Compared with the content of natural radionuclides in 5-15cm layers on the same locations (4), the content of Ra-226 and Th-232 is higher in mats, for U-235 no reliable conclusion can be made and for K-40, its content on limestone soil is lower in mat or there are no differences.

Table 2. Activity of Cs-134 and Cs-137 (Bq/kg) in mat

Location	Cs-137	Cs-134
I1	253 ± 38	23 ± 4
I2	161 ± 24	15 ± 2
II	126 ± 19	10 ± 1
III	450 ± 68	38 ± 6
IV	967 ± 145	91 ± 14
V	860 ± 129	76 ± 12
VI	1225 ± 184	118 ± 18
VII	942 ± 138	83 ± 13
VIII1	813 ± 122	69 ± 10
VIII2	1600 ± 240	158 ± 24

The content of Cs-134 and Cs-137 in mat is significantly higher than in deeper layers (5-15 cm), except on a location (I), with typical shale soil where the nearby river caused high washout. The vertical distribution of both cesium isotopes (their ratio confirming their Chernobyl origin) in soil strongly depends of soil configuration and physico-chemical properties (6).

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Poster N : P41-66

THE RADIATION MONITORING ON THE TERRITORY OF BELARUS.

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Introduction.

On the territory of the Republic of Belarus works the points network of radiation-ecological monitoring of environmental objects by the special programs.

Accoding to the article 40 of the Low of the Republic of Belarus "About law regime of the territory exposed to a radioactive pollution following the accident on the Chernobyl NPP " the Committee for Hydrometeorology of the MES of RB accomplishes a total estimation of radiation situation on the territory of the Republic [radiation monitoring] and a methodical quidance [see figure 1].

General part.

Air, surface water and earth monitoring within the reference observation network.

From 1963 in the Republic has been realized observations of ambient air radioactive contamination. In the presence on 54 stations every day are measured a power of the exposure gamma-radiation dose rates. The measurements in the 100 kilometres zone of NPP influence are realized every 3 hours. On 24 stations placed on all the territory of the Republic of Belarus is controled the level of radioactive precipitation from the land atmospheric stratum [horizontal plane-tables]. In 6 cities of the Republic [Minsk, Mogilev, Gomel, Brest, Mozir, Pinsk] is daily measured the content of radioactive aerosoles in the air using filter-ventilation installation [see figure .2].

The laboratories of the Committee for Hydrometeorology realize constantly the measurements of gamma-radionuclides, strontium -90 and plutonium in precipitation and aerosole samples.

Systematical control of the radioactive surface water and ground accumulations contamination is carried out on 5 main rivers of Belarus such as Dnepr [range in Retchitsa], Soge [range in Gomel], Iput [range in Dobrush], Besed [range in Svetilovichi] and Pripyat [range in Mozir] run on the contaminated territory.

From 1986 have been accomplished the radiation inspection of all the territory of Belarus including settlements, agricultural and forests areas with the participation of the Academy of Sciences, the Ministry of Agriculture, the Ministry of Forestry and other ministries and departments.

From 1992 the Committee for Hydrometeorology has been accomplished the radio-oecological earth monitoring within the reference network included 18 landscape and geochemical grounds and 181 reference sites.

CONCLUSIONS.

The observations accomplished in the period of 1990-94 showed an availability of the short-time [seasonal] risings of atmospheric air radioactiveness. Futhermore were detected the air radioactiveness rising at the expence of the dust lifting in the time of field works, it was possible the radioactiveness rising during the forest fires. But it should be noted that these risings are short-time. By the data of the Committee the short-time radioactiveness rising in the air during the forest fires doesn't influence on the whole the content of surface earth radionuclides.

For all the observed period from 1987 the excuding of caesium -137 content on the surface water of the main rivers of Belarus has not been observed. The caesium -137 taking out by surface water of the territory of Belarus has been considerably decreased in due course.

The observations showed that the intensity of radionuclides migration on vertical line was losely connected with the genetical peculiarity of the earth. The analysis of received data showed that.

Analysis of collected data shows, the most intensive migration of radionuclides are concentraded in the upper 5-8 layer.

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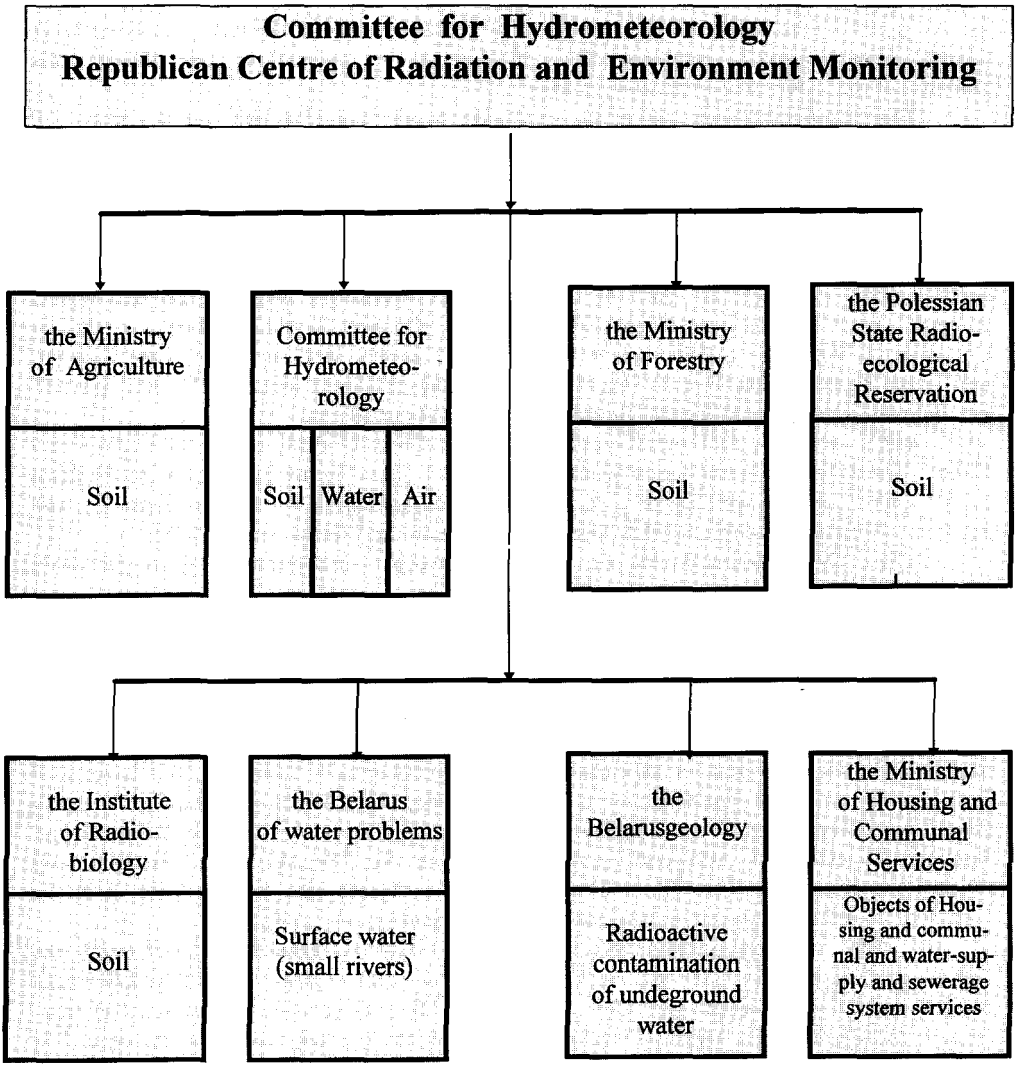
The Radiation Monitoring System

on the territory of Belarus



UP-TO DATE STATE AND DYNAMICS OF RADIOECOLOGICAL SITUATION AT THE EAST-URAL RADIOACTIVE TRACE IN SVERDLOVSK OBLAST

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The 1957 explosion of a tank filled with highly radioactive waste solutions at the "Mayak" nuclear facility caused a radioactive contamination of the northern part of Chelyabinsk Oblast and the southern part of Sverdlovsk Oblast. The contaminated territory was called the East-Ural radioactive trace (EURT).

In 1992-1994, a detailed radioecological study on territory of the EURT in Sverdlovsk Oblast was carried out by combined efforts of a number of organizations under the scientific guidance of the Institute of Industrial Ecology.

The investigated area was about 1600 km^2. More than 3500 soil samples from 370 testing sites were studied. The 5-10 cm layers of soil were cut horizontally to a depth of 40-70 cm. Radiochemical analyses for ^{90}Sr and ^{137}Cs were carried out.

It was discovered that, to this day, there are significant levels of ^{90}Sr contamination in Sverdlovsk Oblast, especially in the Kamensk district. Along the axis of the radioactive trace there is large territory where the average ^{90}Sr contamination is within 1-3 Ci/km^2, with some spots having activities of over 5 Ci/km^2. These territories include the settlements Rybnikovskoye, Bogatenkovo, Shcherbakovo, Kluchi, Kluchiki, Kodynka, Malaya Kodynka, Cheremkovo, Belovodje and Svoboda with the total population of about 3200. In the southern part of the Kamensk district the width of the trace within the limits of 1.0 Ci $^{90}Sr/km^2$ is 12-15 km and within the limits of 0.1 Ci $^{90}Sr/km^2$ the width is about 30 km.

The present ^{90}Sr contamination of the town Kamensk-Uralskiy varies from 0.06 to 6.9 Ci/km^2. The average contamination levels vary from 0.3 Ci/km^2 in the eastern part of the town to 0.7 Ci/km^2 in the western part. The average contamination of the Bogdanovich and Kamyshlov districts is within 0.2-0.4 Ci/km^2, with a maximum value up to 1.2 Ci/km^2.

The ^{137}Cs contamination levels in the investigated part of Sverdlovsk Oblast are within 0.02-3.1 Ci/km^2, with an average value of 0.3 Ci/km^2.

The plutonium content in samples from three soil sections on the EURT axis in the southern part of the Kamensk district corresponds to the level of surface contamination 0.005-0.03 Ci/km^2.

The concentration of ^{90}Sr in the main wood species ranges at present from 1 to 100 Bq/kg air-dry wood for pine trees, from 2 to 230 Bq/kg air-dry wood for birch trees and from 3 to 470 Bq/kg air-dry wood for aspen trees. The contamination of wood by ^{137}Cs is up to 7 Bq/kg air-dry wood for pine trees, up to 35 Bq/kg air-dry wood for birch and up to 50 Bq/kg air-dry wood for aspen. The contamination of mushrooms in the territory of Kamensk and Bogdanovich districts is 6-53 Bq/kg air-dry mushrooms for ^{90}Sr and 29-143 Bq/kg air-dry mushrooms for ^{137}Cs. The ^{90}Sr soil-to-plant transfer factors (Bq kg^{-1}/Bq m^{-2}) in forest ecosystem in the south-eastern part of Sverdlovsk Oblast are shown in Table 1.

In 1962, when the annual monitoring of the lakes of the Kamensk district was began, the ^{90}Sr concentration in the water of Lake Tygish was approximately 200×10^{-12} Ci/L and of Lakes Sungul' and Chervjanoe ~50×10^{-12} Ci/L. The present ^{90}Sr concentration in the water of these lakes is (7-20)$\times10^{-12}$ Ci/L. During the whole period after the accident the concentration of ^{90}Sr in the water of all three lakes decreased by a factor of two over 15 years, i.e. somewhat shorter than the half-life period of ^{90}Sr (Figure 1).

TABLE 1. THE ^{90}Sr SOIL-TO-PLANT TRANSFER FACTORS, 10^{-3} m^2/kg

FOREST PRODUCT	TRANSFER FACTOR
Pine (wood)	1.1
Birch (wood)	2.4
Aspen (wood)	3.9
Mushrooms	1.7-24

TABLE 2. THE ^{90}Sr CONCENTRATION IN FISH IN 1982-1993, 10^{-9} Ci/kg

LAKE	^{90}Sr CONCENTRATION	
	IN FLESH	IN BONES
Tygish	1.0-4.2	6.4-41.7
Sungul'	0.2-2.0	4.5-23.0
Chervjanoe	0.2-1.0	11.6-34.4

For Lake Tygish which is the most contaminated, the mean concentration of ^{90}Sr in the bottom sediments is $(5\text{-}60)\times10^{-9}$ Ci/kg of dry substance.

The annual measurement of ^{90}Sr concentration in fish of the Lakes Tygish, Sungul' and Chervjanoe was carried out from 1968. In 1982-1993, the ^{90}Sr concentration in fish (crucian and carp) from these three lakes varied in the intervals which are shown in Table 2.

From 1968 onwards, the water-to-fish ^{90}Sr transfer factor (Bq kg^{-1}/Bq L^{-1}) is approximately constant and varies over a range of 15-60 for water-to-flesh transfer and 400-2400 for water-to-bone transfer.

The ^{90}Sr concentration in the water of the Kamenka River, which intersects the EURT, in summer 1993 was $(1\text{-}2)\times10^{-12}$ Ci/L. In the water of its small tributaries, which run over the territory where the average ^{90}Sr contamination is within 1-2 Ci/km^2, it was $(1\text{-}15)\times10^{-12}$ Ci/L. The contamination of the bottom sediments of the Kamenka River by ^{90}Sr in 1993 was $(0.6\text{-}2.0)\times10^{-9}$ Ci/kg dry weight, and the contamination of aquatic plants was $(0.1\text{-}1.9)\times10^{-9}$ Ci/kg dry weight.

To predict the concentration of radionuclides in the water of the rivers which run over the contaminated territory, the washout factor of radionuclide from the catchment-areas to the river water (concentration in water, Bq×m^{-3}/average contamination of the catchment area, Bq×m^{-2}) is usually used. For the small tributaries of the Kamenka River the washout factor of ^{90}Sr from the catchment-areas to the river water is 0.01-0.05 m^{-1}.

The ^{90}Sr contamination of foodstuffs produced in the territory of the EURT in the Kamensk district is approximately in 5 times higher than the average contamination in Sverdlovsk Oblast. In 1993, the average concentration of ^{90}Sr in the vegetables from the most contaminated part of the Kamensk district was the following: milk 15.9×10^{-12} Ci/L, potatoes 9.5×10^{-12} Ci/kg, carrot 116×10^{-12} Ci/kg, cabbage 852×10^{-12} Ci/kg.

Because of the high age of the radioactive fallout the radioactive contamination of foodstuffs is quite stable (Figure 2). A major factor conducive to lowering the radioactive contamination of the territory will be the physical decay of ^{90}Sr.

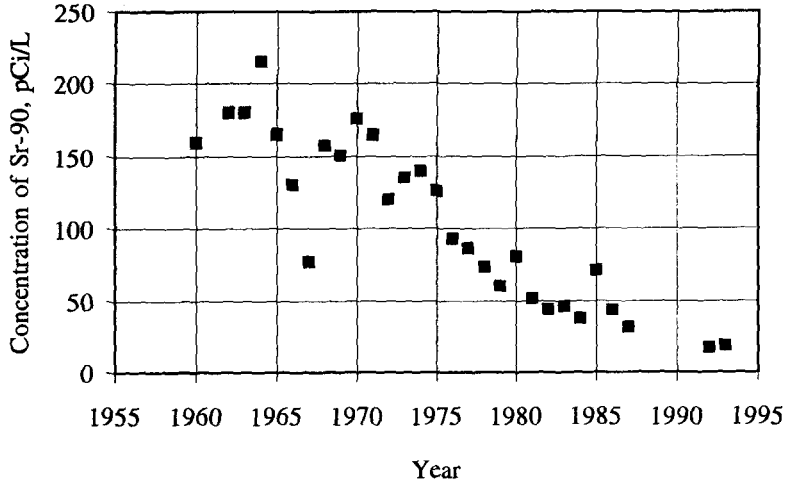

Figure 1. The decrease of the ^{90}Sr concentration in the water of Lake Tygish

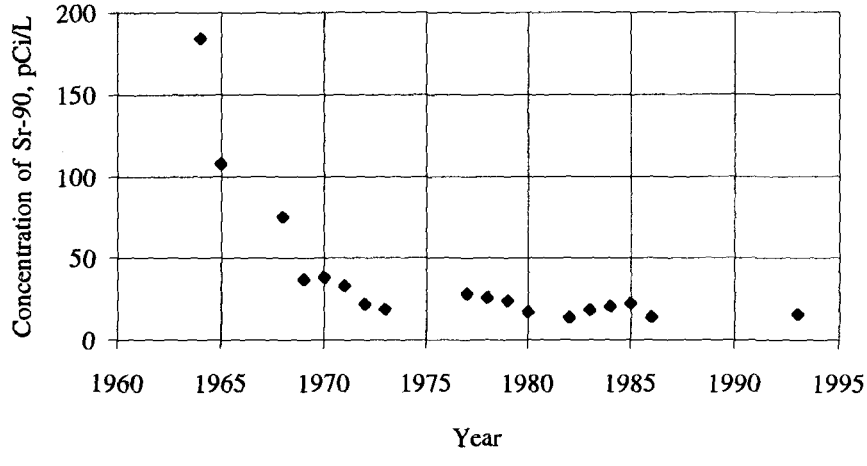

Figure 2. The decrease of the ^{90}Sr concentration in the milk from the EURT territory in Kamensk district

AERIAL MEASUREMENTS WITH HELICOPTERS TO DETERMINE ACCIDENTAL RELEASES FROM NUCLEAR FACILITIES

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ABSTRACT

Aerial gamma ray spectrometry is a useful tool for providing a rapid and comprehensive assessment of the dispersion and deposition of radionuclides over wide areas in post accident situations in nuclear facilities. This paper describes an aerial measuring system for nuclide specific measurements of soil contaminations and for detecting sources of gamma radioactivity. The measuring device consists of several components that can be separately installed in an Alouette II type helicopter. The central unit is a computerized gamma ray spectrometer with a high purity Ge-semiconductor detector and a 12 l volume NaI(Tl) scintillation detector array. At a flight altitude of 100 m, soil contaminations of some kBq/m^2 of ^{137}Cs can be measured (measuring time: 60 s, flight speed: 100 km/h). This aerial measuring system was for the first time operatively applied for measuring the deposition of artificial radionuclides in the southern part of Germany after the Chernobyl accident. Nuclide specific soil contamination measurements were also performed in the former USSR in a region north of Chernobyl, with values of up to 2 MBq/m^2 for nuclide deposition, and in selected areas in Finland. The results from these measuring campaigns are presented here.

INTRODUCTION

The Federal Office for Radiation Protection has been involved in developing measuring methods for monitoring nuclear power plant emissions and immissions since the early 70ies. There were also early investigations into the possibility of using aerial measuring systems in helicopters for monitoring radioactive substances released into the environment, particularly those due to nuclear accidents. As a result of these investigations the Federal Office for Radiation Protection developed and tested a measuring system to be used for aerial measurements. A prototype of this aerial device - from industrial aspects - was constructed by the Henschel Aircraft Corporation in Kassel in 1990 and developed to the point of serial production. By the end of 1993 five industrially manufactured aerial measuring systems were delivered to the Federal Office for Radiation Protection.

MEASURING SYSTEM

The aerial measuring system consists of several components that can be separately installed in an Alouette II type helicopter. The central unit is a computerized gamma ray spectrometer with a high purity Ge-semiconductor detector of a relative efficiency of 50 %. In addition, a NaI(Tl)-scintillation detector array (three individual prismatic detectors each with a volume of 4 l) is used for detecting sources of radioactivity. Each crystal is equipped with its own photomultiplier tube. The detector array is mounted in an aluminum box on shock absorbers and temperature insulated. The computer is equipped with spectrometric cards, each fitting the high voltage and the power supply for the detectors and, further, with main amplifiers and analog-digital-converters. The adjustment of the individual measurement parameters is computer-aided and all measurement values shown in flight are recorded on a monochrome screen. The altitude is continuously measured by radar altimeter. The indicator gauge instrument is fitted to the instrument panel for easy in-flight viewing by the pilot. The position of the helicopter is determined by a GPS satellite navigation system. In addition, the flight route is recorded by a CCD video camera with wide-angle lens and a video recorder with integrated color display. The technician's remarks are also recorded on video magnetic tape. The instruments are designed to work on a supply of 28 Volts DC available in the helicopter. The installation of all components in the Alouette II type helicopter was done by Henschel Aircraft Corporation. Altogether, five helicopters of the type Alouette have so far been properly refitted to house the measuring device. This includes the radar altimeter with antennas, the indicator gauge, cabling and the necessary power outlets in the cell wall. Likewise refitted were mountings and power cables for the video camera.

At a measuring time of 60 s, a ^{137}Cs soil contamination of some kBq/m^2 can be measured from an altitude of 100 m (flying speed 100 km/h). With the NaI(Tl)-detectors sources of gamma radiation of an activity of some GBq can be detected (altitude 100 m, line spacing 300 m, flying speed 100 km/h). Under such conditions, one helicopter can cover an area of about 30 km^2 per hour.

The entire measuring device can be installed or removed from properly refitted machines by two persons within about 30 min. The equipment has a weight of 120 kg. The expenses for procurement, including refitting costs, are about 400,000 DM per unit. The helicopters are operated by the Federal Border Police. The German Federal Aviation Agency has licensed this system to be operated in an Alouette II helicopter.

RESULTS

Measurements in southern Germany

After the Chernobyl reactor accident aerial measurements were performed to measure the deposition of artifical radionuclides in the southern part of Germany. The region between Danube river and the Alps was flown in parallel routes at distances of 10 km and a flying speed of approximately 100 km/h. The average altitude was 100 m above ground level. A measuring time of 60 s was chosen for each individual spectrum. Under these conditions, each of the measured values represent a mean contamination value over a distance of about 1.7 km. For evaluating the spectra, a ^{137}Cs activity distribution in soil was used as determined from soil sample measurements in areas monitored by these flights. In most cases, fields (unknown activity distribution) and smaller wooded areas were of little influence. Figure 1 shows results from a measurement flight in southern Germany with the highest ^{137}Cs-deposition values. Along this flight route from Munich in the direction of the Austrian border, a ^{137}Cs soil contamination was measured, ranging between 10 kBq/m^2 and 60 kBq/m^2.

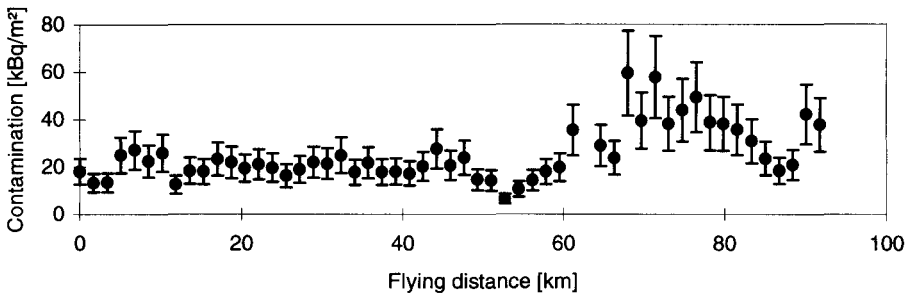

Figure 1. ^{137}Cs soil contamination along a flight route from Munich to the Austrian border.

The measurement uncertainty was estimated to be 30 %. Apart from the uncertainties in determining the photopeak count rate, inhomogenities in the distribution of activity in soil and variations in the altitude above ground contribute essentially to the measurement uncertainty. The measured ^{137}Cs values are in good agreement with the results from measurements obtained by other methods (1).

Measurements in the former USSR

This aerial measuring system (as a prototype) was used for nuclide specific measurements of soil contamination in the former USSR in a region north of Chernobyl with a ^{137}Cs contamination of up to 2 MBq/m^2. One flight was taken from the airport in Mogilov in Belorussia in south-eastern direction to an area near the town Krichev, approximately 100 km south-east of Mogilov and 260 km north of Chernobyl. A further flight was performed south of Mogilov covering a total distance of approximately 200 km. The soil contamination was measured continuously with the HPGe-detector. The flight altitude was approximately 100 m, the measuring time for each spectrum was 60 s. For the evaluation of the measured spectra, a relaxation depth of 6 cm was used. ^{137}Cs and ^{134}Cs could be measured by these flights. As an example, Figure 2 shows the soil contamination due to $^{137137}Cs$, measured during the outward and return flight from the Mogilov airport to the above mentioned measuring site near Krichev. At a flying speed of 150 km/h each value corresponds with a distance of about 2.5 km. At the return flight - another flight route north of the outward flight route - only one area with a ^{137}Cs soil contamination of up to 500 kBq/m^2 could be seen.

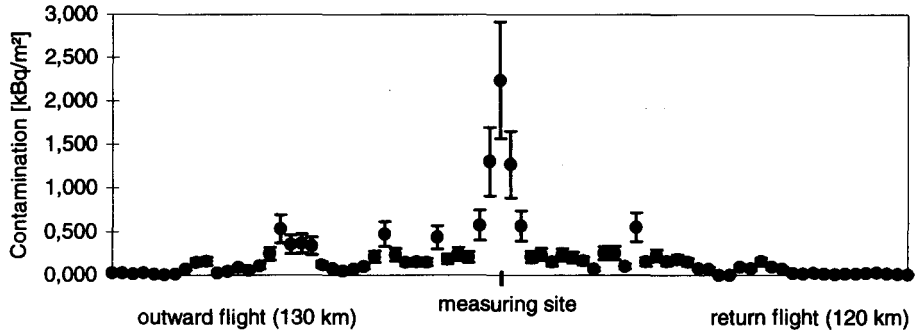

Figure 2. ^{137}Cs soil contamination along the outward and return flight route from Mogilov airport to a measuring site close to Krichev (Belorussia).

The first flight was taken over an area with relatively low contamination in south-eastern direction (< 200 kBq/m^2) to an area with a contamination between 200 and 600 kBq/m^2 in the middle the outward flight route of followed by an area with again low contamination. The following areas with differently high contamination were also clearly detectable. South-west of Krichev at a selected measuring site, a ^{137}Cs soil contamination of more than 2 MBq/m^2 was measured. At this site additional measurements of soil contamination were performed from various altitudes within the scope of an international intercomparison measurement campaign (data not reported here). The measured value for ^{134}Cs was (250 ± 70) kBq/m^2. A comparison with other data show a good agreement within the scope of accuracy (2).

Intercomparison measurements in Finland

In August of 1995, the Federal Office for Radiation Protection had an opportunity to participate in an international measuring campaign in Finland, using the helicopter measuring system. Eleven measuring teams from seven European countries participated in this exercise. Nuclide specific measurements were performed to determine the soil contamination. A selected area (3 km x 6 km), about 120 km north-west of Helsinki, was surveyed in parallel lines at an altitude of about 70 m and in 150 m parallel line spaces. The soil contamination was measured with the here described computerized Ge-semiconductor gamma ray spectrometer. The measuring time for each spectrum was 30 s, corresponding with a flying distance of 850 m (at an flying speed of 100 km/h) for which an average soil contamination value can be measured. For the evaluation of the measured spectra, a relaxation depth of 1 cm was applied, measured from soil samples. Figure 3 shows the results from the 21 flight routes covering this measuring area. The average ^{137}Cs-value was measured to be about 70 kBq/m^2 with a maximum value of up to 120 kBq/m^2.

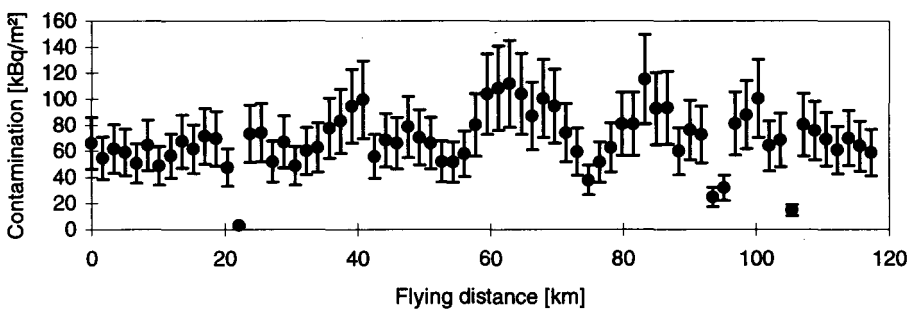

Figure 3. ^{137}Cs soil contamination along the flight routes covering an area of 3 km x 6 km, 120 km north-west of Helsinki (Finland).

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AERIAL PLATFORM EQUIPPED FOR NUCLEAR EMERGENCY MEASUREMENTS

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INTRODUCTION

Aerial Radiological Measuring Systems (ARMS) developed in the late 1950s for uranium ore prospecting have also provide an invaluable tool in environmental surveillance of nuclear power plants and nuclear processing facilities as well as in emergency response for large-scale radiological accidents. Historically the international concern of the potentialities of such systems was triggered in 1978 by the search and partial recovery of radioactive debris scattered over an estimated area of about 100,000 km^2 as a consequence of the impact in the Northwest Territories of Canada of the nuclear powered Soviet "COSMOS-954" satellite(1). Less than 10 years after, in occasion of the Chernobyl accident, many countries had developed such systems and maps of surface contamination have been collected over several European countries showing a large diffusion of the radioactive fallout originated by the release from Chernobyl nuclear plant (2,3). Also in Italy an ARMS mounted on Agusta-Bell 412 helicopter having as detector large NaI(Tl) counters was used by the group of Physics Laboratory of Istituto Superiore di Sanita' to map the ground surface contamination over the central-southern part of the country(4). From that experience it was deduced that aerial platforms without air sampling and air radioactive contaminants measurement capability can only give quantitative information on ground contamination when the air contamination is negligible and then, in the more critical period when the radioactive plume is passing over the country, it can only be used for qualitative assessments. We have now equipped a fixed wing aerial platform with sampling and measurement capability that can be used to give fully quantitative information on the plume as well as on the ground radioactive contamination in a far field emergency situation.

PARTENAVIA OBSERVER P68 AERIAL PLATFORM

In early nineties, ALENIA and ELECOS (5) have developed a measuring system (called SNIFFER) mounted on Observer P68 aircraft to monitor air quality, with respect to aerosol and gaseous pollutants, at flight height variable between 50 and 1000 meters. Observer P68 is a light two-engined plane with the whole structure in aluminium but the front cap that is made of Plexiglas.
The sampling unit is installed in this cap and the sampling probe goes out from the front part (see Fig. 1). Aerosol is collected on a filter positioned along the sampling line. A meaningful sampling of aerosol is provided through a control system that regulates the air flow according to the aeroplane speed and compensates for temperature and pressure variations as well as for the filter progressive cloggage. In this way an active isokinetic sampling is maintained assuring an entrance flow velocity equal to the aeroplane translation velocity. The entrance flow can be regulated within an error of 1% in the range 35-70 l/minute.
The rotating filter holder has four filter locations, one for each of the possible independent aerosol collections during the same flight mission. Behind the filter that is aligned with the suction line (see Fig. 2), a Geiger counter (external diameter 1 cm) and a small (1x1x1 cm^3) BGO counter, allow real-time measurements of gross-beta activity, total gamma activity and low resolution gamma spectra. Beta counter scaler and gamma spectra are stored in predetermined time intervals that can be as short as 1 minute and that can be changed during the flight mission according to the actual contamination levels. Results of measurements are continuously presented on the on-board monitor. The aircraft is equipped with an advanced Global Positioning System that allows the correlation of stored data with the spatial and temporal localisation. In this way the final result of the mission can be presented mapping out the measured values on the mission path flight.
The smallness and intrinsic poor resolution of the BGO counter makes that the gamma spectrum measured, even when only one radioactive contaminant is present, is very broad (see Fig. 3 left) allowing then a valid assessment only on the presence of the radioactive contamination but hardly the identification of the radionuclide

can be attempted. When a mixture of contaminants, as is always the case for a release from a nuclear plant, is present the identification of gamma emitters is out of any realistic possibility.

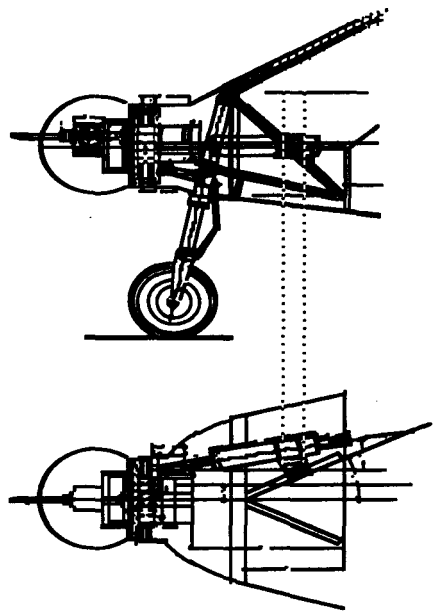

Fig.1 OBSERVER P68 with air sampling unit and HPGe detector

To overcome this failure of the radioactive contaminant measurement, the whole mechanism of the filter holder and connected services has been completely redesigned in a different way in order to allow the insertion of a 20% efficiency, high resolution HPGe detector. A new SNIFFER system with the redesigned mechanical parts was built. In Fig. 2 the scheme of the new system with the insertion of the HPGe detector is shown.

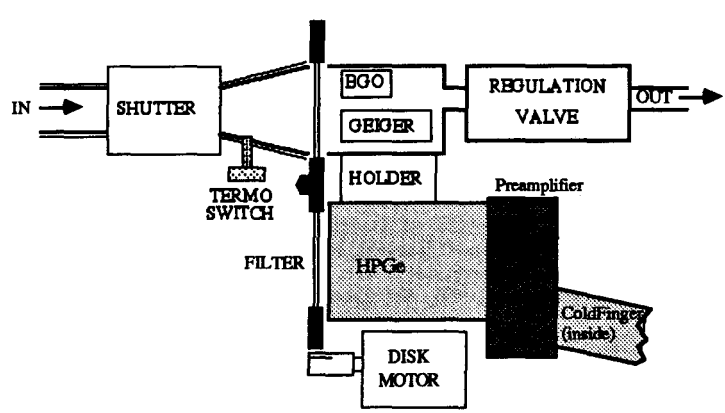

Fig. 2 Schematic diagram of sampling line with radioactive contamination detectors.

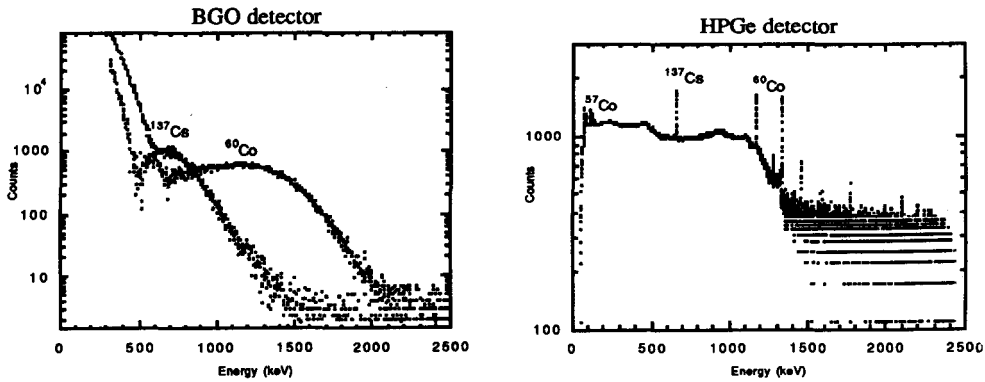

Fig. 3 Left: Separate Gamma Spectra obtained with BGO detector for ^{60}Co and ^{137}Cs sources
Right: Total Gamma spectrum obtained with HPGe detector for the three sources added together

In this way at the end of one sampling period, after the rotation of the filter holder, it is possible to face the filter, that was placed before the rotation along the suction line, to the high resolution detector that allows the identification of each gamma emitting radioisotope in the aerosol sample previously collected. The HPGe detector was specially designed in collaboration with Canberra Semiconductor N.V. to take into account the particular application and the geometrical constraints imposed by the installation of the whole system in the aircraft. A slim Canberra type ACT-1 dewar is used with 12 hour holding time that is well compatible with a flight time duration of 4-5 hours for a single mission. To reduce microfonicity deterioration of the resolution the preamplifier is mounted as close to the crystal as possible. A 20 cm. long cold finger provides the low temperature to the crystal. The resolution of the detector is 2.0 KeV for the 1.33 MeV line of ^{60}Co and 1.28 KeV for the 122.06 line of ^{57}Co. In Fig. 3 right, the spectrum obtained when three sources (^{57}Co, ^{137}Cs and ^{60}Co) are faced to the HPGe is shown. It is possible to see how the improved resolution allows the clear identification of different contaminants. Moreover while with the BGO the minimum detectable energy is little less than the energy of the ^{137}Cs source (661.66 KeV), with the new detector this limit can be lowered up to 100 KeV.

On board of the aircraft, large surface NaI(Tl) counters, which have a much better resolution than the small BGO counter spectrum shown in Fig. 3, and also HPGe high efficiency and high resolution counters are placed to collect gamma spectra from the total environment. From the information collected by these counters after the subtraction of the contribution whose source is in the air, that has been inferred from the air sampling measurements, assessments on the ground contamination can be deduced even in the early phase when the plume, coming from a far field release, cross over the interested country.

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THE DEPOSITION AND METABOLISM OF METHYL [125]IODIDE BY CROPS

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ABSTRACT

Several studies have identified an organic fraction of radioactive iodine releases following nuclear accidents, as well as releases arising from reprocessing activities. Methyl iodide has been identified as this primary component of this fraction.

The following paper describes a system for exposing crop plants with methyl iodide, using ^{125}I as an analogue for ^{131}I and ^{129}I. From these experiments deposition velocities to a leafy green vegetable were calculated at several points within the growth cycle. The subsequent fate of the ^{125}I was monitored until the crops were harvested at maturity.

INTRODUCTION

Radioisotopes of iodine are important radiologically, because of their ability to accumulate in the thyroid gland and their rapid transfer through the foodchain. Emission of radioiodines from nuclear facilities can occur in both routine operation and accident scenarios. Studies of the speciation of radioiodine have reported that a significant fraction is in an organic form (1,2), this is believed to methyl iodide (3). To date there is some work reporting deposition velocities for CH_3I (4,5), but little on its subsequent fate in the crop. The work outlined below was undertaken to rectify this situation by measuring the movement of activity between exposure and final harvest.

MATERIALS AND METHODS

The methyl [125]iodide was prepared by the reaction of potassium iodide and dimethyl sulphate. The resulting $CH_3^{125}I$ was transferred in a stream of helium passing through the reaction vessel and cryotrapped in the exposure vessel which was submerged in liquid nitrogen. Following preparation c.350kBq of $CH_3^{125}I$ was retained in the exposure vessel. To reduce confounding effects from dissociation of the methyl iodide, sources were prepared on the day of use and stored in the dark until required.

Cabbage (cv. Greyhound) seeds were sown in 3 l pots in John Innes no.2 compost on 5/4/95 and thinned to one plant per pot two weeks after germination. Plants were kept in an outdoor enclosure and irrigated automatically, a multipurpose feed (Phostrogen; Phostrogen, UK) was applied fortnightly.

At three occasions during the season, 76, 128, 159 days from sowing (DFS), plants were exposed to the prepared $CH_3^{125}I$. Twenty four hours before exposure twelve plants were transferred to a perspex chamber (0.7m x 0.7m x 0.8m, w x d x h) within a plant growth cabinet. The perspex chamber was part of a recirculating system of pipework that allowed the atmosphere of $CH_3^{125}I$ to be passed over the plants for one hour. The pump of the recirculating system was started after plants were placed in the chamber this allowed the conditioned air of the growth cabinet, 20°C and 50% RH, to enter the exposure system. On the day of fumigation the front of the perspex chamber was sealed and the activity immediately dispensed into the incoming air stream of the chamber. The activity was injected into the chamber by displacing the gas from the exposure vessel with water. The activity circulated within the system for the exposure period of one hour. The air concentration of the ^{125}I in the chamber was derived from a suck sample of 400 cm^3 min^{-1} passed through TEDA impregnated charcoal throughout the exposure period.

At the end of the exposure four plants were randomly selected and dissected into leaves, stems and roots. The remaining plants were returned to the enclosure for three later harvests; two days following exposure, a middle harvest and final harvest at maturity. The plant material was then oven

dried and direct counted on a gamma counter (Compugamma, Wallac). From these counts deposition velocities and the loss of activity from the crop calculated.

RESULTS AND DISCUSSION

The mass normalised deposition velocity (Vd) did not vary significantly between harvests (Table 1.) This suggests that the uptake of $CH_3{}^{125}I$ is directly related to the live weight of the crop. This finding indicates that as the crop grows throughout the season the total deposition of activity per unit area will increase. The rates reported here agree well with those of Voillque and Keller (4), but are an order of magnitude lower than those of Atkins *et al.* (5).

Exposure	Mass normalised deposition velocity ($\times 10^{-3}$)(+\- s.e.) (cm3 g-1 s-1)	Crop
76 DFS	1.51 +\- 0.376	Cabbage
128 DFS	1.38 +\- 0.306	Cabbage
159 DFS	1.27 +\- 0.589	Cabbage
	3.0	Grass (3)
	50	Grass (4)

Table 1. Deposition velocities of crops exposed to $CH_3{}^{128}I$.

The activity directly after exposure was concentrated in the aerial parts at all exposure dates (Figs 1.,2.,3.). As the season progressed activity was lost from all plant components, this loss was more rapid from leaves compared with roots. In fact there appeared to be an indication of transport from the leaves to the roots between 0 and 2 days after exposure. This later finding may be of importance when considering root crops. There may also be the possibility of transfer from the soil to the roots following uptake by the soil during the exposure period, a transfer factor of 9 was reported for the movement of iodine to rice roots (6). The apparent increase in the total activity between 0 and 2 days following exposure at 76 DFS was a result of larger plants at the 2 day harvest (mean weight 0 days from fumigation 18.0 g, mean weight 2 days from fumigation 42.8 g).

The loss of activity from the cabbage plants was greater than that expected from radioactive decay following exposure at 128 and 159 DFS (Fig. 4), Unfortunately the data from the 79 DFS exposure were confounded by the high values recorded 2 days after exposure as discussed previously. There are two possible pathways for this loss of activity, shedding of old leaves containing activity or gaseous re-emission of activity from the plant tissue. Loss of radioiodine in a gaseous form following absorption has been observed (6,7).

In future modelling of the uptake by vegetation of radioiodine as methyl iodide, it should be noted that a significant deposition is observed and that the biological half-life is shorter than that attributable to radioactive decay. The fate of the losses from the system needs further investigation, as these will be of importance for movement of the long lived ^{129}I isotope within an ecosystem.

ACKNOWLEDGEMENTS

The authors wish to thank the UK Ministry of Agriculture Fisheries and Food for the funding of this study.

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Figure 1. Fraction of initial activity in plant components following exposure to $CH_3{}^{125}I$ 76 days after sowing

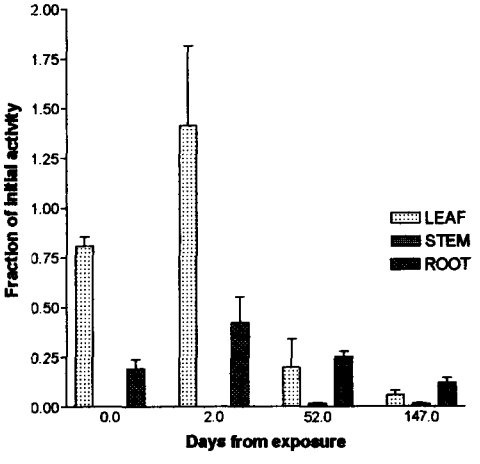


Figure 2. Fraction of initial activity in plant components following exposure to $CH_3{}^{125}I$ 128 days after sowing

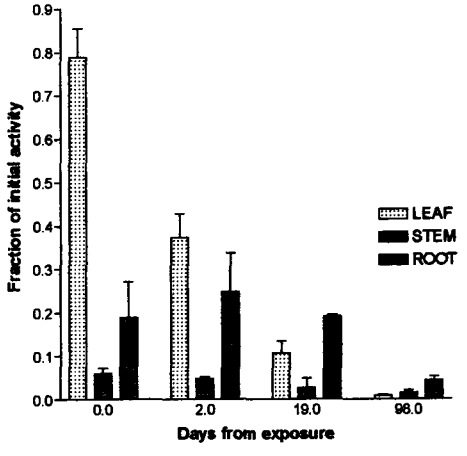


Figure 3. Fraction of initial activity in plant components following exposure to $CH_3{}^{125}I$ 159 days after sowing

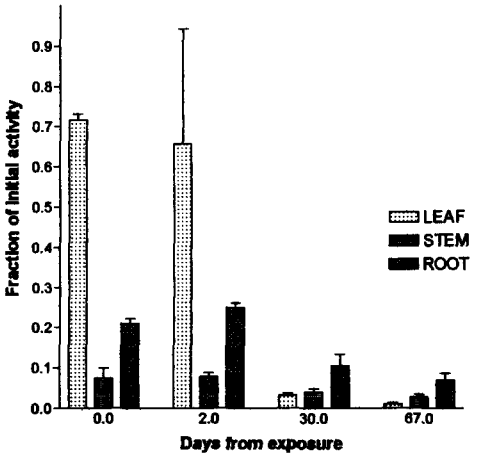
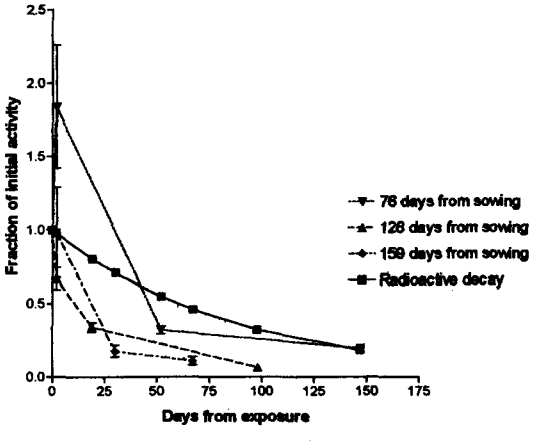


Figure 4. Loss of activity from cabbage following exposure at three separate times in the season



SEMICONDUCTOR GAMMA-SPECTROMETRY SYSTEM FOR AIRBORNE SURVEYING OF CONTAMINATION OF LARGE AREAS.

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INTRODUCTION

The Chernobyl accident has clearly demonstrated the importance of establishing environmental monitoring systems for obtaining rapid information in accidental situation on the radiological conditions at the affected area and providing suitable data to the competent authorities for decision making. Mobile units suitable for monitoring of radiological impact at any selected location in the field play an important role in emergency preparedness for various types of nuclear accidents:

1. A major accident in a domestic nuclear power plant.
2. Accidents in nuclear power plants abroad especially in neighbouring countries. To this category should be added the risk from military nuclear power accidents.
3. Re-entry of nuclear powered satellites. The probability is small but it can not be excluded. This type of accident would result in highly radioactive fragments and particles that would have to be located.
4. Accidents when transporting radiation sources.
5. Illegal handling of radioactive sources and nuclear material.

For monitoring of large contaminated areas especially the possibility of airborne monitoring is important. Airborne systems can be used for the measurement of dose and dose rate in wide range, nuclide specific ground surface contamination and nuclide specific activity concentration in air (with special attention to iodine in aerosol) as well as particle size distribution.

METHOD

For the Radiation Monitoring Network of the Czech Republic a prototype of the small system for dose rate measurements and nuclide activity estimates suitable for aerial gamma spectrometric determination of ground contamination was designed calibrated and tested in National Radiation Protection Institute (1).

The system consists of semiconductor HPGe detector, scintillation 3"x 3" NaI(Tl) detector, two multichannel analyzers, high pressure ionization chamber or proportional counter and portable computer working in multitasking mode for storing and evaluating of the spectra as well as for dose rate data handling, recording, storing, searching and presentation. One of three available HPGe detectors with relative efficiency from 15 to 55% can be used. Parameters of detectors are summarized in Table 1.

Table 1 Parameters of semiconductor HPGe detectors used for airborne spectrometry

Detector	Manufacturer	Rel. efficiency [%]	FWHM [keV]	diameter [mm]	length [mm]
JAKUB	Canberra	16.2	1.8	50	37
FERDINAND	Canberra	35.8	1.9	59.5	53.5
ISIDOR	EG&G Ortec	55	1.8	63.3	84.6

For accumulation of spectra the MCA Portable Plus a InSpector (Canberra Inc.) are used. The information about actual position is provided by the global positioning system (GPS) Garmin on-line connected to the computer. This information is used later on to create the maps of contamination using small desktop mapping (GIS) system. The system is powered from inner batteries or from external 12V car batteries.

One of the most important parameters of the system is the time needed for transferring the spectrum from MCA to storage medium (either PC or tape). Storage on tape takes for MCA S10 about 4 minutes which at the aircraft speed about 120 km/h leads to path uncovered by measurement of about 10 km. This time can be substantially shorten using MCA with remote control directly from PC down to about 5s. Further shortening is possible using the plug-in card emulating MCA in computer and communicating directly via bus. This solution is, however, rather demanding from the point of view of power supply on board of the aircraft and also the use of small notebooks is limited. Other possibility is to use special processors e.g. transputters, but this solution is quite expensive. In NRPI the MCA PortablePlus with LapTop computer Toshiba 1600 (PC AT 286, 12MHz,

math coprocessor, 3MB RAM, 40MB HDD) were tested. The storage time of the 4K spectrum using ASAP software was optimised to 8s. The use of the MCA InSpector together with 486/DX2-66 notebook with 8MB RAM and with Genie-PC software running under IBM OS/2 operating system led to further shortening of storage time to approx. 4s.
Whole system was repeatedly tested on the deck of both fixed wing aeroplane and helicopter with the flight speed of 120 - 130 km/h. The flight height was 60m for the helicopter and 80m for the aeroplane. Spectra were accumulated 60s which represent about 2.2 km of flight.
The classic scintillation spectrometer for geophysical surveys was used for comparison and verification of calibration. The spectrometer consists of NaI(Tl) detectors with total volume of 33.6 L. One additional detector with the volume of 4.2 L is used for measurement of the cosmic component contribution to the spectrum. Measured spectra are accumulated in 256 channel analyzer GR 800 D (Geometrics). The energy range of measured spectra is 0.2 - 3 MeV. The energy calibration is are stabilized using the ^{40}K peak. Measuring time is 1 s which represents about 40 m of flight. The spectrometer was calibrated on calibration pads with known amount of K, U, Th.

RESULTS AND DISCUSSION

The system was calibrated using standard method for in-situ spectrometry, elaborated by Beck. The calibration was verified by measurement with detector ISIDOR (55% relative efficiency) on the deck of MI-17 helicopter hanging 50m above the area with known deposition and depth distribution of ^{137}Cs and known specific activity of ^{40}K. For ^{40}K the homogeneous depth distribution was assumed. The activity of ^{137}Cs and ^{40}K and the depth distribution of ^{137}Cs activity were determined by the on-ground in-situ spectrometry and by soil samples measurement in the laboratory. Detection limits for two tested detectors for ^{137}Cs with various depth distribution in the soil, measuring time 60s and height of flight 80m are summarized in Table 2.

Table 2 Detection limits for two tested detectors for ^{137}Cs

Detector	relaxation depth [cm^{-1}]	MDA [kBq.m^{-2}]
FERDINAND	0.3	5.6
	∞ (plane source)	3.3
ISIDOR	0.3	2.9
	∞ (plane source)	1.7

The system was intercompared with scintillation spectrometer for geophysical surveys on two partly overlapping areas of about 100 km^2 each. The areas were chosen with regard to level and inhomogeneity of Chernobyl fallout i.e. there was spots both high and low deposition of ^{137}Cs. The minimum deposition of ^{137} Cs in chosen area was about 2 kBq/m^2, the maximum deposition was about 40 kBq/m^2. The average activity of ^{137}Cs was about 15 kBq/m^2, i.e. high enough to be reasonably measured also by semiconductor spectrometer. The distance of flight lines was 250 m. For comparison the mean values of activity of ^{137}Cs and ^{40}K from 60 measurements by scintillation spectrometer covering the measuring intervals of semiconductor detector were used. The example of ^{137}Cs activity along one flight line as determined by detector ISIDOR is in Figure 1.

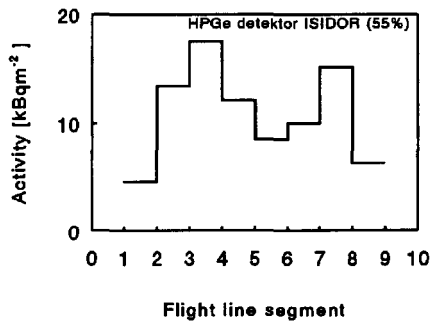

Figure 1 ^{137}Cs activity along one flight line

For detector FERDINAND (36% relative efficiency) the results of 65 measurements of ^{137}Cs deposition were compared with scintillation spectrometer, the mean ratio of activity determined by the system with semiconductor detector and the activity determined by the scintillation spectrometer is 1.11 with minimum of 0.55 and maximum of 1.97. It should be noted that due to the extreme inhomogeneity of fallout in chosen area there were also places where the activity of ^{137}Cs was very near or even below the detection limits of both systems. The extreme values of the ratio were found on such places.
For detector ISIDOR (55% relative efficiency) the results of 214 measurements of ^{137}Cs deposition were compared with scintillation spectrometer, the mean ratio of activity determined by the system with semiconductor detector and the activity determined by the scintillation spectrometer is 1.28 with minimum of 0.55 and maximum of 3.13 (Figure 2). Again it should be noted that due to the extreme inhomogeneity of fallout in chosen area there were also places where the activity of ^{137}Cs was very near or even below the detection limits of both systems. The extreme values of the ratio were found on such places. Efficiency of detector ISIDOR is high enough to enable also the determination of activity of ^{40}K in soil.Results of 214 measurements of ^{40}K activity were compared with scintillation spectrometer, the mean ratio of activity determined by the system with semiconductor detector and the activity determined by the scintillation spectrometer is 0.86 with minimum of 0.53 and maximum of 1.41 (Figure 3).

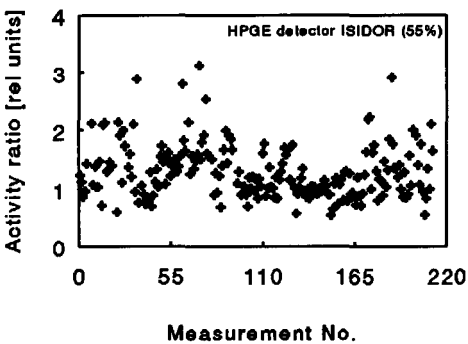

Figure 2 Ratio of ^{137}Cs activity determined by the system with semiconductor detector and by the scintillation spectrometer

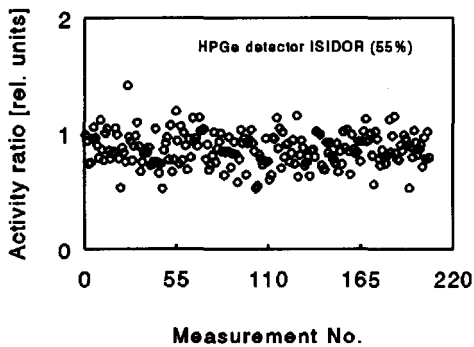

Figure 3 Ratio of ^{40}K activity determined by the system with semiconductor detector and by the scintillation spectrometer

CONCLUSION

The test use of the system with HPGe detector proved that it can be used for contamination mapping on large areas especially in the case of a nuclear accident when its lower sensitivity (in comparison with scintillation spectrometer) can be even an advantage and when one can make full use of its excellent energy resolution for analysis of complex spectra.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Radioactive Contamination of the Adriatic Marine Organisms by Fission Products

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The fallout resulting from the atmospheric nuclear weapon tests represented the dominant route of the radioactive contamination of the Adriatic sea, until 1986 Chernobyl nuclear accident. For the post-Chernobyl period ^{137}Cs and ^{134}Cs were found to be major sea-water and therefore marine organisms' contaminants, while ^{90}Sr activity levels were not affected.

For the post-Chernobyl period transfer of radiocaesium from fallout to sea water and finally to marine organisms' was studied. A simple mathematical model was used to describe decrease in radiocaesium activity in marine organisms. Turnover time for radiocaesium in various marine species was calculated to be approximately two years.

Accumulation of Radionuclides under Determined Conditions in Trouts and Carps

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1)Introduction

In the literature [1] the value of the biological half-life of the nuclides Co-60, Zn-65, Sr-85 and Cs-137in fish ranges from 10 to 1000 days. The object of this study was to determine the intake of radionuclides from water and contaminated food by trouts and carps, two important freshwater fishes, to estimate the biological and effective half-lives, to compare or verify the found values with in the literature and to estimate the transfer factors food/fish after feeding the fishes with contaminated food in aquarium experiments. This study also examines the effect of the salinity of the water an the accumulation of the radionuclides.

2)Results and Discussion

2.1 Accumulation of radionuclides from water:
The accumulation of the radionuclides is expressed by equation (1):

$$[R]_t = [R]_\infty\{1 - \exp(-kt)\} \qquad (1)$$

$[R]_t$ and $[R]_\infty$: concentration of radionuclide at time t and steady state concentration (Bq/kg fresh weight
k : excretion rate of 0,692/ t 1/2, t 1/2 being the biological half-live (days)

All radionuclides with the exception of Co-60 in carps were found in the two species. By fiiting the rapidly increasing part of model curve to measured points the steady state concentration $[R]_\infty$, the rate constant k and the biological half-live are obtained.. With the mathematical formula (2) the effective half-live can be estimated. The calculated values are shown in table 1. The sign "-" means: No determination

$$0.693/T(\text{Eff.}) = 0.693/T(\text{Bio.}) + 0.693/T(\text{Phys.}) \qquad (2)$$

Table 1: Biological and effective half-lives (days) after intake from water in carp and trout.

Nuclide	Carp		Trout	
	T(Bio.)	T(Eff.)	T(Bio.)	T(Eff.)
Mn-54	27.2 ± 2.0	25.2 ± 1.7	27.6 ± 10.1	25.4 ± 8.2
Co-60	-	-	14.5 ± 0.9	14.4 ± 0.9
Zn-65*	-	-	37.4 ± 2.8	32.2 ± 2.1
Sr-85	9.0 ± 4.6	7.9 ± 3.4	26.5 ± 3.9	18.8 ± 2.0
Cs-137	38.7 ± 6.0	38.6 ± 6.0	38.9 ± 4.4	38.7 ± 4.4

* A good fit for zinc to obtain was unavailable. The biological half-lives for Cs-137 agree very well with those determinated in field experiments. In the literature [2] half-lives of 55 days are reported for one-to-two-yeras old trouts and for two-years-old roaches, related to carps.

Potassium and calcium are non-isotopic carrier elements of cesium 137 and strontium 85. If the carrier elements are homeostatically controlled, the concentration factor AF(R) of the radionuclides will be given by the equation (3).

$$AF(R) = q\,[C]_i/[C]_w \qquad (3)$$

$[C]_i$ and. $[C]_w$: Concentration of non-isotopic carrier element in organism or tissue i (µg/kg FS) and concen tration in water (µg/l)
q : Discrimination factor
The disrimination factor is expressed by equation (4):

$$q = \frac{[R]_i/[C]_i}{[R]_w/[C]_w} \qquad (4)$$

$[R]_i$: Concentration of the radionuclide in organism i
$[C]_i$: Concentration of the non-isotopic carrier in organism i
$[R]_w$: Concentration of the radionuclide in water
$[C]_w$: Concentration of the non-isotopic carrier in water
As mentioned above calcium and potassium are non-isotopic carrier elements of Sr 85 and Cs137, i.e. the AF(R)

of these nuclides should decrease with increasing concentration of the carriers. Taking the natural logarithm of each side of equation (3), the equation (5) is obtained:

$$\log AF(R) = \log q\,[C]_i^{-} - \log [C]_w \qquad (5)$$

Sincee $[C]_i$ is a constant, a plot of log AF(R) versus log $[C]_w$ should have a slope of -1, if q is constant. The trouts were kept in water compartments with an average potassium concentration of 1 mg/l, 10 mg/l, 100 mg/l und 200 mg/l and an average calcium concentration of 80 mg/l and 160 mg/l and the carps in one with an average pozassium concentration of 1 mg/l, 100 mg/l and 200 mg/l and an average calcium concentration of 80 mg/l and 160 mg/l. An increase of the calcium concenztation is not possible because a higher calcium concentration may be poisonous [3].The figures 1 - 2 show the results in double logarithmic presentation.

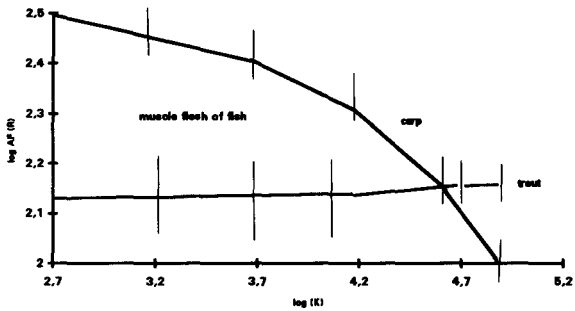

Fig 1: The dependence of the accumulation factors of cesium 137 in the muscle of carp and trout on the concentration of potassium in water.

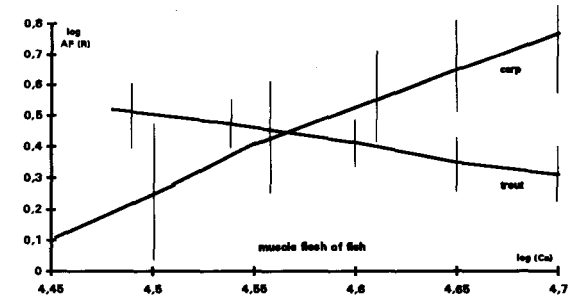

Fig .2: The dependence of the accumulation factors of strontium 85 in the muscle of carp and trout on the concentration of calcium in water.

In contrast to Vanderploeg et al [4], but in accordance with Whicker et al. [5], who examined trouts from mountain lakes in Colorado/USA the potassium concentration was of no effect on the intake of cesium in the muscle of trouts, kept in our compartment. The carps show a decrease of the accumulation factors by increasing potassium concentration from 1 mg/l to 200 mg/l..
The accumulation factor of strontium in the muscle of trout decreases with increasing calcium concentration, a phenomenon which was observed by Templeton et al [6], too. In contrast to the trouts the carps show an increase of the accumulation factor of strontium in muscle with increasing calcium concentration in water.

2.2. Intake of radionuclides from contaminated food:

All radionzclides in the food were found in both species. The accumulation of radionuclides may be described by the equation (1), too. By fiiting the fast ascent of model ,curve to measured points the steady state concentration R_{oo} and hence the biological and effective half-lives are obtained. The values are listed in table 2.

Table 2: Biological and effective half-lives of the radionuclides (in days) after intake of contaminated food in carps and trouts.

Nuclide	carp		trout	
	T(Bio.)	T(Eff.)	T(Bio.)	T(Eff.)
Mn- 54	41.3 ± 8.1	36.5 ± 6.5	15.2 ± 5.8	14.5 ± 5.2
Co- 60	26.5 ± 0.6	26.2 ± 0.6	10.3 ± 1.7	10.2 ± 1.7
Zn- 65	31.8 ± 2.5	28.1 ± 2.0	33.2 ± 5.6	29.2 ± 4.3
Sr- 85	8.0 ± 4.0	7.1 ± 3.6	12.4 ± 0.5	10.4 ± 0.4
Cs-137	39.0 ± 1.8	38.9 ± 1.8	46.7 ± 1.5	46.5 ± 1.5

The half-lives for cesium 137 agree well for the two species already quoted in the literature . There is reported a half-live of 47,2 days for the speckled trout [7] and a one of 35 days for carp[8], both determined after being fed with contaminated food.

The intake of contaminated food by fish is a two-comparment system. The transfer of the nuclides from the compartment food into the compartment muscle will be descibed by the formula (6).

$$T = [R]_{fish} / [R]_{food} \; 1/V \qquad (6)$$

T : transfer factor
$[R]_{fish}$: concentartion of the radionuclide in fish (Bq/kg fresh weight)
$[R]_{food}$: concentration of the radionuclide in food (Bq/kg fresh weight)
V : amount of feed (kg/day)

The calculated transfer factors (day/kg) are listed in table 3.

Table 3: Transfer factors (d/kg) of radionuclides for the pathway food/fish

Nuclide	carp	trout
Mn-54	$2x10^{-3}$	$3x10^{-3}$
Co-60	$2x10^{-3}$	$7x10^{-3}$
Zn-65	$2x10^{-2}$	$4x10^{-3}$
Sr-85	$2x10^{-2}$	$5x10^{-2}$
Cs-137	$6x10^{-2}$	$1x10^{-1}$

The transfer factors do not differ distinctly in the single nuclides in the two kinds of fish.

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The study was prepared within the scope of the project St.Sch. 959 sponsored by the Federal Minister of the Interior and the Federal Minister of the Environment, Natural Protection and Reactor Safety.

RESUSPENSION Cs-137 CAUSED BY PYROCLASTIC FLOW AT Mt. FUGEN, UNZEN VOLCANO, JAPAN

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After 198 years of long sleep, Mt. Fugen of Unzen Volcano has revitalized its volcanic activity in November 1990. Since May of 1991, large scale pyroclastic flows repeatedly occurrred in Mt. Fugen area have caused serious disasters. Pyroclastic flow, which has temperature of 500～650℃ at central portion and 300～400℃ at marginal portion, had frequently occurred since May of 1991. Because the boiling point of cesium and its compounds is known to be rather low (Metallic Cs : 678℃, Cs_2O : 490℃), fallout Cs-137 accumulated on surface soil will be vaporized by the interaction of high temperature pyroclastic flows. Vaporized Cs-137 will likely be exhaled into the atmosphere through extremely porous pyroclastic deposit of which porosity is measured to be ～40 %. In the present paper, we will show the experimental evidence of resuspension of fallout Cs-137 caused by pyroclastic flow.

Samples and Cs-137 measurement

Sampling of volcanic ashes (V), air-borne dusts (A), pyroclastic flow deposits (P), volcanic cinders (C), surface soils (S), leaf mold (L), moss (M) and rain waters (R) were made mainly after the occurrence of pyroclastic flows (from 28 May 1991 to 31 January 1994). Exceptional case is the sampling of volcanic ash from Fugen Pond (17 February 1991). Sampling locations are shown in Fig. 1. Airborne particles were collected immediately after the outbreak of pyroclastic flow at 3.6 and 13 km NNE (leeward of prevailing wind). Sampling was repeated two or three times at one hour interval. Rain waters were collected to evaluate the contribution of fallout Cs-137 derived from rain fall at 8 km E and 11 km NE points (Shimabara City), 35 km NW point (Omura City) from Mt. Fugen. As a control, airborne dusts and rain waters were also collected at Matsue City in Shimane Prefecture located about 420 km NE from Mt. Fugen.

Non-destructive gamma ray spectrometry has been performed to measure radionuclides particularly focusing on Cs-137. In the case that Cs-137 activity is extremely low, Cs-137 was measured by Compton suppression mode using NaI(Tl) detectors as guard counter.

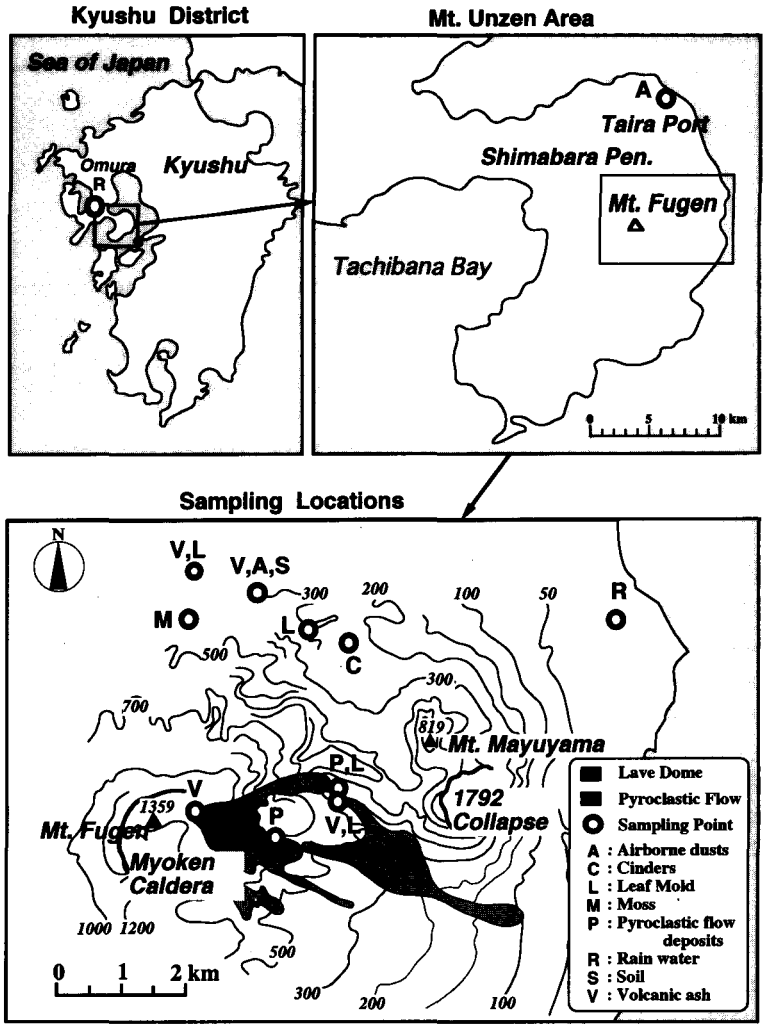

Fig. 1 Sampling locations of volcanic samples of Mt. Fugen, Unzen Volcano.

Results

Results of Cs-137 measurement are summarized in Table 1 and 2. Table 1 shows Cs-137 activities measured for volcanis ashes. As known from the table, Cs-137 was not detected in the sample collected from the shore of Fugen Pond (crater created by 1st eruption) on 17 February 1991 and pyroclastic deposit collected on 28 May 1991. On the other hand, high amount of Cs-137 up to 0.92 Bq/kg was detected in volcanic ashes collected on the same day and thereafter. Cs-137 could not be detected in the volcanic ashes collected since 16 July 1993 due to the reason that the ground surface accumulating high amount of Cs-137 was covered with thick volcanic ash and/or pyroclastic deposits.

Table 2 gives the concentration of Cs-137 in the airborne dusts collected immediately after the outbreak of pyroclastic flows on 28 May 1991 and 21~22 May 1993. As known from the table, Cs-137 activity in the airborne dusts decreased rapidly to 1/4~1/6 in the second sample collected only one hour later. Since Cs-137 was not detected in the airborne dust collected at Matsue, Cs-137 in airborne dusts of Mt. Fugen area may be attributed to the outbreak of pyroclastic flow. Cs-137 detected in volcanic ashes and airborne dusts is considered to be originated from re-suspension or volatilization of Cs-137 (probably as a cesium oxide) accumulated in the surface soils due to the heating by pyroclastic flows and deposits and not from the rain fall.

Above assumption is supported by the fact that Cs-137 was not detectable in rain water collected at Shimabara and Omura City. Concentration of Cs-137 was found to be increasing with depth of volcanic ash accumulated on ground surface soils suggesting that Cs-137 was supplied by underlying soil due to volatilization. Above assumption was supported by other evidences such as Pu-239,240/Cs-137 activity ratios measured for airborne dusts and volcanic ashes.

Table 1 Cs-137 measurments of volcanic ashes.

Location	Date of Sampling	Cs-137 (Bq/kg)
Fugen Pond	17-Feb-91	not detected
Mizunashi Riv.	28-May-91	0.58 ± 0.13
do.	do.	0.54 ± 0.11
do.	8-Jun-91	0.26 ± 0.03
Oshiga Valley	20-Aug-91	0.92 ± 0.07
Akamatsu Valley	13-Mar-92	not detected
do.	3-Sep-92	0.65 ± 0.06
Taira Port (13km)	21-May-93	0.64 ± 0.14
Nakao Riv.	22-May-93	0.21 ± 0.03
	16-Jul-93	not detected
	1-Sep-93	not detected
	8-Oct-93	not detected
	31-Jan-94	not detected

Table 2 Cs-137 measurements of airborne dusts.

Distance from Mt. Fugen	Date & Time of Sampling		Cs-137 activity (mBq/m^{-3})
	Date	Time	
3.6 km	28-May-91	11:06 - 12:00	9.31 ± 0.36
	do.	12:02 - 12:58	2.30 ± 0.10
13 km	21-May-93	18:45 - 19:22	2.34 ± 0.23
	do.	19:24 - 19:54	0.58 ± 0.13
3.6 km	22-May-93	10:47 - 11:47	0.32 ± 0.05
	do.	14:02 - 15:02	0.08 ± 0.02
	22-May-93	15:10	0.015 ± 0.002
	23-May-93	- 10:05	
Matsue (420 km)	31-May-91	9:00 -	not detected
	1-Jun-91	- 9:05	

STATUS RESULTS OF AN ANNUAL CAMPAIGN OF CONTINUOUS MEASUREMENT OF AIR RADIOACTIVITY IN NOVOZYBKOV (BRIANSK REGION) - RECENT DATA OF A RUSSIAN - FRENCH STUDY

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Abstract

At the request of the Association of Radiological Problems of Russia(ARPR), in charge of environmental measurements and clean up of contaminated territories by Chernobyl accident, an instrument measuring continuously the radioactivity of aerosols in the environment was installed and operated for demonstration during a complete annual cycle in NOVOZYBKOV, Briansk Region (RUSSIA), in order to determine locally residual atmospheric contamination.

This measuring campaign was carried out using the so called RADAIR instrument. This campaign was organised in the framework of a Russian - French cooperation between IPPE (Institute of Physics and Power Engineering, OBNINSK), SEC-SNIIP (Scientific and Engineering Center "SNIIP" , MOSCOW) , IPSN (Institut de Protection et de Sûreté Nucléaire , Fontenay-aux-Roses) and MGP-Instruments (LAMANON).

The first step of the operation consisted in a preliminary verification of the instrument calibration in the site, using aerosols sources deposited on filters . This calibration carried out on specialised facilities of IPSN(1) and SEC-SNIIP(2) showed a satisfactory agreement between different results.

The second step consisted in exploiting the instrument. Beside the demonstration that the instrument worked suitably during its continuous operation in variable climatic conditions, the results obtained showed :
- that the indicated volume activities of α and β emitters were low and near the instrument minimum values (for α : 4.10^{-3} Bq.m^{-3} and for β : 10^{-1} Bq.m^{-3}) ;
- that the indicated volume activities of ^{222}Rn lie between 2 and 30 Bq.m^{-3} (the high values are obtained in favourable climatic conditions : no wind, no rain or snow, soil not covered by snow and not frosted, etc.) ;
- that the indicated environmental dose rate values at the point of measurement were of the order of 0.15 μGy.h^{-1} (150 nGy.h^{-1}).

This paper summarises all the results obtained.

1 - INTRODUCTION

In order to survey the air volume activity of a region, for the purpose of radiation protection, most industrialised countries have installed air measuring networks. The characteristics and performances of measuring instruments in the network are specified in IEC standards.

The type testing of these instruments, according to these standards, gives the demonstration that the volume activity indications lie within stated requirements. It is also necessary to study the characteristics of the instrument in the real field and to give to the user the possibility to exploit it, in order to have feedback experience.

In february 1993, in response to a request from the Russian Association of Health Physics Problems (ARPR), the organisation in charge of environmental measuring and clean up of regions contaminated by Chernobyl accident in the Briansk area, and within the framework of BNEN protocol (Office for Standardisation of Nuclear Equipment) a recommendation was made to install and to operate for demonstration purposes, an instrument for continuous measuring aerosol radioactivity in the environment. This project was carried out jointly by IPPE (Institute of Physics and Power Engineering, OBNINSK), SEC-SNIIP (Scientific and Engineering Center "SNIIP", MOSCOW), IPSN (Institute of Nuclear Protection and Safety, Fontenay-aux-Roses and Saclay) and MGP Instruments (LAMANON). It has received financial support from the French Ministry of Foreign Affairs.

(1) ICARE : Installation de Calibration à l'aide d'Aérosols Radioactifs Etalons.
(2) SAS : Special Aerosol Source.

This project was designed for the following objectives :

1. to test and to exploit an instrument, the so called RADAIR, in the real field where climatic conditions might vary from severe cold winter to hot summer, during a year cycle ;

2. to measure continuously the aerosol radioactivity after ten years from the Chernobyl accident in NOVOZYBKOV (Briansk region), specially during seasons where aerosol resuspension might occur ;

3. to receive on line RADAIR data transmitted to Saclay by ARGOS system, so to survey the operation and the status of the instrument.

Prior to this operation, an intercomparison of activity deposited on filters produced by SAS facility (SEC - SNIIP) and ICARE facility (IPSN) serving during type testing of this kind of instruments was organised. The purpose of this intercomparison was to prove that calibration of α and β deposited activities in Russian and French laboratories are comparable within know limits.

2 - INTERCOMPARISON OF ACTIVITY MEASUREMENTS OF RADIOACTIVE AEROSOLS COLLECTED ON FILTERS

This intercomparison took place in 1993-1994 for the reason mentioned above. Filters of the same type, millipore membrane, AAWP 0.8 µm thick, were used. The characteristics of deposited activities were the following :

- for SEC - SNIIP Laboratory ^{239}Pu , ^{90}(Sr + Y) and natural U were deposited by generating with SAS facility polydispersed aerosol of the inhalable sizes.
- for the IPSN Laboratory ^{239}Pu and ^{137}Cs were deposited by generating with ICARE facility monodispersed aerosol of 0.4 µm.

The measurements were carried out in SEC-SNIIP and IPSN Laboratories using absolute and instrumental methods without knowing the values. The results were published in the reference [1]. They showed that calibration in Russian and French Laboratories of deposited activities, when the uncertainties are between ± 2 % and ± 5 %, for all radionuclides, except natural U, are comparable within ± 10 %. For natural U, the deviation of ± 20 % is due to uncertainties in the methods of measuring this radionuclide. Moreover, the fact that instrumental method gave comparable results with absolute one for submicron aerosol particulates proved that their penetration in the filtering media can be considered as negligible. This was a confirmation of the choice of these membrane as a reference filter.

3 - OPERATIONAL PRINCIPLE OF RADAIR INSTRUMENT

This instrument was chosen for the following reasons :

1 - it is rugged, easy to install in open air and to exploit ;
2 - it separate finer thoracic aerosol fraction of less than 10 µm for continuous measurements, while segregating the course extra-thoracic fraction and iodine in a collecting cartridge for later gamma spectrometry analysis.

The operational principle was published in reference [2]. Essentially for the continuous measurement function, RADAIR instrument carry out, on four measuring channels, the following readings :

- volumes activities of α and β artificial emitters in Bq. m^{-3} ;
- volume activity of natural radon in Bq. m^{-3} ;
- ambient γ dose equivalent to ^{137}Cs in µGy.h^{-1}.

The volume activity readings are derived from pulse counting in cycles of 1000 s of the activity deposited on AW 19 filter by sampling the environmental air. The filter automatically moves after remaining 24 hours in front of the stack of two semiconductor detectors. The first detector located above the filter, delivers a net pulse counting rate proportional to the deposited activity. The discrimination between artificial and natural radionuclides is achieved by the following way :

- radial collimating grid is placed between the filter and this detector to improve α resolution ;
- AW 19 membrane filter is chosen because the aerosols are collected superficially and so the α resolution is better ;
- pulse amplitude selection associated with α and β processing algorithms separate artificial from radon daughters emitters.

The second detector located above the previous one corrects the counting rate of the first detector from the influence of γ radiation and by this way performs ambient γ compensation. The counting from the second detector in cycles of 1000 s is converted to a γ dose rate display.

4 - PROJECT IMPLEMENTATION

Russian and French experts worked on the preparation of the project for three months before it became operational (from august 1994 to october 1994).

On october 30, 31, 1994, a RADAIR instrument was installed on site at the interdepartmental Radiology Laboratory in Novozybkov, located about 180 km from Chernobyl. An operational test was performed november 1, 1994. Calibration verifications were performed november 2, 1994, using aerosols of ^{239}Pu, ^{137}Cs and ^{90}(Sr + Y) deposited on filters, which were manufactured by methods using ICARE (IPSN) and SAS (SEC-SNIIP) installations. The results of these calibration tests were published in reference [3]. Taking into account that RADAIR uses a fixed efficiency of 3.10^{-2} pulses.s^{-1} /Bq to convert pulse rate into activity, it was found that deviations of indicated values from the nominal ones were within ± 15 %.

Since november 2, 1994, the instrument was in continuous operation and under surveillance of two IPPE persons who worked in rotation. Between november 2, 1994 and october 13, 1995, various calibration checks were performed. The results of these checks were the same as those obtained at the time the instrument was put into operation, november 2, 1994.

The RADAIR filters which have aspiration tracks lasting from 46 hours 40 minutes and 15 hours were analysed using α spectrometry (technique using a grid chamber or a silicon semiconductor). ^{226}Ra and products of ^{222}Rn and ^{220}Rn were found to be present in the α spectra.

5 - CONCLUSIONS

After examination of the IPPE exploitation report obtained using the results from the volume activities and dose rates indicated by RADAIR printer, the following conclusions have been drawn :

1 - overall, the volume activities of artificial α and β emitters are low and close to the minimum values (for α : 4.10^{-3} Bq. m^{-3} and for β : 10^{-1} Bq. m^{-3}).

2 - in general, the activity concentrations of radon remained lower than 30 Bq.m^{-3} .

The highest concentrations were obtained under the following conditions :

- light wind (speed of $\leq$ 2 m.s^{-1}), and high atmospheric pressure (P > 740 mm of mercury),
- no precipitation, and no snow or ice groundcover.

The lowest values, lower than 5 Bq.m^{-3} , were obtained under the following conditions :

- high winds (speeds of $\geq$ 5 m.s^{-1}), and low atmospheric pressure (P < 730 mm of mercury),
- groundcover of snow or ice,

3 - The γ dose rates were around 0.15 μGy.h^{-1} (150 nGy.h^{-1}).

This project has demonstrated that overall, the RADAIR instrument operated correctly in very severe climates (temperatures as low as -30°C, snow, high winds,...). The static calibration checks show the stability of the readouts, within statistical fluctuation limits.

Considering the minimum values mentioned above, the measurements show that, during winter the air contamination is not abnormally high. It was expected that during agricultural work and harvest in spring and summer resuspension might occur, but this was not observed. Similar results were obtained in United Kingdom and Japan [4], [5] in spring season from april to june. They were explained by the action of natural filter played by the vegetation, grass, trees... . From august to september 1995, volume activity readings more than the minimum values were not observed, even though the natural filter is expected to drop as indicated by the references mentioned above. Several examples shown as daily graphs (24 h), taken from the counting of the activity deposited on the RADAIR filter for the α, β, and ^{218}Po channels in 1000s cycles, and before algorithm processing, confirm these conclusions.

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ANALYSIS AND ASSESSMENT OF THE RESULTS OF INTERCOMPARISON RUNS ON DETERMINATION OF SR-90, CS-137 AND PU-239/240 SPECIFIC ACTIVITY IN SOILS SAMPLED IN THE VICINITY OF "MAYAK"

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I INTRODUCTION

In correspondence with the decision of the Co-ordinating Board on territory rehabilitation of Ural region, intercomparison runs on determination of Sr-90, Cs-137 and Pu-239/240 specific activity in three type of soils, sampled in environmental surroundings of "Mayak", were carried out in 1993-1994 within the Radiation Measurements Quality Assurance Programme under supervision of Principal State Centre of Measurements Unity (VNII Mendeleev, St.-Petersburg), V.G.Khlopin Radium Institute (St.-Petersburg), "Mayak" (Chelyabinsk-45).

Soil samples have been treated and homogenised by special programme. Two of these samples were certified as Standard Reference Material of Russian Federation on specific activity of Sr-90, Cs-137, Pu-239/240. Certified values of radionuclides specific activity are shown in TABLE 1.

TABLE 1. Certified values of Cs-137, Sr-90 and Pu-239,240 specific activity in Standard Reference Materials of soil (Bq/kg).

SRM	Cs-137	Sr-90	Pu-239,240
SOIL-10	48.2 ± 2.2	16.7 ± 2.2	10.6 ± 2.4
SOIL-24	2440 ± 140	560 ± 70	2.2 ± 0.35
SOIL-3	157 ± 8	87 ± 9	1.7 ± 0.3

II. MAIN RESULTS OF THE INTERLABORATORY RUN.

26 laboratories including 6 foreign laboratories participated in the interlaboratory run

Programme of intercomparison runs implied determination of Sr-90, Cs-137, Pu-239/240 specific activity in two type of soils (SOIL-10, SOIL-24). In addition some laboratories analysed third type of soil - SOIL-3. As a rule, a number of parallel determinations for each radionuclide in soil sample was 3-6. On the basis of the reported data an mean weighted activity value ($\overline{A}_i$) for i-th laboratory was calculated together with appropriate value of standard deviation.$S_{\overline{A}_i}$.Those data, which did not

meet a condition $|A_i - \overline{A}| \rangle 3 \bullet S_{\overline{A}}$, were rejected as outliers and not used for calculation of interlaboratory mean activity value. Main results of the interlaboratory run and some statistic for the measurement data obtained by the participants are given in TABLES 2-4.

TABLE 2. Summarised data for Sr-90 obtained by the participants of the interlaboratory run (Bq/kg).

SRM	Range of Lab Data A_i	Lab Mean Values and SD $\overline{A}_i \pm S_{\overline{A}_i}$	Number of data deviated from certified value less than 10%	Number of data deviated from certified value less than 25%
SOIL-10	13 - 57	18.6 ± 2.8	22%	67%
SOIL-3	60 -179	86.5 ± 9.6	38%	69%
SOIL-24	490 -950	550 ± 50	58%	84%

TABLE 3. Summarised data for Cs-137 obtained by the participants of the interlaboratory run (Bq/kg).

SRM	Range of Lab Data A_i	Lab Mean Values and SD $\overline{A}_i \pm S_{\overline{A}_i}$	Number of data deviated from certified value less than 10%	Number of data deviated from certified value less than 25%
SOIL-10	42 - 63	50.8 ± 1.4	75%	95%
SOIL-3	138 - 199	160.5 ± 6.5	80%	100%
SOIL-24	1840 - 2640	2480 ±140	85%	100%

TABLE 4. Summarised data forPu-239,240 obtained by the participants of the interlaboratory run (Bq/kg).

SRM	Range of Lab Data A_i	Lab Mean Values and SD $\overline{A}_i \pm S_{\overline{A}_i}$	Number of data deviated from certified value less than 10%	Number of data deviated from certified value less than 25%
SOIL-10	2.6 - 11.9	8.4 ± 2.9	22%	44%
SOIL-3	0.3 - 3.3	1.67 ± 0.26	33%	53%
SOIL-24	1.4 - 3.0	2.4 ± 0.4	41%	71%

II. CONCLUSIONS

The following conclusions could be made from analysis of the reported data:

- in majority of the laboratories (17 from 20) semi-conductive gamma-spectrometric method for determination of Cs-137 allows to obtain reliable data both at low and high levels of Cs-137 content. For instance, mean weighted value of Cs-137 specific activity in sample SOIL-10 as received from results of interlaboratory runs is 49.8±3.2 Bq/kg. Certified value of the specific activity in the same soil is 48.8±2.2. So, within the measurement error (±4%) these results are satisfactorily agreed with each other.

- after evaluation of reported data on Sr-90, systematic errors were detected in 11 laboratories from 18. Analytical methods in four laboratories shall be tested thoroughly because these methods can not be applied for environmental control on Sr-90 content in soils in range 15-100 Bq/kg and lower.

- as a whole, analysis on Pu content resulted in rather poor quality especially in case of SOIL-10 and SOIL-3. Eight laboratories from 18 reported non confident data for SOIL-10 and 8 laboratories from 15 reported non confident data for SOIL-3. One of the main reason for that (in case of SOIL-10) is application of analytical methods based on partial dissolution of (the many times leaching by acid mixture) of soil subsample. Our special investigations have shown that Pu exists in poorly soluble form in SOIL-10. Total dissolution (≥ 99.5%) of the subsample is required for analysis of this type of soil because only in doing this confident data can be obtained. SOIL-3 contains rather low concentrations of Pu (1.7±0.3 Bq/kg) and sensitivity of analytical methods in some laboratories are not enough.

- in total, the number of confident determinations of radionuclides specific activity in three type of soil from Ural region is as it follows:

Cs-137 - 85 % (47 from 55)
Sr-90 - 75 % (40 from 53)
Pu-239,240 - 60 % (30 from 50)

- the laboratories who have shown positive results in the intercomparison runs could certify their analytical methods in D.I.Mendeleev VNIIM together with V.G.Khlopin Radium Institute.

The laboratories who have reported non confident data may correct their analytical methods accounting for results of intercomparison runs and certified values of soil SRMs.

ANALYSIS OF GAMMA ACTIVITY OF HEAVY WATER AT RB REACTOR

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Abstract

The RB experimental nuclear reactor still works with heavy water obtained in 1959 from the former USSR. Gamma activity of heavy water was periodically controlled during the past time. In this experiment measurements were carried out with two samples: D_2O taken from RB reactor and D_2O that has never been used in reactor. Two germanium spectrometers were used as detectors. Gamma spectra's data were evaluated manually and using several computer codes. Results of the experiment show that gamma activity D_2O of RB reactor is at the level of background in "Vinča" Institute, without contamination with fission products.

Key words: Reactor RB, heavy water, gamma activity

1. Introduction

The RB zero-power, heavy water reactor started operation in 1958 with heavy water obtained in former USSR. That water was removed to RA reactor at the end of 1959, when the RA reactor was put in operation. New heavy water was obtained from USSR and it is still used in RB reactor, after 37 years of continual operation. This heavy water is controlled each year, and even more frequently if it is necessary. Control is related on isotope composition of heavy water, it's pH factor and it's possible contamination with fission products.

2. Experimental procedure

Gamma activity analysis of RB reactor heavy water was performed in 1995, for the first time on low-background (Pb shielded) germanium spectrometer. Using that measurement system, analyses of gamma background were carried out up to energies of 2.85 MeV with average integral count rate of 0.5 cps. Gamma background spectrum was also, as usual, measured using coaxial Ge detector with corresponding nuclear electronic modules, placed in experimental room of the RB reactor. These measurements were carried out in energy range between 50 keV and 1.85 MeV, with average integral count rate 16.167 ± 0.009 cps, because it has no special shielding from the background radiation.

For the first time, in order to verify results of measurements, compared gamma activity analyses were carried out with samples of D_2O taken from the RA reactor. That heavy water was bought in former USSR in late 1980's and never used in reactor.

Calibration gamma radioactive sources were used in energy range from 30 keV and 2.7 MeV for energy calibration and determination of the efficiency of the low-background Ge gamma spectrometer. These sources were specially made solutions of naturally gamma radioactive materials in water, placed in plastic bottles of 300 ml volume, "Cedevita" type. One of these calibration sources

contains 12.633 g of KCl solved in 300 ml distillate water, and it provides gamma line with energy 1460.8 keV from ^{40}K nuclide. Second calibration gamma source was made as a solution of 15.063 g $La(NO_3)_3 \times 6H_2O$ in 300 ml distillate water, and it provides gamma lines with energies 788 keV and 1436 keV from ^{138}La, as well as gamma lines with energies 154 keV, 270 keV, 351 keV, 403 keV and 832 keV from radioactive successors in ^{227}Th natural radioactivity chain.

Energy calibration and absolute efficiency curve were determined by using gamma lines and their absolute activities determined from pulse amplitude spectrum, collected in Canberra series-35 MCA with 100 MHz ADC and 8192 channel memory:

$$E_\gamma[keV] = 0.34999 \cdot ch + 8.711, \quad ch \in [1,8192] \tag{1}$$

The absolute activities of calibration sources were determined using known mass of radioactive nuclides in the solution. The absolute efficiency of the spectrometer in whole energy range was obtained by least square fitting method, through efficiency-energy points, as a hyperbolical function:

$$\varepsilon = \frac{4.65699}{E_\gamma(keV)^{1.04466}} \tag{2}$$

Samples of 300 ml of heavy water from RB and RA reactors were prepared on the same way as the calibration sources (in plastic bottles of 300 ml volume, "Cedevita" type) and analyzed by low-background Ge gamma spectrometer. Measurements of the samples of heavy water from RB reactor were carried out for 240000 s, with total integral of 135093 counts, that provides average integral count rate of 0.563 cps (i.e., at the gamma background level of the spectrometer). Measurements of the samples of heavy water from RA reactor were carried out for 96 000 s, with total integral of 55751 counts, which provides average integral count rate of 0.581 cps (i.e., again at the gamma background level of the spectrometer).

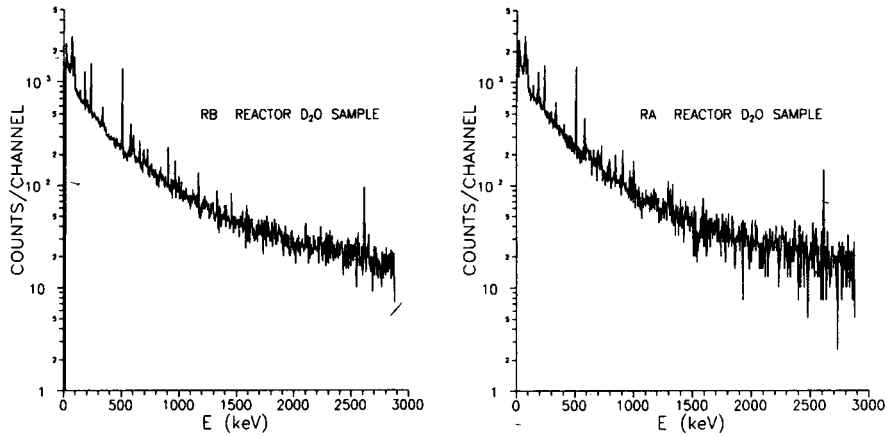
Germanium gamma spectrometer system, used in the experimental room of RB reactor, was energy and efficiency calibrated with several point sources on different distances from the detector axis. These calibrations were verified with four volume gamma radioactive sources in two energy points. These volume calibration sources were made as solutions of KCl in distillate water (with concentrations of 1 g K in 50 ml and 5 g K in 50 ml), and solutions of ^{137}Cs in distillate water with activity of 137.7 ± 5.5 kBq and 275 ± 11 kBq on 1995-03-29. Samples of heavy water from RB and RA reactor and calibrated solutions were made in plastic bottles of 54.0 ± 0.1 ml volume.

RB reactor's heavy water gamma activity analyses were carried out at 15 cm distance from the detector, during 231678 s, with total integral $3{,}7766 \cdot 10^6$ counts, which provides average integral count rate of 16.301 ± 0.009 cps, what is at the gamma background level of the spectrometer. Measurements of gamma activity of the samples of heavy water from RA reactor were carried out for 352449 s, with total integral of $5.6809 \cdot 10^6$ counts, providing average integral count rate of 16.118 ± 0.007 cps, (i.e., at the gamma background level of the spectrometer).

3. Results of measurements

Data from measured gamma spectra were evaluated manually, as well as with several computer codes. Three of used computer codes are commercial ones, GANAAS [1], MicroSAMPO [2] and APOGEE [3] from Canberra Inc., and the fourth one is ANA computer code [4] developed in Laboratory NET. Only representative gamma lines in measured spectra are given. The measured

amplitude gamma spectrums for both samples of heavy water (from RA and RB reactors), obtained at the low-background Ge spectrometer are shown at Figure 1. Results are evaluated at the same measuring time of gamma spectra: 240000 s.



a b

Figure 1. Gamma spectra of heavy water sample from RB reactor (a) and heavy water sample from RA reactor (b)

4. Conclusion

Gamma activity analysis of RB reactor heavy water, including comparations with gamma activity of sample of heavy water (obtained from RA reactor) never used in reactor, shows that its gamma activity is on the gamma background level in "Vinča" Institute, without contamination by fission products.

5. References

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[2] MicroSAMPO Version 1.2, User's Manual CISE 511, Canberra Inc., 1988
[3] APOGEE Version 1.3, Gamma Spectrum Analysis Code, Canberra Industries Inc., 1986
[4] Avdić S., Pešić M., "ANA, A Program for Aútomatic Gamma Spectrum Determination with Library of Nuclides Gamma Lines", IBK-NET-Vinča computer codes library, Vinča 1987

IN-SITU GAMMA SPECTROMETRY INTERCOMPARISON EXERCISE IN SALZBURG / AUSTRIA NUCLIDE SPECIFIC EXPOSURE RATE MEASUREMENTS

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ABSTRACT

In-situ gamma spectrometry has become a useful method to assess the nuclide concentration of artifical and natural gamma-emitters in the soil. With such measurements the determination of activity concentration in the soil can be calculated and also the nuclide specific exposure rates. So it will be possible to decide which radionuclide contributes the main part of the exposure rate in a given radiation field. For the quality assurance of the measurements periodically intercomparison exercises are essential. To meet this requirement exercises were organised in different European countries since 1989. The last exercise in September 1994 was organised in Salzburg / Austria. The participation of 24 measurement teams from all over Europe emphasises the importance of the inter comparison. Salzburg was selected because the Province of Salzburg / Austria was among the heaviest contaminated region outside the former USSR by the Chernobyl fallout. Two different typical sites were selected for the measurements: Site 1 was inside the urban area of Salzburg on intensively used agricultural land which had not been disturbed since the fallout. This site is representative for intensively used agricultural regions in the Province of Salzburg. Site 2 was in the mountainous region of the Tauern on elevated altitude of approx. 1700 m. This site represents the soil and contamination conditions of the Alpine region. Both sites differ significantly in terms of different soil distribution and contents of natural and artifical radionuclides. The main origin of the specific exposure rates in this regions depends from this facts. Results are presented.

INTRODUCTION

In-situ gamma-spectrometry has become an important method for the rapid detection of radio nuclides in the environment, since its development in the late sixties and at the beginning of the seventies. Behind the determination of activity concentrations in the soil also nuclide specific exposure rates can be calculated after such measurements. So it will be possible to decide which radio nuclide contributes the main part of the exposure rate in a given radiation field (1). The development of semiconductor detectors improved the usefulness of this method for the quantitative determination of natural and artificial radio nuclides as well. After the reactor accident in Chernobyl periodical inter comparison exercises applying in-situ gamma-spectrometry have been organised by different institutions of the EU member states and neighbouring countries in order to improve the scientific basis of the method, for information interchange of the scientific community and of increasing importance also to develop methods and to gain a good basis for quality assurance. In-situ gamma-spectrometry is used by an increasing number of radiation protection institutions, governmental and provincial environmental monitoring laboratories and authorities, health physicists of nuclear power plants and scientific institutions. The great importance is documented by the ever increasing number of participants at the in-situ inter comparison exercises in the past.

Considering the technical aspects of the method and its applicability for the determination of radioactive contamination and the nuclide specific exposure rates, the problems of the determination of activity concentrations are closely related with the age of the fallout, whereas the exposure rates estimates are much less sensitive to variations in radio nuclide distribution and soil characteristics and so very accurate estimates of individual nuclide contributions to the total external exposure rate can be made from the field spectra.

In-situ gamma spectrometry can be applied for the measurement of natural and for artificial gamma-ray emitters as well. The photon flux of the gamma-emitters in the soil and also the exposure rate is a function of soil parameters (density, humidity), the radio nuclide, of soil-depth distribution and the spatial variation of the fallout within the affected area.

METHODS, IMPLEMENTATION AND OBJECTIVES

The inter comparison exercise was implemented in the Province of Salzburg at two different sites. Three reasons determined the selection of these sites:

* The contamination by the Chernobyl fallout in the province of Salzburg was rather high varying between 10 kBq/m² and 80 kBq/m² at the time of the fallout deposition in April 86
* The contamination on site 1 / (Krauthuegel, city of Salzburg) was studied in detail (2).
* The two sites selected represent two different specific environments in the Province of Salzburg: Site 1 (Krauthuegel, city of Salzburg) is typical for low-lying land used intensively for agricultural purposes;

Site 2 (Nassfeldalm, Badgastein) represents mountainous environment with extensive seasonal usage for agriculture and the natural radioactivity of both sites varies significantly.

The specific task of this exercise was the calculation of the specific activities for natural and man-made gamma emitters and mainly for this work their contribution to the exposure rates.
The participants used HpGE-detectors exclusively (3 n-type and 21 p-type detectors), mounted 1m above ground, with relative efficiencies ranging from 10% to 54% of relative efficiency. At measurement site 1 13 positions and at measurement site 2 15 positions were predefined for gamma-spectrometric measurements. Each position was marked and numbered, on average between 6 and 8 measurements of approximately 1/2 hour acquisition time were made.

RESULTS AND DISCUSSION

The participating measurement groups were from different institutions encompassing environmental measurement laboratories, radiation protection institutes, research centres and university institutes as well. Though soil depth profiles for ^{137}Cs were distributed to the groups, only a few participants used this information as a basis for the calculation of the site-specific photon flux and thereby the concentration for $^{137,134}Cs$ in the soil [$Bq.m^{-2}$]. Not all participants informed the authors about the methods they used. Out of those who provided this information most of them used for the calculation assuming real distribution appropriate α/ρ factors (α = reciprocal of the relaxation length of the assumed exponentially distributed source activity with depth, cm^{-1}. ρ is the soil density, g/cm^3) for an exponential distribution derived by BECK (1) or factors recommended in the ICRU report (3). Only a few applied other methods including double exponential distribution, assuming Lorentz-distribution or calculation of the flux based on the experimentally derived depth distribution. For the photon flux calculation and thereby the specific activity of the natural radio nuclides [$Bq.kg^{-1}$] uniform distribution was assumed. The exposure rates were then calculated to take into account conversion factors from Beck (1) [eg. $nSv.h^{-1}/Bq.m^{-2}(kg^{-1})$].
The results of the calculation for the exposure rates of $^{137,134}Cs$, ^{226}Ra, ^{232}Th ^{40}K and the total exposure rates for the two measurement sites are shown in Table 1 and 2 and on Figure 1 and 2.

Table 1: Results of exposure rate measurements, Krauthuegel, Salzburg [$nSv.h^{-1}$]

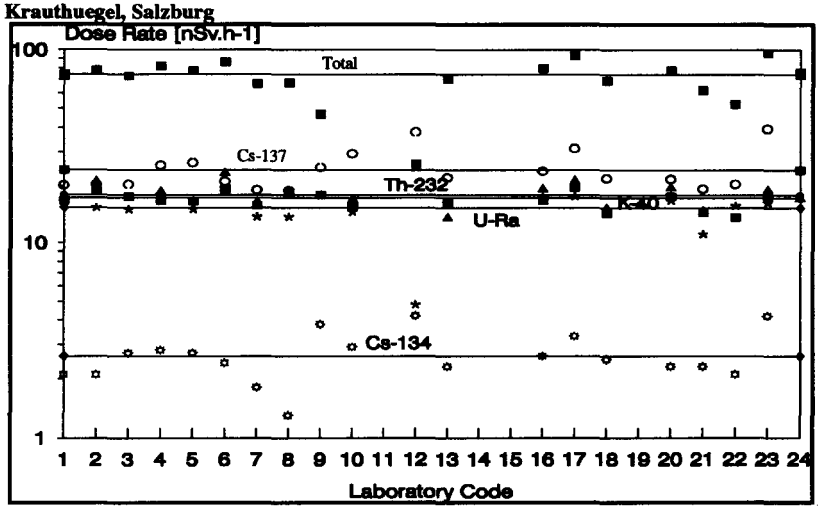
Statistical parameter	Uran-Radium Series	Thorium Series	^{40}K	^{137}Cs	^{134}Cs	Total
Mean	15,3	17,9	17,3	24,3	2,6	77,4
[%]-Total	19,7	23,1	22,4	31,4	3,4	100
SD	4,2	4,8	3,2	6,1	0,8	12,4
SD(%)	27,5	26,8	18,5	25,1	30,8	16,7
Number n	18	18	19	19	19	18
Min	13,7	14,8	11,9	20,4	2,0	73,1
Max	18,0	20,9	20,4	28,2	3,1	79,2

Table 2: Results of exposure rate measurements, Nassfeldalm, Badgastein [$nSv.h^{-1}$]

Statistical parameter	Uran-Radium Series	Thorium Series	^{40}K	^{137}Cs	^{134}Cs	Total
Mean	19,4	16,3	16,9	51,4	5,2	109,2
[%]-Total	17,7	14,9	15,5	47,1	4,7	100
SD	8,0	4,8	2,5	7,6	0,4	13,7
SD(%)	41,2	29,4	14,8	14,8	7,7	12,8
Number n	18	18	19	19	19	18
Min	14,5	13,0	15,0	43,8	4,2	93,3
Max	22,3	19,6	19,5	57,1	6,0	117,2

For each group a mean value was calculated from the results obtained from the different measurement positions of that measurement site. The mean values of each laboratory for a measurement site are the data-basis for the calculation of the statistical parameters given in Table 1 and 2. The inter comparison shows that the most of the laboratories got acceptable results, some of the participants have to spend more work for the calibration of their devices. As an interesting result of the inter comparison we found, that in Salzburg and a little bit more in Badgastein the dose rates of the artificial radio nuclides are in the same range as the natural radio nuclides (without cosmic radiation) !

Krauthuegel, Salzburg



Nassfeldalm,Badgastein

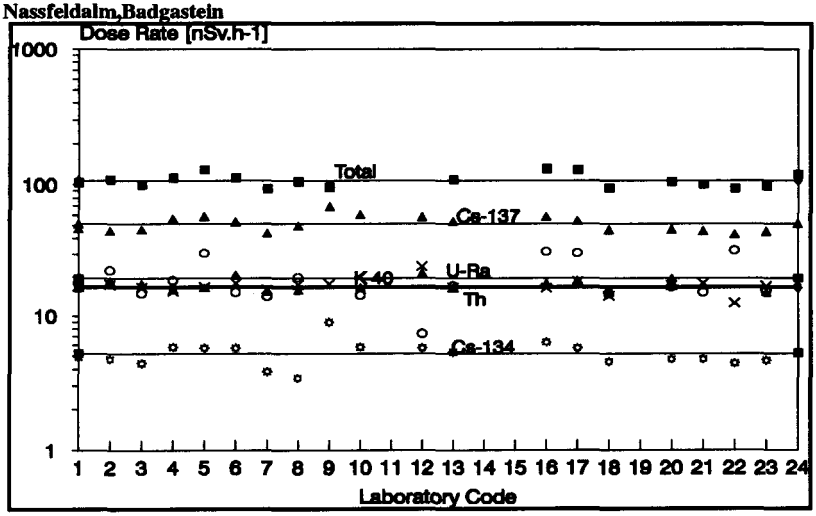


Figure 1 and 2: Mean values for dose rates, calculated from all points measured from each group for the measurements sites. Mean values are shown as horizontal lines.

CONCLUSION

The periodical organisation of inter comparison exercises will be an acceptable basis for quality assurance by providing a unique opportunity for the participants to check the reliability of their calibration and calculations, which will be most important for the control and safeguards of nuclear power plants and official authorities.

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3. ICRU 53. International Commission on Radiation Units and Measurements. Gamma-ray Spectrometry in the Environment. ICRU Report 53, Bethesda, Maryland, 1994.

RADIOECOLOGY OF VARDAR RIVER BASIN

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ABSTRACT

Beside the determination of the distribution coefficients for individual elemnts of interest, radiological measurement including total alfa, total beta as well as the specific activity for both artificial (H-3, Cs-134, Cs-137) and natural (Uranium and Actinium series:Pb-212, Pb-214, Tl-208, Bi-214, Ac-228, Rn-222 and K-40) radioisotopes presents in different environmental (water, soil, food and atmosphere)samples along the Vardar river, were performed.

Taking into consideration the results obtained so far and by application of adequate radioecological model, a global view on radioactivity of the investigated area, its origin, pathway of transport and places of accumulation as well as their influence on the total irradiation of the population have been evaluated and partly presented in this paper.

INTRODUCTION

Various assignment of the surface and underground water flows of the Vardar river (water supply, irrigation etc.) to which gravitate cca 2/3 of the total population of the Republic of Macedonia /see Fig.1/,are subject of particular interest regarding their protection and rational usage.

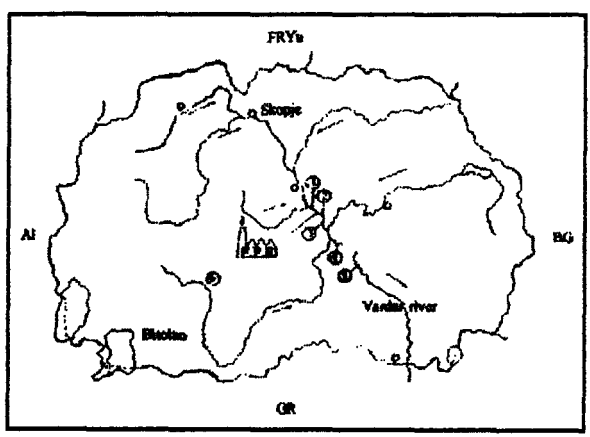

Fig.1 Location of sampling pints along the Vardar river basin
1-T.Veles, 2-Zgropolci, 3-Nogaevci, 4-Ulanci,5-Krivolak,
Pelagonija, FPF-factory for phosph. fertillizers

Parallel to the other investigations related with the determination of the bio-chemical pollution of the Vardar river basin (1), radioecological investigations took a significant attention, too. In this sense,several profiles along the Vardar river had been selected and collection of various samples on a regular base were performed.

In the very begining, starting with the determination of the distribution coefficients for individual elements in various components of the Vardar River Basin(2),showing the place of their accumulation, radiological measurements /determination of the total - α and total- β as well as the specific activitiy of individual natural (Pb-212, Pb-214, Tl-208, Bi-214, Ac-228, Rn-222 and K-40) and artificial (Cs-137, Cs-134 and H-3) radioisotopes/ in different environmental (water, soil, food, and atmosphere) samples along the Vardar river , were also introduced, identifing on such way the samples and locations with the increased level of radioactivity.Different the total -α and total -β measurements which were done on Canberra - 2200 ,anticoincident gas proportional counting system, specific activity for individual radioisotopes were performed on Tricarb , M 3320, Packard instrument, β Liquid Scintillation Counter or Canberra γ – spectrometer suupplied with MCA - 40 and high resolution semiconducting detector.(3).

These investigations gained in their importance specially after the Chernobyl accident.This was a period when an increased environmental radioactive contamination has been registerd particularly in the regions which are known to have received the highest amounts of precipitations during the period following the Chernobyl release(4).

By application of an adequite mathematically supported radioecological model (5-7) and taking into the consideration all relevant parameters (human habits, food chain etc.) ,the annual effective equivalent dose received (through ingestion , the most important pathway) by the local population, was calculated.

RESULTS AND DISCUSSION

While the values of the total-α activity in the investigated water samples from the Vardar river, for the last 10 years, were in the range between 0.02 and 0.4 Bq/L, the other samples showed values increased up to 63,7 Bq/kg in case of cultivated soil, with a maximum value following the period of Chernobyl release /26th of April,1996/, common for the other radiological measurements, too.

Values for the Total - β activity for Vardar river water and other samples are registered to be below 2.76 Bq/L and 5210 Bq/kg, respectively, with K-40 as a dominating radionuclide.For an example, on Fig.2 , the values for Total - β activity in water samples from the Vardar river /profile - Krivolak,which

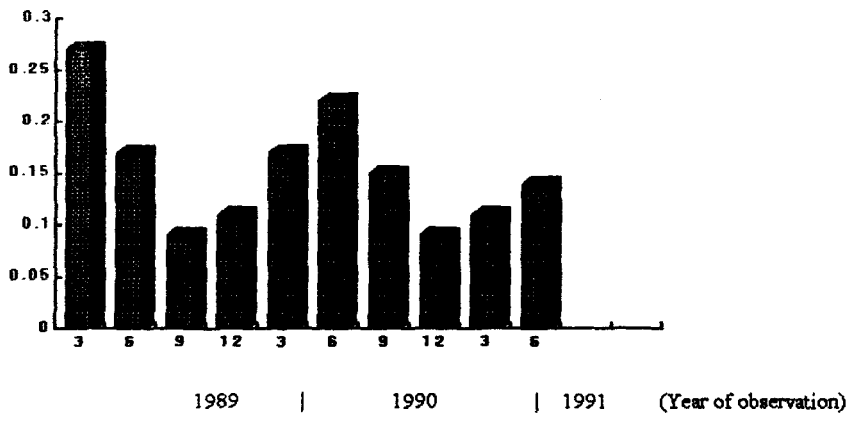

Figure 2. Total –β activity in water samples from the Vardar river /profile-Krivolak/

together with the previous up stream profile -Nogaevci,showed an increased activity/ with time is presented.It is evident that the changing of the total - β activity is following the shape of sinusoidal oscillation.Although this function is periodical, some characteristics of damped oscillations due to the decreasing of the amplitude with time, are evident.This can be explained by washing out of the contaminated soil during the melting period of snow /March - June/ .

Different the tritium concentration in Vardar river water samples showing values between 5 and 15 T.U.[1] falling down to the level of the period before 1952 (starting period of thermonuclear explosions), the valus for Pb - 214 and Ac- 228 are increased specially at Nogaevci and Krivolak profiles. In particularly the concentration of Pb-214 in the sediment samples during the 1989-1991 period is reaching the value 125 Bq/kg which is several times higher in comparison with the previous observations.The reason for such an increasing of the mentioned activities has to be looked and related with the activities of the factory for phosphatic fertillizers, located upstream of these profiles.

Analysing the results obtained by the determinatin of the Cs-isotopes,it has been found out that the concentration of Cs-134 in comparison with this of Cs-137 during the period of increased contamination i.e. after the Chernobyl release,was twice less and disappeared earlier due to its shorter half life time.The concentration of Cs-137 in water samples from Wardar river for longer period of observation(`86-`95) showed a tendency of faster activity decreasing comared with the natural decay, themost probably, due to the presence of other comparative processes of dissolving,extraction,diffusion etc, approaching the level of early eighties (5).

Different the specific activity of the observed radionuclides (Cs-137, Pb-212, Pb-214, Tl-208, Bi-214 and Ac-228) in cultivated and uncultivated soils which showed values between 10 and 70 Bq/kg, various food samples had values below 25 Bq/kg or L, see Figure 3.

Figure 3.Average specific activity for the observed radionuclides in various food samples for the period `89 - `91

It is worth to mentioned the milk samples whose values were not higher than 4 Bq/L.This is of particular interest for the radiation burdening of children, dominantely consuming this product.

On the basis of the above mentioned results and by application of an adequite mathematically supported radioecological model (6-8),taking into the account all relevant parameters(human habits, fodd chain etc.), the anual effective equivalent dose received through ingestion(the most important pathway of the human irradiation) was calculated for the period `89-`91 and the value of 1.5mSv/y was obtained, while the Cs-137 contribution significantly decreases in comparison with the early period following the Chernobyl accident(5), and was not higher than 60 μSv/y .

[1] One tritium unit , 1 T.U.=H^3 - atom/$10^{18}H^1$ - atoms

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Mini-Presentation

PAPER TITLE: REMOTE METHODS OF RADIATION MEASUREMENTS AND SEARCH OF LOCAL RADIOACTIVE SOURCES IN THE CHERNOBYL AREA

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CODE: 119034 CITY: Moscow COUNTRY: Russia

MAJOR SCIENTIFIC TOPIC NUMBER: 4.1. Measurements

ABSTRACT:

It is the common knowledge that liquidation of Chernobyl accident consequences was carried out with significant losses: personnel had been exposed by great dose of irradiation, in most cases unwarrantedly great. This could be explained by deficiency of experience in accident consequence liquidation of such scale. Now we have got such experience. Huge amount of experimental data has been collected, a number of reports regarding Chernobyl accident consequences liquidation have been issued. But this experience must be generalised, main problems should be realised, and steps for their solution should be defined.

One of these problems is absence of equipment which would be specially designed for situations like Chernobyl accident. It seems to us that apart from everything else it is necessary the equipment for solution of three following tasks:
1. Estimation of radiation situation at various objects.
2. The search of local sources of ionising radiation in hardly accessible places.
3. Defining of radioactive areas in the country and measurements of surface radioactivity.

These three tasks must be solved at minimal personnel irradiation. This could be reached with help of remote controlled robot systems or by measurement of radiation effect at sufficiently large distance from the object, when parameters of radiation field are calculated according to definite algorithm.

Possible ways of such equipment designing in application to all three tasks are considered in presented paper. The search of local sources of ionising radiation in hardly accessible places with help of special measuring equipment placed on remote controlled vehicle (in other words - mobile robot) is considered in detail.

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PAPER TITLE Automatic station of environmental control

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

Abstract. A prototype of an automatic station of environmental monitoring has been designed and manufactured. In the design advanced technologies tested at NPO Energia when developing space technology were used. The general view of the station is presented in the Figure.

The station provides monitoring of water and atmosphere parameters, can be set stationarily or transported by telephone cables or a radio channel (including a satellite as well).

The station power supply is provided from the electric mains and in distant and difficult of access regions from an off-lain solar array wind facility or micro HEPP.

An equipment complex of the station has been developed and manufactured to measure the concentration of harmful gases in the atmosphere, the chemical composition of aerosols, to provide a detailed radiation analysis.

The equipment characteristics are given, the concept of using stations in large cities and around ecologically dangerous productions is discussed.

Key words: environmental monitoring, automatic station, monitoring of atmosphere, monitoring.

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PAPER TITLE Apparatus for radio-ecological control of rooms condition

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

Among ecological problems the task of performing the monitoring of environment radiation parameters is of great importance. In the given report the experience concerning the development of equipment intended for carrying out the complex inspection monitoring of state of both dwellings and industrial rooms including the monitoring of RADON content in the air is presented.

In accordance with the conception of radiation safety being adopted in the Russian Federation, the versions of equipment "REKS" and "REKS-alpha" have been developed. The equipment provides the conduction of inspection monitoring of external radiation dose equivalent rate, density of beta radiation flux from surfaces and RADON concentration in the air according to the value of "latent" energy.

The equipment versions differ by the method of measuring the "latent" energy value and by additional capabilities. In "REKS" version the procedure of measuring the beta activity of the air sample being taken at filter is used. The removable sensor being included into "REKS" composition permits to perform the specific activity measurements of liquid and loose samples (calibration is carried out by Cs-137). In "REKS-alpha" version the procedure of measuring the sample alpha-activity is used. The removable sensor being included into "REKS-alpha" composition permits to perform the measurements of the radon concentration, being averaged over the exposure time, with the use of carbon adsorbers. "REKS-alpha" version is an equipment which corresponds by its metrological parameters to the best similar equipment being known in the world.

PC CREAM: A PC PACKAGE TO ASSESS THE CONSEQUENCES OF RADIOACTIVE DISCHARGES DUE TO NORMAL OPERATIONS

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INTRODUCTION

PC CREAM is a Windows™ based software package which implements CREAM (**C**onsequences of **R**eleases to the **E**nvironment: **A**ssessment **M**ethodology). CREAM was developed to assess the consequence of radioactive discharges as a result of normal operations, a description of the methodology was published by the European Commission (EC) (1). The methodology consists of a series of interlinked models which describe the transfer of radionuclides through the atmospheric, terrestrial and aquatic environments, the pathways by which people may be exposed to radiation and the resulting health detriment. The package contains models for the following processes; atmospheric dispersion; external irradiation from activity in air; transfer through foodchains; resuspension of deposited activity; external irradiation from deposited activity; dispersion in the marine environment; discharge to river systems; external irradiation from both marine and freshwater sediments. PC CREAM has been developed to calculate radiation exposure to individuals as well as the collective dose to specific population groups.

SYSTEM REQUIREMENTS

PC CREAM has been designed to run on a Personal Computer with Microsoft Windows version 3.1 or later, 4 Mb of RAM, a 3½ inch floppy drive and a mouse.

PACKAGE DESCRIPTION

PC CREAM consists of a main assessment package CREAM, with a number of supporting interlinked models, as illustrated in Figure 1. The supporting models can be used alone to provide intermediate data such as activity concentrations in air or activity concentrations in marine and terrestrial foodstuffs, or to supplement the default data sets supplied with the assessment package. A brief description of each of the supporting models is given below.

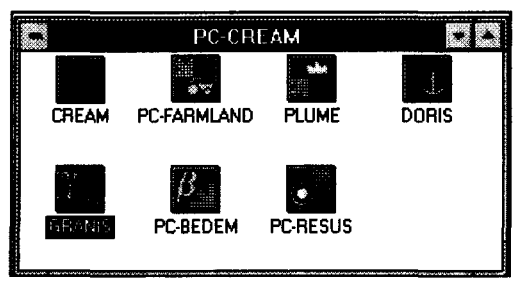

Figure 1 The PC CREAM program group.

- PC FARMLAND - A terrestrial foodchain model capable of modelling radionuclide transfer into foods from a continuous deposition from atmosphere. The output being activity concentrations in the following foods: Cow meat, liver and milk, Sheep meat and liver, leafy green vegetables, root vegetables and grain products.

- PLUME - A Gaussian plume atmospheric dispersion model incorporating models for deposition and external gamma exposure from activity concentrations in air.

- DORIS- A marine dispersion model for European waters capable of calculating activity concentrations in seawater, seafood and marine sediments.

- GRANIS- A model for calculating external gamma doses from deposited activity in a number of media.

- PC BEDEM - A model for calculating external beta doses from deposited activity.

- PC RESUS - A time dependent resuspension model for calculating activity concentrations in air due to the resuspension of previously deposited activity.

Discharges to river systems are intrinsically site specific, for that reason the two river models in PC CREAM,

a simple dilution model, and a more complex dynamic river model, are contained entirely within the main assessment package (CREAM).

THE ASSESSMENT PACKAGE CREAM

The assessment package follows many of the conventions of other Windows™ based applications, making use of familiar features such as buttons, check boxes and list boxes. The package has been design to lead the user through an assessment in a logical and straightforward manner, the welcome screen is illustrated in Figure 2. The assessor initially selects the site details such as type of facility and location. The location, population distribution and agricultural production distribution for the major European Union sites are held within CREAM for the purpose of collective dose calculations. Assessments may be undertaken for discharges to either the atmospheric or aquatic environments, or more comprehensively, releases to both environments may be considered. Site and assessment specific details are recorded as the user progresses through the data entry screens. An example data entry screen is illustrated in Figure 3. Having entered such data as radionuclide discharge rates, the location and age group of the exposed individuals, the assessor can select from a comprehensive list of exposure pathways. When all the relevant data has been entered and selected the assessment can be run. The calculated doses are written to a file or may be displayed within CREAM in a tabular or graphical format. An example pie chart is presented in Figure 4.

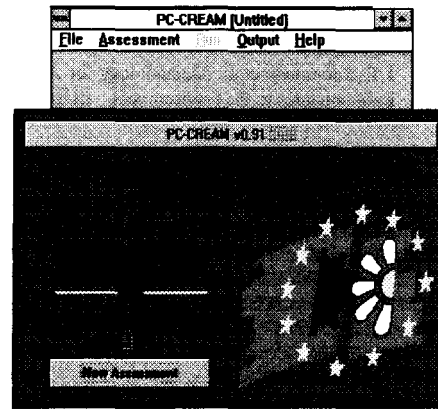

Figure 2 CREAM welcome screen

At all points throughout the assessment a comprehensive online help facility is available.

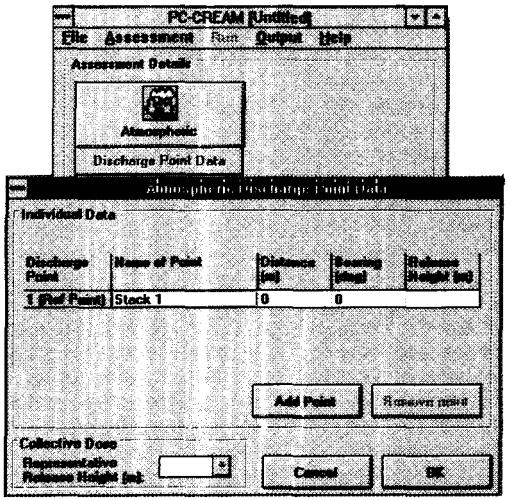

Figure 3 Atmospheric discharge point data entry screen

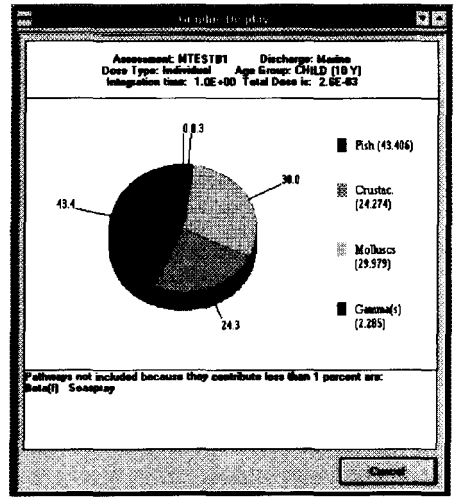

Figure 4 Graphical output from CREAM

SUMMARY

PC CREAM can be used to assess the radiological impact of releases from virtually any type of installation including nuclear power plants and nuclear fuel cycle facilities. It is envisaged that one of the main uses of the package

will be for prospective assessments of the consequences of proposed new plants, or for discharge authorisations. The development of PC CREAM has been carried out under contract to the EC.

The PC CREAM screens presented in the figures of this compact were taken during the development stage of the package, there may be slight differences in the final version.

REFERENCES

1. J R Simmonds et al, Methodology for Assessing the Radiological Consequences of Routine Releases of Radionuclides to the Environment. EUR 15760 EN (1995) (Luxembourg, OOPEC).

PROBABILITY SAFETY ASSESMENT (PAS-LEVEL 3) AND EMERGENCY PLANNING

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INTRODUCTION

To assess the radiological and health consequences of hypothetical accidents of Paks NPP the PC COSYMA program (1) was used. Probabilistic runs with cyclic sampling (using 144 samples) have been carried out. In the calculations 16 sectors (22.5° each) and 10 distance bands (from 0.4 to 450 km) were used. Detailed calculations were performed on the border of an exclusion zone of 3 km around the reactor. Whenever the estimated maximum value of early dose exceeded the limits of countermeasure criteria, the impact of protective measures on the early and late health effects have also been calculated. The code has been previously adapted to Hungarian circumstances (using site specific data on meteorology and population). The DATA library of PC COSYMA is filled with actual data on agricultural productivity using a grid of 10 km x 10 km taken from the Statistical Yearbook of Hungary (2).

CALCULATIONS ON DBA EVENTS

The consequences for the population of Hungary of the above postulated accident scenarios from early (7 days) and chronic (50 years) exposures have been estimated. The calculated doses and health effects are presented in Table 1. for the smaller (Ø 73 (A), Ø 93 (B), Ø 111 (C), Ø 233 (D)) and larger (Ø 492 in cold leg (E), Ø 492 in the hot leg (F), steamline break (G), collector cover opening (I)) accidents.

At the events listed above in case of estimation of radiological and health consequences not any protective measures were taken into account, since even in the worst case (results in parentheses in Table 1): Pasquill category F, and rainy weather, the calculated doses were much lower than the dose criteria given for any countermeasure.

Table 1 Calculated radiological and health consequences for postulated accidents at the border of exclusion area of Paks NPP

Event	Doses, 90th percentile (max)		Health effects, 90th percentile (max)	
	Early [Sv]	Collective committed [manSv]	Early [Number of total mort.]	Late [Number of total mort].
A	3.6E-14 (1.4E-12)	4.3E-09 (9.1E-09)	-- --	-- (4.1E-10)
B	6.0E-14 (2.4E-12)	1.8E-08 (2.4E-08)	-- --	-- (1.1E-09)
C	1.0E-13 (4.0E-12)	3.2E-08 (4.2E-08)	-- --	-- (1.9E-09)
D	5.6E-13 (2.1E-11)	1.6E-07 (2.2E-07)	-- --	-- (1.0E-08)
E	8.3E-09 (8.5E-07)	7.0E-03 (1.1E-02)	-- --	-- (5.5E-04)
F	5.2E-09 (3.5E-07)	5.4E-03 (9.9E-03)	-- --	-- (5.0E-04)
G	1.3E-07 (1.3E-05)	4.2E-02 (2.9E-01)	-- --	-- (1.4E-02)
I	5.1E-06 (5.3E-04)	1.31 (3.8)	-- --	-- (1.8E-01)

EARLY EFFECTS

Deterministic runs for the critical direction (from the radiological point of view) of the plume have been made and compared with the results of probabilistic runs with cyclic sampling. Detailed calculations have been carried out at two preselected distances (at the border of exclusion area and at the center of Paks town (5 km)). At the calculations of dose contributions from cloudshine, groundshine, inhalation and resuspension an integration time of 7 days was taken into account. The dose contributions from various pathways and organs at the border of exclusion area are presented in Table 2.

Table 2 Mean individual 7 day doses at 3 km

		Percentage			
Organ	Dose (Sv)	Cloud	Ground	Inhalation	Resuspen.
Lung	1.3E-04	2	38	59	1
Thyroid	1.5E-03	0	4	93	3
Eye lens	5.7E-05	6	94	0	0
Ovaries	6.0E-05	3	74	22	1
Effective	1.6E-04	2	29	67	2
B. Marrow	7.4E-05	3	64	33	0
GI-Tract	1.7E-04	1	25	73	1

The number of people and the magnitude of land [km^2] affected by various protective measures are presented in Table3.

Table 3 The number of people and the magnitude of land affected by countermeasures

Countermeasures	Number of people affected		Magnitude of affected (km^2)	
	Maximum	Mean	Maximum	Mean
Evacuation	9.6E+02	3.8E+01	5.3E-00	5.4E-01
Sheltering	6.4E+03	6.8E+02	4.7E+01	6.6E+00
Stable-iodine	1.0E+03	7.8E+01	5.3E+00	1.0E+00

Respecting that the expected doses - even in the most pessimistic deterministic case (where the maximum values are encountered) - will be much lower than the threshold for deterministic health effects, no early morbidities or mortalities are detected in our calculations.

LATE EFFECTS

Latent dose and health effect calculations are made for bone marrow, bone surface, breast, lung, stomach, colon, liver, pancreas, thyroid, gonads, which are the organs most important from the radiological point of view. Calculated doses are given also by pathway and by organ and for doses up to 50 years (this last involves cloudshine, groundshine, inhalation, resuspension) at the border of the exclusion area in Table 4.

Table 4 Late dose contribution by organs and by pathway at severe accident

Organ	Percentage			
	Dose (Sv)	Ground	Inhalation	Resuspen.
B. Marrow	1.237E-03	67	31	2
B. Surface	1.577E-03	56	41	3
Breast	9.623E-04	88	11	1
Lung	1.072E-03	82	17	1
Stomach	9.162E-04	85	13	1
Colon	1.012E-03	74	24	1
Liver	9.152E-04	85	13	1
Pancreas	8.484E-04	85	14	1
Thyroid	4.227E-03	23	71	6
Gonads	9.350E-04	86	13	1
Remainder	1.011E-03	86	13	1
Effective	1.192E-03	71	27	2

The maximum value (119) is due to the weather sequence 62, where the meteorological parameters are very unfavourable from the radiological point of view (e.g. Pasquill F category, dry weather, wind direction is toward Budapest, and heavy rain, when the plume reaches the highly populated areas of Budapest). Fortunately, these occurrences (where the population maximum and the contamination maximum is in coincidence) are very rare. The contributions to the total late mortalities by organs are presented in Table 5.

Table 5 Number of total late mortalities (contributions by organs)

Organ	Mean	Maximum
Bone Marrow	0.30	12.6
Bone surface	0.01	0.4
Brest	0.43	19.3
Lung	0.52	22.7
Stomach	0.45	20.3
Liver	0.17	7.5
Colon	0.23	10.5
Pancreas	0.24	10.9
Thyroid	0.17	5.2
Remainder	0.21	9.6
TOTAL	**2.73**	**119**

CONCLUSIONS

In calculation on DBA events doses are below the countermeasure reference levels even if the steam generator collector cover opens (spiking assumed). As a consequence of calculated low doses there are no early fatalities and even number of late fatalities is always less than one. In case of severe accidents no early morbidities or mortalities are detected in our calculations. Even the maximum of late effects are negligable (120) in comparison the natural cancer mortality. (New patients with malignant neoplasm was 23,900 in 1993 (2)).

REFERENCE

[1] J. A. Jones, P. A. Mansfild, S. M. Haywood, A. F. Nisbet, I. Hasemann, C. Steinhauer and J. Ehrhard: PC COSYMA: An Accident Consequence Assessment Package for Use on PC, Report EUR 14916.

[2] Statistical Yearbook of Hungary 1993.

INTERACTIVE SIMULATION OF ENVIRONMENTAL CONSEQUENCIES OF NUCLEAR ACCIDENTS

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INTRODUCTION

A program system has been developed to follow the consequences of radioactive releases of a (hypotetical) nuclear accident. Atmospheric dispersion, plume depletion by dry-out and wash-out, cloudshine and groundshine doses, dose commitments from inhalation and ingestion, early and late health effects are computed. Effects of the introduction of countermeasures are taken into account.

Actual values of all major quantities (from core inventory to countermeasure criteria) can be specified quite arbitrarily by the user, however, defaults are offered for all data. Meteorological conditions can be user-specified (and refreshed by every step) or selected automatically from a library containing data observed for one year (8760 hours).

MODELING GOALS

A towfold task is fulfilled by the program:

- it is a tool for education: it helps the training of those people who are responsible for introducing countermeasures;
- in case of a real accident: the program gives fast, easy-to-understand visual information on the radiological state of the environment and on expected consequences.

SIMULATOR FEATURE

The program works as a simulator: various scenarios can be specified (different emissions, meteorological conditions, countermeasures, etc) and results obtained by these scenarios can directly be compared to each other. Computations are carried out in one hour steps upto a maximum of 48 hours, meteorological and emission parameters can be newly set and results are given for analysis at each step. There is a possibility to step back to earlier phases and repeat the simulation under different conditions. Simulation is terminated by user request, or at 48 hours after the very beginning, or when the total plume leaves the border of the country (or other preset region of interest).

SOURCE MODELING

The emission of 16 isotopes are followed. There is a built-in inventory, present default values refer to a medium burned core of a 440 MW VVER reactor. (At the beginning of each run the user can change the defaults for his/her actual inventory). Release rates (i.e. fractions of the inventory released to the environment in the given hour) can be specified either to isotope groups (e.g. noble gases), or to each isotope individually. The dispersion of the iodine isotopes emitted in vapour form is followed separately.

Building dimensions, release height and the release of sensible heat are also needed, the latter being used for buoyancy calculations.

A maximum of four (not necessarily adjoining) emission phases can be defined within one run.

ATMOSPHERIC DISPERSION

Each hour's emission is cut into 9 puffs, each carrying one-ninth of the emission. Since meteorological data (wind direction and velocity, precipitation and stability class) are changed at one-hour steps, the 9 puffs travel always parallel to each other. A Gaussian dispersion of the puffs is assumed, with total reflection at the ground and at the inversion layer, the dispersion parameters are corrected with surface roughness). Deposition (both dry and wet) is computed for 16 wind direction sectors and 34 rings, i.e. altogether for about 500 segments.
Puffs are depleted at the end of every hour according to deposition and radioactive decay.

DOSE CALCULATION

External gamma air kerma rates, organ doses and effective doses are computed for all segments due to cloudshine and groundshine. Skin beta doses are also determined.

Inhalation doses are calculated from the clouds passing and from resuspension. Foodchain pathway is included in the computation of the committed doses.

Effects of shieldings by housing are taken into account according to user-specified occupancy and shielding factors.

In addition to the individual doses, collective doses are derived by taking into account the population data (for several doses, adult and child doses are calculated separately.

HEALTH EFFECTS

Early morbidities and mortalities (fatalities) are estimated from deterministic health effect models. Late effect probability calculations are based on the no-threshold, linear dose-effect relationship hypothesis.

COUNTERMEASURES

Three types of short-term countermeasures can be implemented during the simulation: sheltering, iodine prophylaxis and evacuation). Warnings are given to introduce these countermeasures, whenever preset criteria are reached or exceeded for one, ore more of the segments. The actual decision (for the implementation) is for the user. Or, and this is a unique advantage of the interactive simulator, the user can execute two parallel runs — with and without countermeasures — and compare the consequences.

For the long term countermeasures (relocation and food ban: which are to be introduced after finishing the direct simulation but whose introduction affect the dose commitments) actions are assumed to be taken automatically if the preset levels are exceeded.

Two types of countermeasure scenarios can be introduced by the user: intervention criteria may be based on either activity concentration or dose type quantities.

COMPUTATIONAL ENVIRONMENT - DATA FILES

Since the program is intended to serve education and to present quick information for laymen, one of our main purposes was that it had to be easy-to-handle and results had to be presented in an easy-to-understand, demonstrative manner. To fulfill these tasks the program works under WINDOWS operation system with the usual menu system, with dialog and message boxes. Results (path of the plume, contaminated area, doses, health effects) are displayed on maps with spectacular symbols, and/or in tables showing the computed values.

In the version first developed and presented the emission point is taken to the site of the Paks nuclear power plant. The maps, four maps with increasing magnification, cover the area of Hungary. All the site-specific data bases are relevant to Hungary (population and food consumption data) or even to the Paks site/reactor (meteorology, core inventory, building dimensions). The segmented structure of the program, however, makes the adaptation for another site relatively easy, assuming that the required data relevant to that site are available.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE ASTRAL : A SOFTWARE FOR THE ESTIMATION OF ACCIDENTAL RELEASES OF RADIONUCLIDES IN THE ENVIRONMENT

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Although every precaution is taken about the use of nuclear energy, the occurrence of a major accident cannot be totally ruled out. The Institute of Protection and Nuclear Safety set up a Technical Crisis Centre to face such an eventuality. This centre is often tested through crisis exercises. In order to meet the needs in environmental consequences assessment of nuclear accidents a Software called « ASTRAL » (technical assistance in post-accidental radioprotection) is under development. It allows to : characterise the radioactive state of the environment, forecast its evolution in time and space, estimate the efficacy of rehabilitation actions and countermeasures and compare the calculated values with regulatory levels.

The ASTRAL software is based on the following elements :

- A **data bank** containing information such as :
  - population, land use, characteristics of nuclear facilities, potential radionuclides releases, counter-measures
  - computation parameters such as transfer factors, radiological data...,
  - specific crisis data.
- A **cartographic module** based on a Geographical Information System, used to visualise the concerned zone, extract data, and display results,
- a **calculation module** integrating dynamic mathematical models, especially developed or coming from the literature (ECOSYS, AIEA...),
- tools for the **graphical presentation** of results,
- an **ergonomic interface** Machine - Man in order to manage the system.

Special attention has been given to the simplicity of use in the case of a crisis and to the minimisation of the risk of errors.

COMPUTERIZED SYSTEM TO CREATE RADIATIONAL AND HYGIENIC PASSPORT OF SITE AND TO RESEARCH THE PROGNOSTICATED CONSEQUENCE SENSITIVITY DUE TO COUNTERMEASURES

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Information system "Rahypnap" consists of data presented as a radiation-hygienic passport of settlement, that contains information on demography, nutrition regimes and radiation conditions. Initial data preliminary is collected in off-site sanitary and hygienic offices and written into special paper forms of passport. It includes 19 different forms according to data kinds. Special screen form was created to input, correct and view data in computer for each paper form of passport. Data base also contains tables of transforming coefficients from surface contamination density and food-stuff contamination density to annual individual effective dose. Those coefficients can be modified according to supposed countermeasures.

System's data base permits mean individual dose computations in 5 population groups of different age and for adults (age of 18-59), beside that, in 6 professional groups for urban and rural settlements apart; it also permits the collective dose computations for the distinct settlements and their compositions.

Dose computing procedure is based on the linearization of individual absorbed dose in a finite time period presented as a function of radionuclide contamination density of environmental objects (surfaces, food-stuffs).

System's interface is realized as a menu set that is structured as a tree. It provides an object selection: passport data forms, standard viewing and printing forms, up to 5 user defined forms and calculations. Corresponding set of operations is provided for each object: data input and correction (for passport data forms), data view and data print (for all objects), form design (for user defined forms), initial data actualization, coefficients set correction and performing (for calculations). All operations with data base are performed for preliminary selected region or single settlement only.

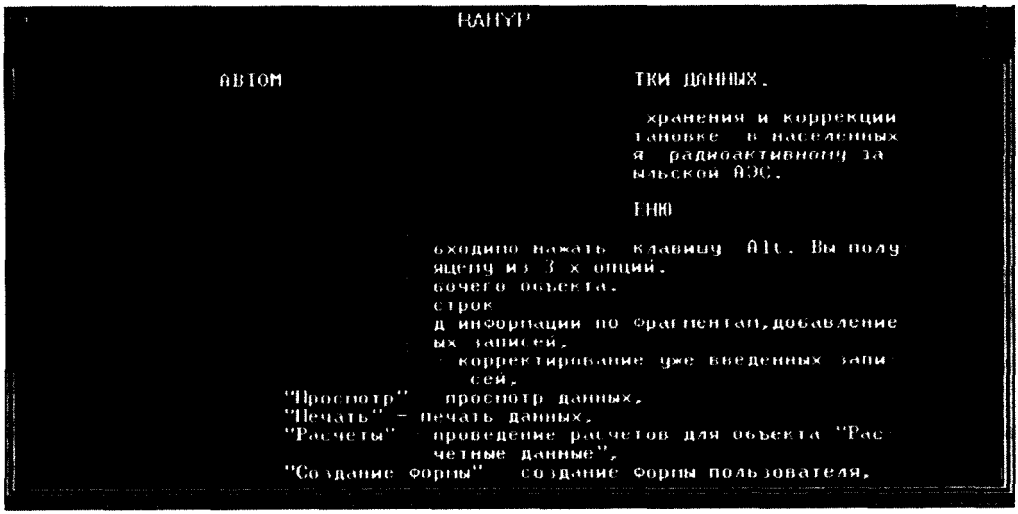

Figure 1. Sample of standart output form selection

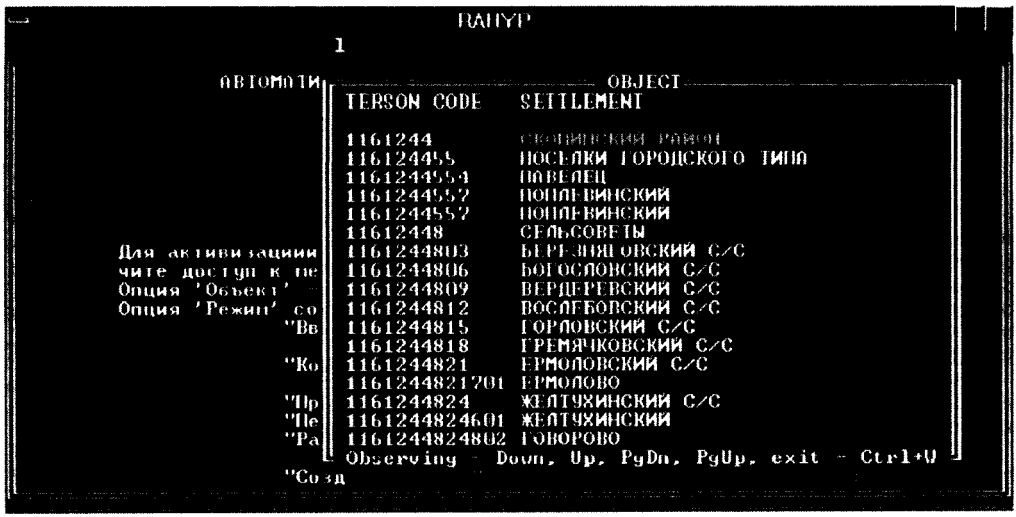

Figure 2. Sample of region selection

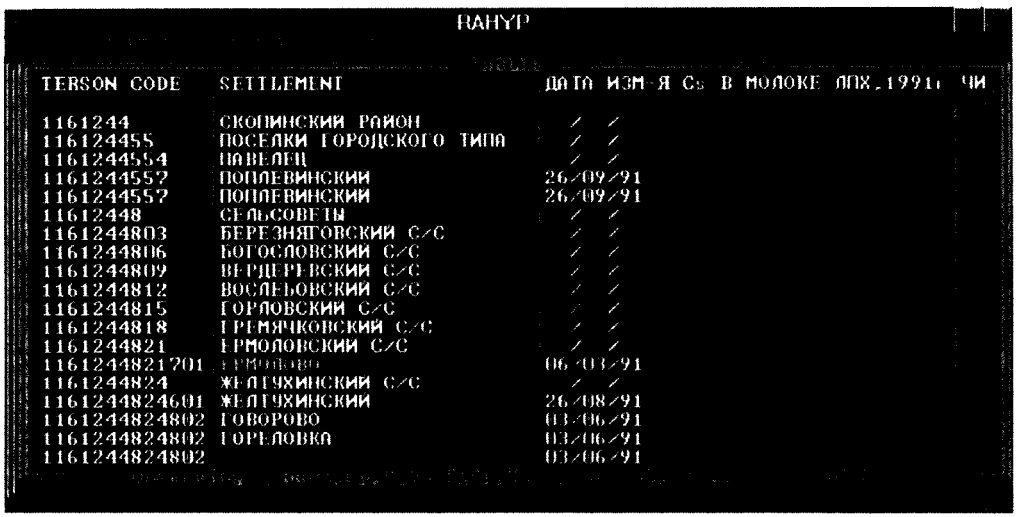

Figure 3. Sample of output standard form view on the screen

System was applied to estimate radiation consequences of Chernobyl accident for central European part of Russia: Ryazan, Tambov, Lipetsk and Kursk areas. System's interface is very easy and useful. It allows to use this system not only by scientists, but by staff of local sanitary and hygienic offices without special training. It was a key to successful distribution of system and following promotion of local data selection and integration.

RADAL: A DYNAMIC MODEL FOR THE TRANSFER OF RADIONUCLIDES THROUGH AGRICULTURAL FOOD CHAINS.

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ABSTRACT

The contamination of agricultural products by radionuclides is a mechanism which results in radiation dose commitment to the population, following fallout deposits from the atmosphere to the landscape. This paper describes the structure of the dynamic food chain model RADAL. This model simulates an acute environmental transport of fallout radionuclides through agricultural food chains to man and estimates the levels of radiation doses resulting from consumption of contaminated food. The development of RADAL was based on different existing models. For mathematical representation the transport of radionuclides was modeled through compartments representing environmental elements and/or food products. The model solves a set of linear, first-order, differential equations to estimate the concentrations of radionuclides in soil, vegetation, animal tissues and animal products as a funtion of time following their deposition. Dynamic physico-chemical processes of the model include the following: deposition and foliar interception, weathering, foliar absorption, soil resuspension, transfer from soil surface to the root zone, absorption by plant roots, transfer to deep soil, transfer to animal products, and human consumption of agricultural products. A parameter sensitivity analyses, performed for the main parameters of the model, showed that the foliar interception constant and resuspension factor are the most influential parameters over the radiation doses / model output.

INTRODUCTION

The deposition of the radiactive material on the land surface has associeted a radiological risk to man because of ingestion of contaminated food. There is a short time risk resulting from direct deposition of radioactive fallout on the agricultural products and a long time risk due to migration of contaminants through the human food chain.

The practice of the radiological assessment models in the environment, has been taken great importance nowardays. In recent years several more dynamic models for evaluation of the radionuclides behavior in the terrestral food chain and human intake, have been developed, e.g. ECOSYS (1), FOOD-MARK (2), RADFOOD (3) and PATHWAY (4). It is proper to remark that the trend has been toward more complex models; however, the increased complexity has not necessarily improved the accuracy of estimates of dose and, in certain cases, has had the opposite effect (5).

In the present paper the simple dynamic assessment model RADAL for simulation of an acute environmental tranfer of radionuclides through agricultural food chains to man, has been developed.

GENERAL DESCRIPTION OF THE RADAL MODEL

RADAL took as models of reference mainly those described by Koch J. and Tadmor J. (RADFOOD) and Whicker F. W. and Kirchner T. B. (PATHWAY), taking

into account a more understanding and complete dynamic assessment model to represent the transfer of radionuclides through food chains following release into the environment. With the present study we are offering a simple dynamic model where potentially important pathways, ten groups of vegetables and animals food and forage, and the critical radionuclides (Sr-89/90, Ru-106, I-131, Cs-134/137 and Pu-239) that can contribute to the radiation dose resulting from consumption of contaminated food, are identified. Fig. 1 shows the general structure of RADAL.

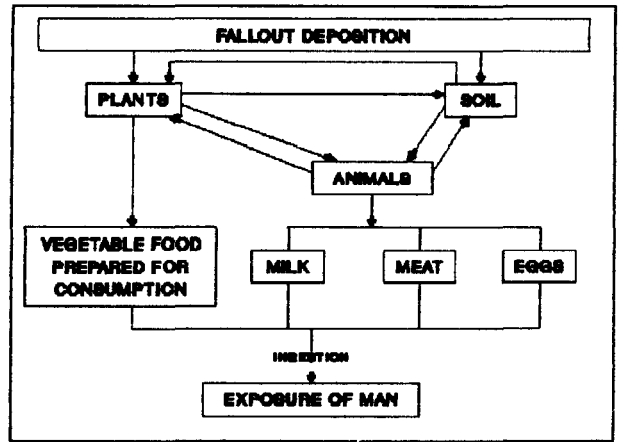

Figure. 1 The general structure of the dynamic model RADAL.

Mathematical model for the environment

The mathematical modelation of the agricultural food chais for a dynamic analysis of radionuclides transfer is carried out considering that environmental elements are united compartments between them by the transport process of mass. The description of radioactivity behavior in the determined compartment, knowing possible processes of transport and exchange between the contiguous compartments, leads us to obtain a set of differential equations with correspondent initial conditions. This model employs nine compartments or state variables, comprising the fruit surfaces, plant leaves, internal tissues of fruits, soil surface, root-zone soil, deep soil, milk, meat and eggs.

Physico-chemical processes of the model

The following transfer processes of radionuclides through agricultural food chains are considered in RADAL.

1. Deposition of radioactive fallout onto the soil surface and the fraction intercepted by leaves and fruits of plants.
2. Removal of deposited radioactivity on the plant and fruit surfaces by weather factors such as wind and rain, and other factors too.
3. Transfer of radionuclides from external parts of plants to internal tissues. Foliar absorption.
4. Resuspension of deposited material on the soil surface and subsequently deposit on the ground and/or vegetative surfaces.
5. Transfer of radionuclides from soil surface to the root zone.
6. Absorption of radionuclides by plant roots.
7. Transfer of radionuclides in soil from root zone to deep soil.
8. Ingestion of contaminated pasture and soil by animal.
9. Transfer of radioactivity to meat, milk and eggs, as well as other fractions are given back to the soil by excretion.
10. The fodstuffs obtained from animals may be treated and/or storaged. In these cases the physical decay is taken into account.

Radioactive decay reduces the radionuclide inventories in all compartments of the model over time.

Model parameters

The determination of environmental model parameters is a hard work that requires of a long time of research. For the RADAL model different radionuclide-dependent and independent parameters and methodology to determine them on reported literature (4-6-7-8-9) have been selected. Some specifical parameters of Cuban crops, soil and human diet, have been recorded.

Equations that describe process

The kinetics of the radionuclides transfer is described by a set of linear first-order differential equations that caracterize variation of state variables of system (10). Each equation corresponds to a state variable and expresses the change with time of radioactivity concentration in the compartment. In reference (11) the equations of RADAL are shown. The model estimates the levels of radiation doses resulting from consumption of contaminated food (12)

APPLICATION AND SENSIVILITY OF THE MODEL

The application of assessment model RADAL is to screen for the more important radionuclides and exposure agricultural food chains. The model solves a set of differential equations using a Runge-Kutta method (10), by a computer program elaborated by the authors.

A rigorous uncertainty analysis has not yet been performed, but a preliminary sensitivity analysis showed that the foliar interception constant and resuspention factor are the most influential parameters over the radiation dose/model output. The dose is not sensitive to changes with the tranfer parameters in soil.

CONCLUSIONS

RADAL as the dynamic model, solves in a simple and understandable way the transfer processes in the agricultural food chains. The internal radiation dose resulting from consumption of contaminated food, is estimeted too. This model may be used for prognostic and evaluation of the radioactive contamination in agricultural food chains following an acute release into environment, and so estimate, guide and be able to take immediate decisions to minimize the health detriment of the population by ingestion of contaminated foodstuffs.

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ON-LINE VALIDATION OF A PREDICTION MODEL IN CASE OF NUCLEAR FALLOUT

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INTRODUCTION

After a large-scale nuclear fallout the early prediction of the exposure of the population to be expected is of great importance to enable early decisions on countermeasures to be taken and to optimize such decisions to ensure an maximum reduction of the expected exposure if required. This includes the prediction of the contribution of each exposure pathway, in particular the ingestion path, and the contribution of important foodstuffs to this path.

The prediction should be fairly precise which requires a number of data at an early stage after the accident on a nation-wide scale. This is impeded by a number of limitations in availability of data in the early phase which are caused by various reasons: Some important data such as the activity concentration integral in air, total wet and dry deposition and radionuclide concentrations in important fodder are not available before the end of the passage of the plume and the resulting fallout. But also the time delay in taking samples of relevant items on a nation-wide scale, delivery of the samples to the measuring laboratories and the need of sample preparation especially with regard to certain non-gamma-emitting radionuclides contribute to this unavailability of data relevant for the prediction in the early phase.

Despite these problems responsible authorities require a realistic assessment of the situation and a prediction of the exposure to be expected at an early phase of radionuclide release and fallout.

Therefore, in Austria the decision was taken to perform the required predictions of the ingestion exposure at various stages during and after the fallout ensuring that a very rough prediction of the exposure pathways is available at a very early stage and improvements of the prediction quality are obtained at various stages after fallout along with the arrival of more reliable and more accurate data in the course of the event. In addition, predicted values of activity concentration in fodder and in foodstuff are compared to actually measured values as they become available and the calculated values are improved accordingly.

PREDICTION PHASES

The prediction of the exposure was segmented into 4 major chronological stages at which certain data or data sets are expected to be available. These stages and the relevant input data which are presumed to be available or most reliable among those available and which are used for the calculations, are given in table 1.

Phase Nr.	Time after start of plume [d]	Criteria for initiating the prediction calculation	Scope and Accuracy of prediction	Relevant input parameters
1	2 - 4	after passage of a significant part of the plume	very rough	- activity concentration in air - gamma dose rate monitoring stations - precipitation data
2	4 - 6	after the passage of the major part of the plume	refined prediction	- activity concentration integral in air - activity concentration in precipitation - precipitation data
3	6 - 12	after the passage of the total plume	regionally refined prediction	- activity concentration integral in air - act. conc. in precipitation and soil samples (wet and total deposition) - precipitation data
4	10 - 16	after availability of fodder and foodstuff activity measurements	very refined, taking activity concentrat. in foodstuff into account	input data as in phase 3, further refinement by comparison to actually measured values in foodstuff

Table 1 Stages of exposure prediction after nuclear fallout

Computational predictions are performed with the prediction model ECOSYS (1) which had been adapted to Austrian climatic conditions and food consumption patterns several years ago and thereafter named OECOSYS (2). In order to enable proper predictions, the code requires a set of data independent of date of occurrence and the activity in the plume. These are preset as described before (3,4). Other input parameters are only established at or after the fallout. These comprise:

activity concentration integral in air [Bq h m^{-3}]
wet deposition of each radionuclide [Bq m^{-2}]
amount of precipitation [mm]

The regional distribution of these values is required if for a specific region the exposure of persons living only on local foodstuff is to be derived, but also if the exposure of the general population in that area is to be predicted taking into account the transport of foodstuff from different regions of different fallout and growth stage levels within one country or in Europe.

In the early stage of a nuclear fallout only a few parameters will be readily available for evaluation in Austria:

1. data of the early warning system (336 gamma dose rate monitoring stations distributed all over Austria and transmitted in real-time and on-line to a central coordination and warning centre
2. activity concentration in air by a 6 on-line aerosol monitoring stations (total β-activity) and aerosol samples evaluated by gamma spectroscopy at a few places, mainly large cities and the Research Centre Seibersdorf
3. precipitation values by appr. 100 on-line meteorological monitoring stations which are also well distributed over Austria (TAWES - Teilautomatisches Wetter-Erfassungs-System)

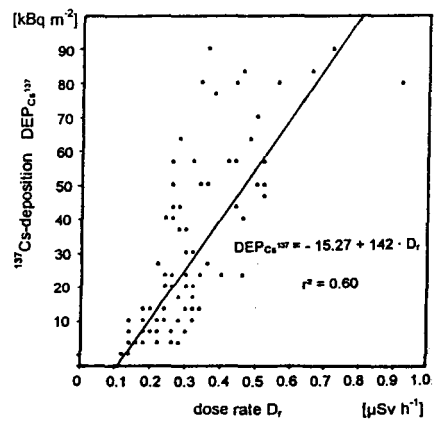
DERIVATION OF DEPOSITION VALUES FROM GAMMA DOSE RATE MONITORING

It has been demonstrated that from the on-line dose rate D_r monitoring early warning system the ^{137}Cs-activity deposition on ground $DEP_{Cs^{137}}$ (dry and wet) may be derived if data on radionuclide spectra are available (2,5):

$$DEP_{Cs^{137}}(t) = -15.27 + R_{Cs^{137}} \cdot D_r = -15.27 + 142 \cdot \frac{\sum k_i(x) \cdot g_i \cdot e^{-\lambda t}}{\sum k_i(\text{Chernobyl}) \cdot g_i \cdot e^{-\lambda t}} \cdot D_r \qquad (1)$$

k_i is the ratio of each radionuclide to ^{137}Cs, and g_i is the conversion factor D_r / DEP for each radionuclide. This takes into account that a relationship between dose rate monitored by the early warning system and ^{137}Cs-deposition values already was derived after the Chernobyl accident as displayed in Figure 1. Since the ratio of dose rate versus ^{137}Cs-deposition depends on the radionuclide distribution at the time of evaluation, the conversion factor may be arithmetically derived from the sum of the contribution of each radionuclide to the dose rate.

The deposition calculated by formula 1 is rather crude, but gives a quick and fairly reliable geographical distribution due to the large number of monitoring stations involved. The major reason for the rather crude estimate at this early stage is the fact that neither the complete activity concentration integral in air nor the total radionuclide deposition will be kown. Also the relationship between ^{137}Cs-activity deposition and gamma dose rate depends on the homogenuity of radionuclide distribution and on local variations of the ratio of the gamma dose rate displayed by the early warning station and ground deposition (2).



radio-nuclide	conversion factor g_i D_r / DEP	nuclide ratio on 10 May 1986	dose rate contribution
^{99}Mo	$0.8.10^{-4}$	0.07	0.004
^{103}Ru	$3.0.10^{-4}$	1.29	0.38
^{106}Ru	$1.4.10^{-4}$	0.32	0.04
^{132}Te	$1.7.10^{-4}$	1.19	0.16
^{131}I	$2.6.10^{-4}$	4.22	1.07
^{132}I	$13.0.10^{-4}$	1.19	1.25
^{133}I	$3.6.10^{-4}$	0	-
^{134}Cs	$9.4.10^{-4}$	0.56	0.53
^{137}Cs	$3.8.10^{-4}$	1.00	0.38
$^{140}Ba/La$	$16.4.10^{-4}$	0.28	0.43
$\sum k_i(\text{Chernobyl}) \cdot g_i \cdot e^{-\lambda t}$			4.00

Figure 1 Relationship between gamma dose rate monitored by the early warning system and ^{137}Cs-activity deposition on 10 May 86 and contribution of radionuclides to dose rate

With a plume possibly passing for 2 - 8 days, the inhalation of air-borne activity and also the fallout will not be finished at the first prediction date. Therefore, as soon as the major part of the plume passed (4 - 6 days), a fact given when the amount of activity concentration in air is less than 5 % of original values and no further increases due to changing wind direction are forecasted, the second prediction calculation is performed. The input parameters include activity concentration in precipitation (wet deposition) replacing the less accurate deposition values derived from the early warning system by formula 1.

After 6 - 12 days the plume definitely has passed and besides precipitation activity also activity concentration of soil samples which had been collected and evaluated gamma-spectroscopically, will be available. This gives more reliable data on total wet deposition than precipitation activity data (times precipitation values) and therefore will replace these values in the input data set for the ECOSYS-prediction calculation.

ON-LINE VALIDATION PROCEDURE

After the end of fallout and with sufficient data for the primary input to the prediction code, a further improvement of the prediction is achieved by measuring samples of fodder and foodstuff and comparing the activity of these with values predicted by the model. A set of samples as given in table 2 is taken over the whole territory of Austria. The major fodder samples taken would be grass, maize and barley. The relative amount varies according to season. The dominant foodstuff considered in this evaluation of prediction accuracy will be milk since milk samples taken in the dairies from one milk collecting tour will average over an area of app. 0,05 - 0,25 km^2 (6), thus providing very reliable data for prediction comparison. Also other foodstuff samples would be taken in order to assure that activity concentrations of these are adequately predicted, but these are of less importance than milk and therefore a lower number of samples is being foreseen. The sample collecting plan fixes the type of sample taken, the approximate number of samples, the authority responsible for taking the sample, the measuring laboratory and the transport means of the sample to the laboratory in advance.

type of sample	appr. sample number	sample specification	sample taking spot
maize - fodder	80	1 kg of leave and 1 kg of cob *	near TAWES station
barley - fodder	80	1 kg corn at harvest or 1 kg plant 14 d before	near TAWES station
beets - fodder	80	1 kg beet (well cleaned of soil) *	near TAWES station
milk	~ 600	1 l (frequency depending on season)	each milk collecting tour
cereals	100 - 150	minimum 0,8 kg corn, 0,4 kg straw *	1 sample per 400 km^2
potatoes	50 - 80	1 kg at harvest or 1 kg plant 14-30 d before	1 sample per 800 km^2
leavy vegetables	80	1 kg, only outdoor plants,according to season	1 sample per 800 km^2
fruit vegetables	80	1 kg, several types of plants at season	1 sample per 800 km^2
fruit	80	1 kg, several types of fruit, at season	1 sample per 800 km^2
meat	~ 6000	min. 0,7 kg	statistical selection **

Table 2 Samples taken nationwide to test and improve on accuracy of prediction model
* mixed sample of 10 individual samples out of several fields
** beef: each 50th animal, veal: each 30th animal, pork: each 300th animal

CONCLUSION

The prediction model established in Austria to guarantee a rapid and reliable prediction of the activity concentration in foodstuff and the resulting ingestion dose after a nuclear fallout has been considerably improved by performing the required predictions at various stages after the arrival of the plume. The first very rough prediction uses the readily available data of on-line transferred data from the gamma dose rate stations of the early warning system and the meteorological monitoring system. In further prediction computer runs more refined input parameters such as precipitation and soil activity concentrations are used to obtain more precise results. In the later stage fodder and foodstuff activity concentrations as actually measured are used to further refine the model predictions.

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PAPER TITLE COMPUTER DSS FOR ASSESSMENT OF COUNTERMEASURE STRATEGY IN THE LONG-TERM PERIOD OF LIQUIDATION OF THE CONSEQUENCES OF A NUCLEAR ACCIDENT (AGROSPHERE)

AUTHOR(S) NAME(S) B.I. YATSALO, O.A. MIRZEABASSOV, I.V. OKHRIMENKO, V.A. KUTKOV, V.F. DEMIN, M.N. SAVKIN, T.I. KHAIMOVICH

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MAJOR SCIENTIFIC TOPIC NUMBER 4.2 ... (see page 7); 4.5, 8.4

ABSTRACT (See instructions overleaf)

The objective of the works on development of the computer system indicated is: development of up-to-date approaches, models and program tools for investigation of the consequences of contamination of lands with long-lived radionuclides; estimation of effectiveness of countermeasures (CMs) in agro-industry; creation of the computer decision support system on protection and rehabilitation measures in areas subjected to contamination as a result of a nuclear accident. The work is close to the several studies in Europe, USA and Japan; however we pay special attention to the problems of CM-strategy including comparison different (agricuitural, administrative) CMs. The developments have some features: complex character (agro and radioecological, radiobiological, demographic, socio-economic, etc., models and databases); development of methods for assessing/comparison of a wide class of CMs on the basis of radiation protection principles and documents; the possibility of assessments using laws of distribution of values and parameters being used through the main chains under investigation; use of raster and vector GISs; creation of an open computerized system based on the principles of object-oriented programming that makes it possible to use any module (model) from the library; creation of the multilevel DSS, both in the aspect of the scale of the object being investigated (region-district-[farm]-settlement) and from the point of view of the subject of investigation (three-level interface). These works are participated by specialists from several institutes of Russia (RIARAE (Obninsk), RRC "Kurchatov Centre" (Moscow), RRC-VNIIEF (Kremlev); collaboration is planned with institutes of the CEC countries and Japan.

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PAPER TITLE PROBABILISTIC METHODS IN RADIOECOLOGY: EVIDENT ADVANTAGES AND NON-TRIVIAL PROBLEMS

AUTHOR(S) NAME(S) B.I. YATSALO, O.A. MIRZEABASSOV, I.V. OKHRIMENKO, I.A. PICHUGINA

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MAJOR SCIENTIFIC TOPIC NUMBER 4.2. (see page 7)

ABSTRACT (See instructions overleaf)

Methodological aspects in realization of Probabilistic Methods (PM, i.e., those taking into consideration the corresponding distribution laws for values used) are scrutinized. The idea of using PM in radiology is not new; however, the major part of the works describe inhomogeneities observed (distribution laws) and do not deal with the problems of including the values under consideration in more complex estimation based on operations with these values. Besides, any analysis of problems connected with the use of PM is lacking; so, groundless optimism appears in practical operations with random/distributed values. The wish to "improve" prediction of some values under investigation by using the PM can face ambiguous estimates which cannot be verified in many cases. The problem of final presentation of the values' distribution is far from limited to differences in the modes of integrating the distributions characterizing different compartments. There is also the problem that no distribution laws for source values exist a priori; distribution law can be considered only after detalizing the probabilistic spaces. Significance of the differences appearing in determination of the structure of plant and animal products contamination is considered by concrete examples of contaminated areas. The main charakteristics of production contamination structure connected with the use of alternative methodological approaches, different kinds of distribution of input values, as well as ambiguous choice of the main parameters of these distribution are considered.

HUMAN RADIATION DOSE RESULTING FROM FORESTS CONTAMINATED BY RADIONUCLIDES: GENERIC MODEL AND APPLICATIONS TO THE CHERNOBYL ECOSYSTEMS

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ABSTRACT

Forest ecosystems have been found to contribute significantly to the human radiation dose in the intermediate and long terms following radionuclide releases. Evaluation of the internal and external radiation dose for these critical population groups requires knowledge of radionuclide transport processes in forest ecosystems, as well as the extent of forest utilization by these populations. The high complexity of the problem requires the use of models to define and analyze the properties of the forest as well as to evaluate the ecosystem response to possible human intervention. A generic FORESTPATH model is used to calculate the internal and external radiation doses for different critical groups of consumers at different times following radionuclide release. The model is tested using the information available for contaminated forests in Belarus. Uncertainty of the model predictions are estimated by means of Monte-Carlo simulations.

EXPOSURE ASSESSMENT

Several pathways exist whereby people are exposed to radionuclides from contaminated forest ecosystems. In general, external and internal components of the total radiation dose can be distinguished. The external dose results from direct irradiation due to the radionuclides present in the local environment, while ingested and inhaled radionuclides contribute to the internal dose. In the case of contaminated forest ecosystems, the external dose can be received by the general public via the direct gamma shine while walking in the forest or through irradiation from radionuclides incorporated into construction materials and paper. Occupational exposure can be received by working in the contaminated environment and by irradiation from by-product materials and from timber cut for forest industries and for fuel. The internal dose arises from the ingestion of forest products (berries, mushrooms, birds, game, etc.) and from inhalation of resuspended radionuclides from soil and plants; in addition, radiation dose is received from ash caused by forest fires and from wood burned for heating purposes. These exposure routes can affect populations far removed from the contaminated zones.

A scenario is considered for a rural population inhabiting an area near the Exclusion Zone of Belarus contaminated by 5 Ci/km^2 of ^{137}Cs. This population consumes forest products collected from a pine forest near the village which has 5 Ci/km^2. Large segments of this population are involved in forestry and agriculture activities. Work in the forests implies an annual occupational exposure of about 1,000 hours in areas characterized by a surface deposition of 20 Ci/km^2. Table 1 presents probability distributions for the average annual consumption of forest products, exposure time for occupational forest use, forest characteristics which are typical of the 30-year-old pine plantations in Belarus and conversion factors.

Table 1. Parameters used for probabilistic modeling of risk associated with contaminated forest ecosystems

Parameter	Notation	Unit	Distribution			Reference
			shape	mean	range	
residence area contamination	RAC	kBq/m^2	constant	185		site-specific
forest contamination	FOC	kBq/m^2	constant	740		site-specific
Organic Layer accumulation	OL	% initial inventory	triangular	55	15 - 75	1
Organic Layer biomass	OLB	kg/m^2	triangular	3.1	2.4 - 4.5	2
Understory biomass	UN	% initial inventory	triangular	0.2	0 - 2.2	1
Understory biomass	UNB	kg/m^2	triangular	0.5	0.01 - 1	estimates
mushroom consumption	MUC	kg/yr	triangular	10	5 - 15	3
mushroom processing retention	MPR	fraction	triangular	0.2	0.1 - 0.4	4
berry consumption	BEC	kg/yr	triangular	6	5 - 20	3
berry processing retention	BPR	fraction	triangular	0.6	0.7 - 0.8	4
external dose rate factor	DRE	(nSv/yr)/(Bq/m^2)	triangular	24.5	22 - 30	3
internal commited dose rate	DRI	nSv/Bq	triangular	13	10 - 22	3,5
occupational time spent in forest	FOO	hr/yr	triangular	1000	500 - 2000	estimates
dose-fatal cancer risk conversion factor	DRC	Sv^{-1}	uniform	0	0.08	6, estimates

Two major routes of exposure for this population are the external irradiation due to gamma emission from the forests and the internal exposure due to radionuclide ingestion from forest berries and mushrooms. The resulting average external dose for the i-th year following the deposition can be calculated using:

$$EP_i = (RAC_i + FOC_i * FOO) * DRE \quad (1)$$

where EP_i is the annual average external dose (mSv/yr).

The average internal dose can be calculated using:

$$IP_i = ((OL_i/OLB) * MUC * MPR + (UN_i/UNB) * BEC * BPR) * DRI \quad (2)$$

where IP_i is the annual average internal dose (mSv/yr),
OL_i is the radionuclide concentration in the Organic Layer (% of initial inventory),
UN_i is the radionuclide concentration in the Understory (% of initial inventory),

Additional irradiation dose can be encountered due to radionuclide resuspension by soil or ash (if timber is used for heating purposes), use of contaminated construction materials and paper, forest fires, consumption of the game, etc. The resulting dose from these and other processes is a subject for future investigation and is not considered in this paper.

The risks of low-dose-rate external exposure to radiation have been obtained by extrapolation from high-dose-rate data (5). Recent compilation of the available data for different exposed population (7) provides reasonable doubt that low-level radiation is harmful. Thus, a uniform distribution from zero to twice the value recommended in (5) was used for dose-risk conversion factor in risk calculations:

$$R_i = (EP_i + IP_i)) * DRC \quad (3)$$

RADIATION DOSE AND RISK

The FORESTPATH model (1, 8) is used to establish the radionuclide distribution among forest compartments. In this paper, coniferous forest ecosystems which are typical for the Chernobyl Exclusion Zone are considered. An initial contamination density of 5 Ci/km^2 and generic FORESTPATH parameters (1, 8) were used, as well as the forest characteristics presented above. The current radioactivity of mushrooms is significantly higher than that found in berries because mushrooms take up nutrients and, therefore, absorb radionuclides from the Organic Layer (1, 8). On the other hand, the activity in berries, which have a deeper root zone and extract their nutrients from the Labile Soil, is shown to increase with time reaching a maximum at about 10 years following the accident (1, 8).

Figure 1a presents the external radiation dose resulting from living in an area with contamination density of 5 Ci/km^2, as well as the additional radiation doses due to occupational exposure in the forest and to forest product food consumption. Work in a contaminated forest of 20 Ci/km^2 leads to an annual external dose of 4.2 mSv (sum of public and forest worker exposures) for the fifth year following the accident. This is about 2.5 times greater than the average external dose received by the non-forest-worker population in this area. Dose rate measurements for forest workers were conducted at several Belarussian sites (2). Their occupational exposure was found to be 1.75 to 2.9 times higher than the external dose to the general public living in the same area, which is in agreement with the model predictions. After eight years, the human radiation dose is slowly decreasing with a half-time of about 30 years due to the physical decay of ^{137}Cs.

Monte-Carlo simulations were conducted to address the influence of the uncertainty in input parameters (Table 1) to the internal radiation dose (Figure 1b). Each Monte-Carlo simulation produced 300 estimates of the internal dose resulting from the berry and mushroom consumption seven years following the contaminating event. Parameter values were selected using the random Latin Hypercube method. Cumulative distribution functions for internal dose reveal a wide range of variation depending on consumption of the forest products and site contamination.

The cumulative distribution function for the risk of developing a fatal cancer for the general public is presented in Figure 2a. The high range of variation results from the uncertainty in input parameters (Table 1). To find the most important parameters for the model performance, a sensitivity analysis was conducted using the technique of partial rank correlation (Figure 2b). The uncertainty in the DRC (dose-risk conversion factor) dominates the range of model outputs. Unfortunately, current literature does not allow any better estimation of this parameter. All other parameters are significantly less important. Among the other parameters, the organic layer contamination (OL) and internal commited dose rate (DRI) are relatively important.

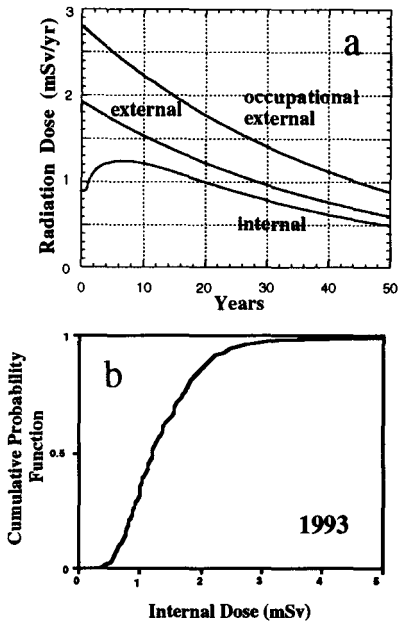

Figure 1. Excess radiation doses received by different populations (a) and cumulative probability function for internal dose received in 1993 (b).

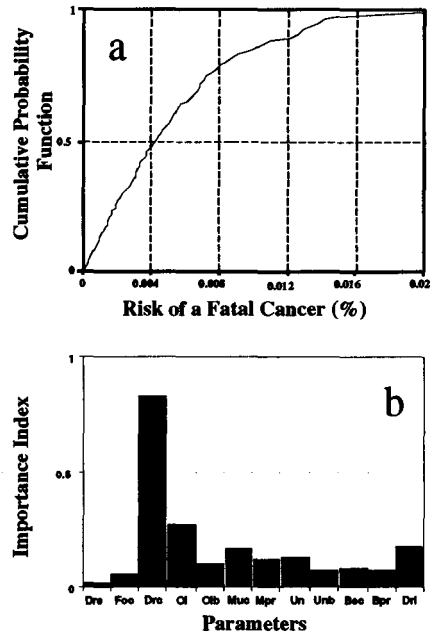

Figure 2. Cumulative probability function for risk of a fatal cancer due to consumption of forest product by a critical population group (a) and importance index for model parameters (b).

CONCLUSIONS

The complex problem of radionuclide contamination in forest ecosystems requires the use of a model to synthesize and analyze the properties of the entire ecosystem. Modeling can provide not only estimates of the radiation dose but also can predict future trends of dose accumulation. The generic forest model developed and used here provides a starting point for evaluating internal and external radiation doses and risks over long time periods. Contaminated forests constitute a significant hazard to the public over long periods of time depending, in a major part, on the forest food intake. Occupational exposure can be several times greater than that for the general public. The use of a dynamic model can facilitate the decision-making process and helping to design efficient abatement, and remedial and social policies in radionuclide contaminated regions.

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APPLICATION OF A DYNAMIC MODEL FOR EVALUATING RADIONUCLIDE CONCENTRATION IN FUNGI

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ABSTRACT

Global fallout from nuclear weapon tests in the 1960s revealed the potential of fungi as an enhanced accumulator of radioactivity. Data derived from Chernobyl fallout sampling has shown fungi to be a major accumulator of radiocesium and an important food-chain contributor to the human radiation dose. Fungi can significantly affect the radionuclide cycling in forests. According to experimental data and estimations, forest microflora, particularly fungi mycelia, could retain up to 40% of radiocesium. This paper illustrates the application of the dynamic model FORESTPATH to evaluate the contamination dynamics in fungi and the relative importance of fungal species for forest cycling. Only a few studies have been made to model fungi contaminated by radionuclides and these utilize Transfer Factors to describe soil-to-fungi uptake of radionuclides. Such an approach has serious limitations, since equilibrium conditions and specified soil sampling depths must be assumed. The FORESTPATH model uses rate of uptake and residence half-times for radionuclides to describe this process. The model was applied to describe radionuclide dynamics in fungi for the case of chronic deposition and for the accidental release of radionuclides. Experimental measurements of fallout from nuclear weapon tests and from the Chernobyl accident were used to test the FORESTPATH predictions.

INTRODUCTION

Data derived from sampling of the areas contaminated by the Chernobyl fallout have implicated the Organic Layer as being a major accumulator of radiocesium (1, 2). Fungi are the highest living biomass in the decomposing organic layer and are the primary sources of the enzymes necessary to degrade the litter and, thus, are very important for radionuclide migration in forests. Fungi mycelia and fruit bodies are mainly located in the upper part of the organic layer. Although above-ground mushrooms are a minor contributor to the total fungal biomass (according to Olsen (3), this contribution may be as low as 1%, leaving 99% in the below-ground biomass) they can contribute significantly to the human radiation dose being consumed as a food.

Very little is known about the mechanisms involved in the radionuclide uptake and retention by fungi. Fungi can play a significant role in radionuclide retention by the organic layer. Olsen et al. (4) estimated that some 32% of the total Chernobyl-derived radiocesium present in a forest soil in Norway is contained in fungi. This estimate was confirmed by experimental data (5). Their experiments with artificial washout of humus samples show that forest microflora (particularly fungi mycelia) could retain up to 40% of radiocesium. The absorption of nutrients (and thus radionuclides) by fungi from the ground occurs in aqueous solution usually from the soil solution. Fungi use enzymes to break down macromolecular complexes in solution for digestion. Once broken down, most substances are thought to move into the *hyphae*, being bound to specific carrier molecules. The rate of this uptake by both saprophytic and symbiotic fungi depends on concentration of the ion, moisture content of the soil, the growth rate of the plant and, in the case of symbiotic fungi, the transpiration rate of the host plant.

Several studies have examined the differences in the ability of saprophytic and symbiotic fungi to accumulate radiocesium. Results of several studies (5, 6) indicated that the symbiotic fungi accumulated, on average, statistically higher concentrations of radiocesium although both types of fungi collect radiocesium quite effectively. These observations can be explained by the differences in mycelium location in the soil profile as well as the migration of Chernobyl-derived radiocesium through forest soil. Other hypotheses include differential retention of radionuclides by fungi (5).

The life cycle of soil fungi may have an impact on the time of immobilization of radionuclides in mycelium. Differences exist between the life time of different species of fungi. According to Olsen et al. (4), the lifetime of mycorrhizal fruit bodies (mushrooms) is normally between 10 and 17 days. The life of fungal biomass in

soil is not as well understood and is suggested to be on the order of two years. After death, the release of nutrients proceeds primarily by leaching. As water comes in contact with the cell wall of the fungi, nutrients such as K^+ diffuse across their concentration gradient and into solution. Another less important release pathway is due to animal consumption.

Quantitative evaluation of the relative fungi contamination is a challenging task. The traditional approach is to evaluate the Transfer Factor (TF) for soil-to-fungi transfer. TF can be defined as the ratio between the activity in plant (Bq/kg dw) divided by the activity in soil (Bq/kg dw) or by the deposition per unit area (Bq/m^2). Large variability in the TF values has been reported. TF across different species varies over several orders of magnitude (7).

The TF approach has serious fundamental limitations because it is applicable only to equilibrium situations, which can only be achieved in natural ecosystems a long time after the contamination event (10-20 years for the case of forest ecosystems (1,8)). Multi-layered soil structure with variable depth of soil horizons of the natural environment and inhomogenuity of the radionuclide deposition and its soil migration requires definition of conditions for sampling depth and volume. By including an estimate of the depth to which the radiocesium has migrated (i.e., Cs-134/Cs-137), Guillitte et al. (5) were able to obtain a more accurate estimate of the ability of a particular fungi species to take up radiocesium. Thus, it is problematic in defining the soil volume where nutrient uptake occurs. The dynamics of fungal biomass growth and decay are not well understood and questions remain regarding the nutrient requirements of this plant over time.

MODELING OF ORGANIC LAYER AND MUSHROOM CONTAMINATION

The FORESTPATH model (1, 8) calculates a time series of inventories for a specific radionuclide distributed within the following six compartments: Understory, Tree, Organic Layer, Labile Soil, Fixed Soil and Deep Soil. In this paper, the model was developed further to incorporate details of radionuclide migration in the Organic Layer and fungi. The Organic Layer is represented by three horizons: Ol (litter), Of and Oh. Reported residence time for Cs in these horizons (see (1) for review) along with values for other FORESTPATH parameters for Chernobyl forests (1, 8) were used for model simulations. Figure 1 shows radionuclide accumulation by the Organic Layer as well as by the Labile and Fixed Soil compartments, in a coniferous forest in Chernobyl described in (1, 9) (an initial deposition of 5 Ci/km^2 was assumed in the calculations). Organic Layer compartments exhibit complex time dynamics. The litter compartment is significantly contaminated immediately after the deposition, but looses all its activity during the first year due to wash-off and leaching towards deeper layers. Of and Oh horizons show accumulation peaks at about two and four years respectively following the initial deposition.

Mushroom mycelia exist in different soil horizons. Given the complexity of the nutrient uptake by mushrooms and lack of theoretical description of this process, it could be assumed that mushrooms have the same contamination density as the layer in which their mycelia are located. Using this methodology, calculations have been made for contamination of two types of mushrooms in coniferous forests. The first type has mycelium developed in the Of horizon, while the second type has mycelium located in the Oh-upper layer of mineral soil. A representative edible mushroom species of the first type is *Boletus badius* (or *Xerocomus badius*) (10), while *Boletus edulis* is representative of the second type (5). Based on the FORESTPATH model, *Boletus badius* would be expected to have the same dynamic of contamination as the Of horizon (i.e., have maximum accumulation during the second year following the accident (Figure 1). *Boletus edulis* which takes nutrients from the labile soil, would be expected to increase its contamination during the first decade following the accident.

MODEL VALIDATION

Model predictions are validated based on: 1) compilation of the reported ^{137}Cs concentrations in these two mushroom species in Central Europe (generic model), and 2) concentrations reported for contaminated forests in the Exclusion Zone (site-specific model). Figure 2 shows compilation of the existing literature values for mushroom contamination (see (1) for compilation tables). It is clear that *Boletus badius* exhibits an accumulation peak around 1988, while *Boletus edulis* concentration still increases (at least until 1991). Significant variability in the reported data can be explained by different contamination levels, climatic conditions, soil structure and types, etc. In the Chernobyl forests, *Boletus badius* and *Boletus edulis* exhibit the time dynamics expected for plants with roots developed in Of and Oh-Mineral Soil compartments (Figure 3). Predicted values from the FORESTPATH are about two times larger than the values measured by the Forest Research Institute, Belarus. Due to the fact that the experimental data were obtained based on reported values for the TF, the differences of a factor of two can be explained by the large uncertainty incorporated in the TF determination.

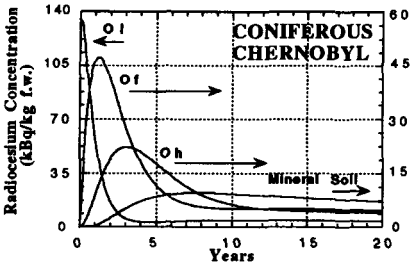

Figure 1. ^{137}Cs concentration in soil compartments of a coniferous forest in Chernobyl over 20 years from an initial acute deposition (kBg/kg f.w.). The scale of the ordinate is 140 kBq/kg for the Ol horizon and 60 kBq/kg for all other compartments.

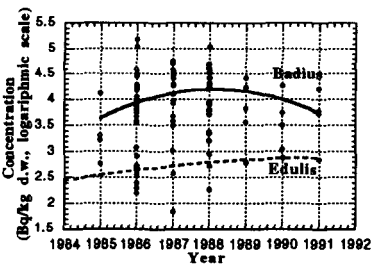

Figure 2. ^{137}Cs concentration in *Boletus badius* and *Boletus edulis* collected in European countries.

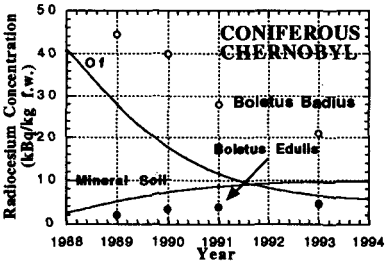

Figure 3.
Cs-137 concentration in Boletus badius and Boletus edulis collected in Chernobyl forests (circles, after Ipatyev (11)) compared to those predicted by FORESTPATH for species having mycelia in Of and Mineral Soil horizon (solid lines).

CONCLUSIONS AND RECOMMENDATIONS

FORESTPATH can be applied for detailed evaluation of the radionuclide accumulation in a specific forest compartment. The model predicted an enhanced accumulation by fungi with shallow mycelia during the first years after an acute deposition, while the accumulation peak for deep-rooted species is delayed. These predictions have been validated based on the experimental data collected in Europe (generic approach) and in the Chernobyl Exclusion Zone (site-specific application).

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THE ECOLOGICAL PROBLEMS IN BELARUS AND UKRAINE: INVESTIGATION AND MATHEMATICAL MODELLING OF RADIONUCLEDES IN ATMOSPHERE SOILS AND SEAS

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ABSTRACT

The report deals with some ecological problems in artificial radioactivity of Belarus and Ukraine in the atmosphere, soils, Black sea, rivers and forests after Chernobyl accident.

INTRODUCTION

The accident at the Chernobyl nuclear power station and the global scale of consequences had spurred intensive research on radioactive contamination of Black Sea waters.

Among more than 30 radionuclides detected in the blow-outs from the destroyed reactor, cesium-137, cesium-134, and strontium-90 radioisotopes are of greatest interest as the half-life of other isotopes with significant activity lasts for several days. Also, plutonium radioisotopes, which accounted for 3 % of the overall discharged by the reactor, are was of great interest (Information, 1986). Given this setting, the soviet investigators concentrated on the study of the concentration of cesium and strontium radioisotopes in Black Sea waters.

EXPERIMENTAL

Figure 1 exhibits the results of measurements of cesium-137 concentration in Black Sea surface waters carried out in the period from 15 to 23 June, 1986 (Cruise 33 of the R/V AKADEMIK VERNADSKY). The largest cesium-137 concentrations were documented at st. 35 - 720 Bq/m^3 of cesium-137, cesium-134 concentration being 372 Bq/m^3 . The range of variation of cesium-137 and cesium-134 concentrations is within 34 (st.30) - 720 and 17-372 Bq/m^3 , respectively. The density of cesium-137 distribution at station 35, located off the South Crimea, reached 32.3 KBq/m^2 (876 mCi/m^2) in the upper mixed layer. From Polikarpov's data (Polikarpov and Kulebakina, 1989) the cesium-137 concentration attained the value of 815-840 Bq/m^3 in this area in May 1986 [1].

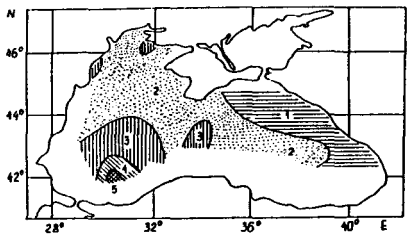

Fig. 2. The distribution of Cs-137 concentration in the Black Sea surface waters. December, 1986. (See Fig. 3.8 for denotements).

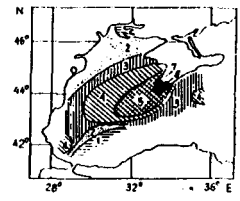

Fig. 1. Cs-137 distribution in the surface waters of the western Black Sea in June 1986.
1 = less 100 Bq/m³; 2 = 100- 200 Bq/m³;
3 = 200- 300 Bq/m³; 4 = 300- 400 Bq/m³;
5 = 400- 500 Bq/m³; 6 = 500- 600 Bq/m³;
7 = more 600 Bq/m³.

The data given in Figure 1 are smoothed, compared to the real field of cesium-137 distribution in the Black Sea surface layer, not only because of the rarity of stations. Obviously, in June 1986, an important role in smoothing of the originally patchy contamination of surface waters belonged to physical mixing processes. Against the general background of Black Sea surface water contamination, the northwestern shelf area (where in June 1986, the lowest cesium-137 and cesium-134 concentration, being about 100 Bq/m^3, was registered) stands out. From our data, the mean Cs-137 and Cs-134 radionuclide content reached 272±171 Bq/m^3 and 136± 86 Bq/m^3 , respectively. It should be noted then that their mean values for the northwestern shelf amounted to 107,20 Bq/m^3 and 56±9 Bq/m^3 respectively. The Cs-134/Cs-137 ratio acquired for the period at issue equalled 0.50,0.005. The variation coefficients for cesium-137 and cesium-134 concentrations and the Cs-134/Cs-137 ratios reached 63 %, 63 %, and 10 %, respectively. This points to the simultaneous input of cesium radionuclides into the Black Sea surface waters. The temporal trend of cesium isotope surface concentration identified in (Livingston et al., 1989) at the same stations can be explained by the water mass mixing. All the investigators note that Cs-137 and Cs-134 concentrations in the surface waters, both in June and September, vary dramatically within the range between the two sampling stations. We can explain this by the variability of Chernobyl radionuclide deposition to the Black Sea surface. Livingston et al., (1989) emphasize that in September, compared to June, a growth in cesium radionuclide concentration resulted from the surface circulation and transport of the more active waters from the norhern Black Sea to the Bosporus region. This pattern implies that the Chernobyl radioactive depositions in the northern Black Sea are higher than in the southern area. This is confirmed not only by meteorologic models (World Health, 1987)

but also, by the results of the investigations implemented in November-December 1986 practically throughout the Black Sea, except for the Turkish economic zone (Chudinovskich and Eremeev, 1990).Figure 2 depicts the distribution of cesium-137 concentration in the surface waters during this period. The sampling stations are marked by points. Obviously, over the seven months, since the moment radionuclides had entered the Black Sea, considerable smoothing of the initial patchy contamination took place.

In the spring of 1987, the flood was awaited with particular concern. Patterns in the spring snow thaw showed the possibility of considerable injection of radioactive products into the Black Sea shelf zone. The greatest alarm was caused by the radioactive situation in the northwestern shelf, whose fluvial discharge accounted for 80 % of the total run-off into the Black Sea.

Analysis of the results obtained permits us to claim that no appreciable influx of cesium radionuclides with river waters into the northwestern shelf area occurred. Over the year after the accident the mean values of cesium-137 and cesium-134 concentrations had essentially decreased. In June 1987, these amounted to 96± 47 and 25 ± 13 Bq /m^3 , respectively, for the western Black Sea. A different situation occurred in the spring and summer of 1987 in the eastern Black Sea. In the waters off the Caucasus, the concentration of cesium radionuclides increased 1.5 - 2 times. The cesium-137 concentration in the surface waters in December 1986 being 150 Bq/m^3 , at most, in June 1987, it varied within the range of 200 - 300 Bq/m^3 .

The mean values of cesium-137 and cesium-134 concentration in November-December 1986 reached: - 102 , 39 and 45 ± 15 Bq/m^3 in the eastern Black Sea, and 177 ± 80 and 81 ± 41 Bq/m^3 in the western Black Sea part, respectively. It should be noted that despite the general decrease in the mean values of the surface radiocesium concentrations in December 1986, as compared to June, their presence in the waters of the northwestern shelf slightly increased. In June, the mean values of cesium-137 and cesium-134 concentrations in the shelf areas reached 107 ± 20 and 56 ± 9 Bq/m^3 , respectively, whereas, in December, they were 143 ± 63 and 59 ± 19 Bq/m^3 , respectively. The increase in radionuclide concentration may have occurred both through transfer of the more contaminated waters as a result of surface circulation and the delivery of radionuclides with river waters during the period of autumn flood. We will not elaborate on the features of radioactivity inputs by the river run-offs, as this problem will be considered further.

The cesium radionuclide concentration distribution in the eastern Black Sea during this period was uniform and lay in the range of 50-150 Bq/m^3 (Fig. 2).

The intensive snow thaw in the mountains and the influx of cesium radionuclides with melting waters. The mean values of cesium-137 and cesium-134 concentration for the entire eastern Black Sea during this period reached 119 ± 83 and 30 ± 21 Bq/m^3 , respectively.

The range of variation in cesium-137 concentration lies within 25 and 280 Bq/m^3 , and for cesium-134, the variation was from 7 to 70 Bq/m^3 .

From data compiled by the same authors, Sr-90 concentrations in the Black Sea waters in 1987, on average, reached 35,2 Bq/m (the range of measurement being within 25 - 78 Bq/m^3) in the northwestern corner, 22.9 Bq/m^3 (17 - 31 Bq/m^3) in the western area and 22.6 (17 - 24) Bq/m^3 in the eastern area. On the whole, the mean values of Sr-90 concentration for the Black Sea reached 19.6 Bq/m^3 . Similar investigations were conducted in 1988-1994.

During the ten years since the Chernobyl accident and radionuclides input into the Black Sea a gradual decrease in cesium-137 and cesium-134 concentration was observed in Black Sea surface waters. In 1991 the cesium-134 concentration in the majority of samples was already too low to be accurately determined, whilst Cs-137 concentration did not exceed 50 Bq/m^3.

Thus, the Black Sea surface waters have exhibited a great capacity for "self-purification". It is obvious that "self- purification" of this kind occurs, primarily, because of the processes of water mixing and radionuclide redistribution between the surface and deep waters, as well as because of the physico-chemical processes and bioaccumulation leading to sorption and absorption of radionuclides and, finally, their transfer into bottom sediments.

RADIOACTIVE CONTAMINATION IN BELARUS

In the Republic of Belarus, 23% of the territory (3,666 sett lements) have been contaminated by radiation with a density of Caesium -137 contamination of more than 1 Ci/sq. km. The highest density of Caesium -137 contamination, exsept for a zone of 74 settling out, is revealed in Chericovsky region of Mogilev oblast (146 Ci/ sq.km). There were a total of settlements in a zone of primary settling out (more than 40 Ci/ sq.km with the Caesium 137). The area of agricultural territories contaminated by radioactive Caesium with densities more than 1 Ci/ sq.km equals 1.6 mln. hectares. About thousand hectares of fofest are contaminated by radionuclides [2].

The contamination by Strontium -90 has more local character. The sites with the contamination levels of 3 Ci/ sq. km are found out in Hoiniky region, particular sites with densites 2-3 Ci/sq.km are found in Hoinikovsky, Braginsky, Dobrushsky and Vetcovsky regions of Gomel oblast. The soil contamination by Plutonium-238, -239,-240 higher than 0,1 Ci/ sq.km is revealed basically in a zone settling out, particular spots are presente in a number of regions in the Gomel oblast.

RADIOACTIVE CONTAMINATION OF WATERS OF BELARUS

The monitoring system for the radionuclides content in surface waters of the main rivers of Belarus (the Dnepr, the Sozh, the Priryat, the Iput, the Besed) showed that immediately after CNPP accident, the concentration of Sr-90 in down stream of the Pripyat river have exceeded permissible concentration of this radionuclide in water, but already in May 1986 its concentration in water has descreased to $1\ 10^{-10}$ Ci/l and in 1994 was already $1.0\ 10^{-12}$ Ci/l. For the whole monitored period since 1987, the excess of Cs-137 content in surface waters of the large rivers of Belarus was not recorded. Caesium-137 loss by surface waters on the territory of Belarus considerably descreased over time. The substantial share of Cs-137 comes into the Dnepr river from the Belarus-Bryansk Caesium spot with warer of the Sozh and Iput rivers. One should note, that because of the Sozh river, the Cs-137 loss descreased over time from 280 Ci/year (1987) to 20 Ci/year (1994) (Fig.3 and 4).

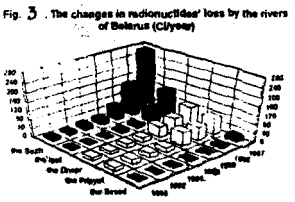

Fig. 3 . The changes in radionuclides' loss by the rivers of Belarus (Ci/year)

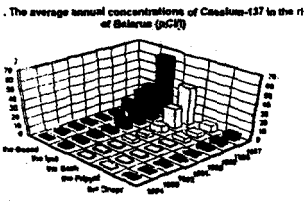

Fig. 4 . The average annual concentrations of Caesium-137 in the rivers of Belarus (pCi/l)

If in the first days after accident at CNPP the increase of radionuclides concentration in water was stipulated by direct fall of radionuclides on wares surfaces, now the contamination levels of water systems by secondary processes: exchange with bottom sediment, radionuclides washing out from surface of the river catchment, as wellas at the expebse of thawed and flood waters.

RADIATION CONTAMINATION OF THE SOILS AND FORESTS

As the result of CNPP accident, 4.8 mln. ha of the teritories Belarus are contaminated by radiation, including 1.2 mln. ha (14.5%) of the republic's forests. The main weight of radionuclides is concentrated in the top 5-centimetre soil statum. In sandy unbogged soils with attributes of redundant humidifying, the depth of radionuclides penetration is 8-9 cm., in the same soil, but alluvium -16-17 cm, in gleyey and gley soils- 11-12 sm. Strontium-90 because of less strong link with soil absorbing complex, migrates faster than Cs-137. The depth of its penetration lies below 15 cm statum.

THEORETICAL INVESTIGATION

The mathematical theory of diffusion and convective transfer of radionuclides in vaious media (atmosphere, seas and ocean, riwer and other) are discussed. The main bounlary-value direct and inverse problems of radionuclides transfer through plane, celindrical, spherical walls were solved under the conditions of diffusion, convective transfer radioactive fussion, absorbtion fnd relaxation process. The discrete spectrumof direct problems was determined, the eigen function of spectral problems and their roots were built. Using of the eigenfunctions the solutions were built in the form of discrete spectral decompposition, the concentration distribution of the diffusion and convective flow in the dispertion and porous media were calculated at different radionuclide disrtributions in the volume and at the bound of the fields.

The new integral transfomation in the finite and infinite were built using eigenfunction of the spectral problems. A number of invers problems are formulated.

The migration of natural and artifical radioactive isotopes in the turbulent atmosphere and stratified lake and sea are studied. The equations global and vertical transfer of radionuclides in the turbulent boundary layers with classical and generaliszed boundary conditions are presented.

Mathematical models,numerical algoritms and computer programs for numerical solutions of diffusion in atmosphere, lake and sea are presented.

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THE MATHEMATICAL MODELLING AND EXPERIMENTAL INVESTIGATION TRANSFER OF RADIONUCLIDES IN BOUNDARY LAYERS OF THE NEAR-SURFACE ATMOSPHERIC LAYER IN THE BLACK SEA REGION AND BELARUS IN 1986-1994

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ABSTRACT

The migration of natural and artificial radionuclides in the turbulent atmosphere in the Black sea region and Belarus after Cherobyl accident are discused.

INTRODUCTION

Investigations of radioactive contamination of the near surface atmospheric layer in the Black Sea region were conducted by MHI, Ukrainian Academy of Sciences, from May 1986 through March 1994. FPP filters were used to concentrate air radionuclides. The air was supplied to the filters by the blast blower. The volume of the air that passed through the filters reached (120-200)*10 m , the averaging period being 6-7 days. In the investigations accomplished in May-June 1986, the period of testing lasted 12-24 hours. Measurements of gamma-activity in the filters were conducted using the low-background gamma-spectrometer featuring a semiconducting detector. Later on, when air aerosol gamma-activity was basically formed by cesium radionuclides, measurements were carried out using a scintillation sensor with a crystal of NaI(Tl).

EXPERIMENTAL

Figure 1 illustrates the gamma-spectrum of air aerosol in the sample taken in June 1986 in the near-surface atmosphere of the Black Sea. The spectrum given does not display lines of short-lived isotopes (I-131, Te-132), whose concentration was too small to be accurately measured [1].

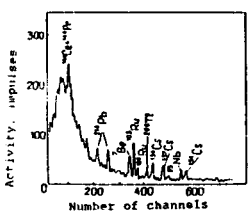


Fig. 1. Gamma-spectrum of the air aerosol of the Black Sea near-water atmospheric layer in June, 1986.

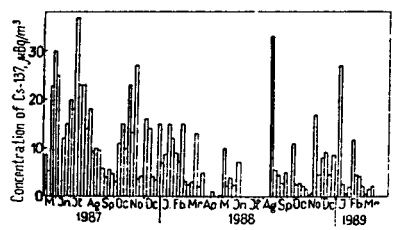


Fig. 2. The diagram of variations in Cs-137 and Cs-134 concentrations in the Black Sea near-water atmosphere (st. Katsiveli). May,1987-February, 1989.

Cesium radioisotope concentrations are given for the moment of sampling. The variation in concentration of cesium radionuclides in the near-water atmospheric layer during the investigations ranged from 1.0 to 5.5 mBq/m^3 for cesium-137 and from 0.62 to 2.79 mBq/m^3 for cesium-134. The mean value of the Cs-134/Cs-137 ratio reached 0.53±0.3 (the ratio of cesium isotopes, which was observed in the radionuclide flux in the area of the reactor accident (Buesseler, 1987). The root-mean-square deviation was 0.07, i.e., the variation coefficient equalled 13%, which was within the margin of error for nuclide concentration determination.

The peak values of cesium-137 and cesium-134 radioisotope concentration registered during the investigations were in the central Mediterranean Sea. In transition from the Mediterranean to the Black Sea, cesium radionuclide concentration in the near surface atmospheric layer had decreased. The largest values of radiocesium concentration obtained exceeded several times the identical values over the Mediterranean Sea in 1963, i.e., just after the nuclear tests in 1961-1962. During this period Cs-137 concentration ranged between 0.7 and 1.1 mBq/m^3 (The radioactive contamination, 1964). The total concentration of cesium-134 radioisotope, which were absent in the global fallouts, and cesium-137 concentration in the near-surface atmosphere of the Black Sea and the Mediterranean one month after the Chernobyl accident exceeded cesium-137 concentration documented during the most intensive nuclear weapon tests, by almost a factor. The peculiarity of the situation after the Chernobyl accident consisted in the fact that after the passage of the radioactive cloud, formed during the most intensive discharge of fission products from the reactor, a drastic decrease in radionuclides concentration in the atmosphere was observed over several areas. From the analysis of the data

presented in (Izrael et al., 1987) it may be deduced that radiocesium concentration in the nearwater atmosphere of the Black Sea during the transport of the radioactive cloud (May 1-8, 1986) reached a value of the order of 20-200 mBq/m^3 . The radiocesium concentrations measured at the end of May in the near-water atmosphere of the Black Sea were already lower than those measured in the Mediterranean Sea between 18-23 May 1986. The most likely explanation for this may be that the time between the transport of the radioactive cloud and the moment of sampling was longer for the Black Sea than for the Mediterranean.

For two years, beginning from May 1987, the radioactive cesium concentration in the near-water atmospheric layer of the Black Sea off the Crimea coast (stl. Katsiveli) was being contunuously measured. The averaging period was 7 days. Figure 2 gives a diagram of the variation of cesium-137 and cesium-134 concentration over the period of investigation. The range of variation of cesium-137 concentration was within 3.2-38 mkBq/m^3 ; and for cesium-134, it was 1.4-28 mkBq/m^3 . Thus, as compared to May 1986, cesium radionuclides concentration in the atmosphere decreased more than 100 times.

From the mean cesium-137 and cesium-134 concentrations one may trace a tendency for a decrease in radionuclide concentration in the near-water atmospheric layer. Cesium-134 concentration decreases faster than that of cesium-137. There are at least two reasons for it. The first reason is associated with the radioactive decay, as the cesium-134 half-life (T = 2.07 year) is much shorter than that of cesium-137 (T = 30.17 years). The second reason is the injection of cesium-137 accumulated in the stratosphere as a result of nuclear weapon tests. The lack of experimentally-determined concentrations of cesium-137 before the Chernobyl accident has not allowed us to determine the fraction of globally deposited cesium radionuclides in the nearwater atmospheric layer of the Black Sea. Regular observations of the concentration of man-made radionuclides in the near-water atmospheric layer the Black Sea region were carried out during the intensive nuclear weapon tests. In 1960, cesium-137 concentration was of the order of 10-100 mkBk/m^3 (The Radioactive contamination, 1964). After the Chernobyl accident, radiocesium concentration was 100-1000 times higher than the values mentioned above. In the spring of 1987, however, this fell to the level of 5-30 mkBq/m^3 . 'To estimate the ratio between global radiocesium in the atmosphere and that of Chernobyl, one can use the presence of cesium-134, the only source of which is Chernobyl. At the moment of the blow-out the Cs-134/Cs-137 ratio equated 0.53. Cesium-134 may be used as "Chernobyl" tracer only provided the behaviour of both radionuclides in the atmosphere is identical. The analysis of the observations conducted in 1986-1989 showed, however, that an intensive isotope fractionation took place there. Cesium-134 concentrations are given for May 1, 1986. Incidentally, in the absence of isotope fractionation, the radionuclides ratio remains constant. Actually, during the first two years after the Chernobyl accident, cesium-134 was mainly removed from the near-water atmospheric layer. As a result, the Cs-134/Cs-137 ratio dropped to the value of 0.05. The cause for isotope fractionation in the atmosphere, in our opinion, lies in the processes related to the formation of radioactive aerosols at the moment of condensation, as well as to the concentration of cesium -134 and cesium-137 on particles of various sizes. Some evidence of radionuclide distribution on aerosol particles of various sizes may be obtained from the data in (Iost et al., 1986). The measurements were accomplished in Switzerland but they may refer to other areas where the Chernobyl radionuclide fallouts were observed. The peak of I-131 activity is shown by particles with an overall size of 0.35 and 0.71 m for Te-132, Cs137 and Ru-103.

The measurements taken in Sweden, Finland and Germany (Devell et al., 1986, STUK-B-VALO, 1986; Lockhart et al., 1965) showed that 80 % of I-131 were in a gaseous phase. Cs-137, Te-132 and Ru-103, conversely, may have been released by the reactor in the form of particles or were readily sorbed on fine-sized aerosols, which coagulated then with other particles during transport. It can be appropriately supposed that some inorganic iones (sulphates, nitrates or ammonium) may have transported radionuclides over great distances. For example, Chernobyl-derived Cs-137 distribution, in agreement with the size of particles, proved to be similar to that of the nuclear test (Lockhart et al., 1965). Today, the absence of information on the distribution of cesium-134 in agreement with the size of aerosols deprives us of the possibility to consider the likely mechanisms for cesium radioisotopes fractionation in the near-water atmosphere of the Black Sea.

RADIOACTIVE CONTAMINATION IN BELARUS

After the accident on CNPP the concentration of Stontium-90 in the lower reaches of Pripyat river exceeded maximum permissible concentration of this radionuclide and was 4.0 10 Ci/l, but now does not exceed 1.10^{-12} Ci/l. Caesium -137 loss from the most contamined spot has descraesed from 280 Ci/year in 1987 to 20 Ci/ year in 1994.Now the levels of water systems contamiation are determined by the secondary processes: excange with ground sediments, washing of radionuclides from a surface of the river water header, but also at the expense of thawed and spring flood waters.

In the Republic of Belarus, 23% of the territory (3,666 settlements) have been contaminated by radiation with a density of Caesium -137 contamination of more than 1 Ci/sq. km [2].

The highest density of Caesium -137 contamination, exsept for a zone of settling out, is revealed in Chericovsky region of Mogilev oblast (146 Ci/ sq.km). There were a total of settlements in a zone of primary settling out (more than 40 Ci/ sq.km with the Caesium 137). The area of agricultural territories

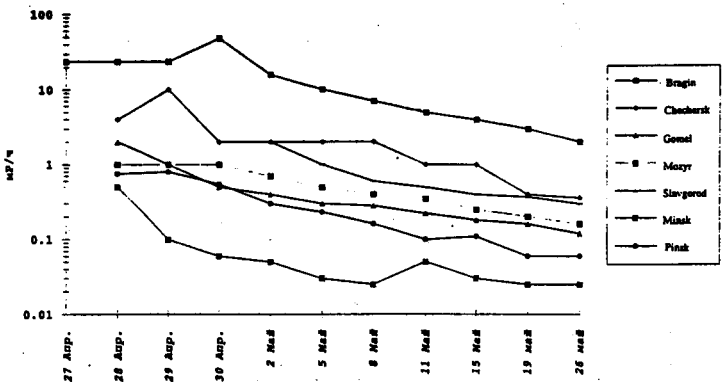
contaminated by radioactive Caesium with densities more than 1 Ci/ sq.km equals 1.6 mln. hectares. About thousand hectares of fofest are contaminated by radionuclides.

The contamination by Strontium -90 has more local character. The sites with the contamination levels of 3 Ci/ sq. km are found out in Hoiniky region, particular sites with densites 2-3 Ci/sq.km are found in Hoinikovsky, Braginsky, Dobrushsky and Vetcovsky regions of Gomel oblast. The soil contamination by Plutonium-238, -239,-240 higher than 0,1 Ci/ sq.km is revealed basically in a zone settling out, particular spots are presented in a number of regions in the Gomel oblast.

RADIOACTIVE AIR CONTAMINATION IN BELARUS

The monthly average values of aerosol radioaktivity concentration in regional centres in 1994 (Caesium-137) canged from 3.10^{-6} Bq/cub. m up to 46.10^{-6} Bq/ cub.m. The monthly average value of Strontium -90 in aerosol samples in 1994 was increased from $0{,}01.10^{-5}$ Bq/cub.m up to $0{,}24.10^{-6}$ Bq/ cub.m in regional centres. Radionuclide concentration in a lower statum of atmosphere is determined by the content of dust in the air of sities, i.c. by the processes of secondary wind rise and radionuclide transfer. Gamma radiation exposure dose rate is daily measured and the level of the radioactive precipitates from the ground statum of the atmqsphere is monitored on 22 stations, the measurement of gamma-radiation from radionuclides,Strontium-90 and Plutonium is carried out using samples of precipitates and aerosols.

Fig. 3. The levels of gamma-radiation exposure dose-rates from 28 April to the end of May, 1986



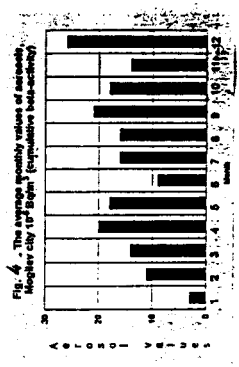


Fig. 4 - The average monthly values of aerosols, Mogilev city 10⁻⁴ Bq/m³ (cumulative beta-activity)

THEORETICAL INVESTIGATION

The migration of natural and artifical radioactive isotopes in the turbulent atmosphere and stratified lake and sea are studied. The equations global and vertical transfer of radionuclides in the turbulent boundary layers with classical and generalised boundary conditions are presented.

Mathematical models,numerical algoritms and computer programs for numerical solutions of diffusion in atmosphere, lake and sea are presented.

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SPATIAL DISTRIBUTION OF AN INSTANTANEOUS POINT RELEASE OF A FINITE AMOUNT OF GAS.

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Eduardo Grassi, Enrique Ponce.

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The risk to human health from exposure to vapors can be assessed by first quantifying an individual's potential exposure to, or dose of, the toxicant of concern. This is followed by a comparison of the exposure with an ascribed hazard determined from a known exposure-response relationship or exposure limit.

Occupational exposures to chemicals occur as a result of inhalation of workroom air contaminated with the vapor or aerosol of the chemical. The first step in the dose assessment process usually involves a determination of the concentration of the toxicant in the air that might be inhaled by the worker. Second, an estimation is made as to how much contaminated air would be inhaled. Next, the amount of toxicant potentially absorbed into the body, transported to potential target organs, and available for physiological response is estimated. The final step in the total assessment process is to compare the estimated potential human dose with the dose response from a toxicological/radiological data base. In the case of inhalation of a toxicant with an established exposure limit, the process needs only to determine the concentration in the workroom air and actual exposure time. Comparison of the workroom air concentration with an established limit,often is considered an adequate inhalation risk assessment.

A simple but usually very conservative approach to inhalation exposure assessment is to calculate the concentration of contaminant in air. The indoor air pollution model (also known as general ventilation model) describes the concentration of contaminant in the workroom air as a balance between the total input of contaminant, and the removal of contaminant from the workroom air by general ventilation with mixing volumes of clean air and the loss of contaminant from the combined sources of absorption, adsorption chemical transformation and radioactive decay.

In addition to the above temporal elements, the modeler also must deal with spatial factors that impact on workplace concentrations of airborne toxicants. It is intuitively obvious to most observers that the aiborne concentrations around point sources in large industrial rooms are much higher near the source, than at more distant locations in the room.In consecuence we can find a gradient of airborne concentration versus distance from the source.

This gradient is inversely proportional to air movement in the room, which in turn is a function of temperature gradients, movement of physical objects within the room and ventilation rate.

A rational approach would be to determine or estimate the rate of contaminant diffusion in a room.

This information is used to determine the subsequent dimensions of a volume around the source in wich the majority of the contaminant concentration would be found relative to the time frame choosed. This conceptual model is the affected volume, and it assumes omnidirectional (random) air movement at any point in the room. Its dimensions are determined by the manner in wich the air contaminant is dispersed from the source.

Diffusion theory is based on the random motion of the diffusing molecules resulting in net movement of the molecules from areas of high concentration into areas of low concentration. The rate at wich the concentration of diffusing particles change with time is equal to the flow into the

region less the flow out. Each flow depends on the flux of particles, or flow per unit area. The flux of particles in a fluid from point to point is directly proportional to the difference in the concentration of the particles, at the two points.

The coefficient of proportionality is called the diffusion coefficient, or D and its value must be determined experimentally.

The relation between flux and the rate of change in the concentration is known as Fick's Law. The mathematical formulation of this physical state of affairs leads to a differential equation called the diffusion equation. The solution is a mathematical expression that gives the concentration of the diffusing substance at every point in the space for every time. For example, if the diffusing substance is initially concentrated on the surface of a permeable membrane that divides a box in two at its center, the solution of the diffusion equation is a family of bell-shaped curves. The center of each curve coincides with the center of the box and as time passes the curve becomes broader and flatter.

There is another way to interpret each bell-shaped curve. Every point on the curve can be considered the probability density for the diffusion of a single Brownian particle from the central membrane in the box. The probability density multiplied by an appropriate measure gives rise to a probability. For the bell shaped curve the appropriate measure is a lenght, it is the distance between two points along the horizontal axis of the graph. The product of that distance and the average height of the bell-shaped curve in the interval between the two points is a probability.

The probability that the Brownian particle will be found in a region of the affected volume at a given time is the area under one of the bell-shaped curves between two vertical lines. Each vertical line passes through one of the two points on the horizontal axis that correspond to the boundaries of the given region in the box. According to this interpretation, the solution of the diffusion equation is not an expression that gives the distribution of molecule concentrations, instead the solution is a probability distribution. A convenient probabilistic measure of the displacement of a Brownian molecule is the root-mean-square displacement, or R.M.S. displacement. The probability that a Brownian particle diffuses at most as far as the R.M.S. displacement away from the central membrane in the box is 0,68; the probability that it travels more than twice as far as the R.M.S. displacement is less than 0.5.

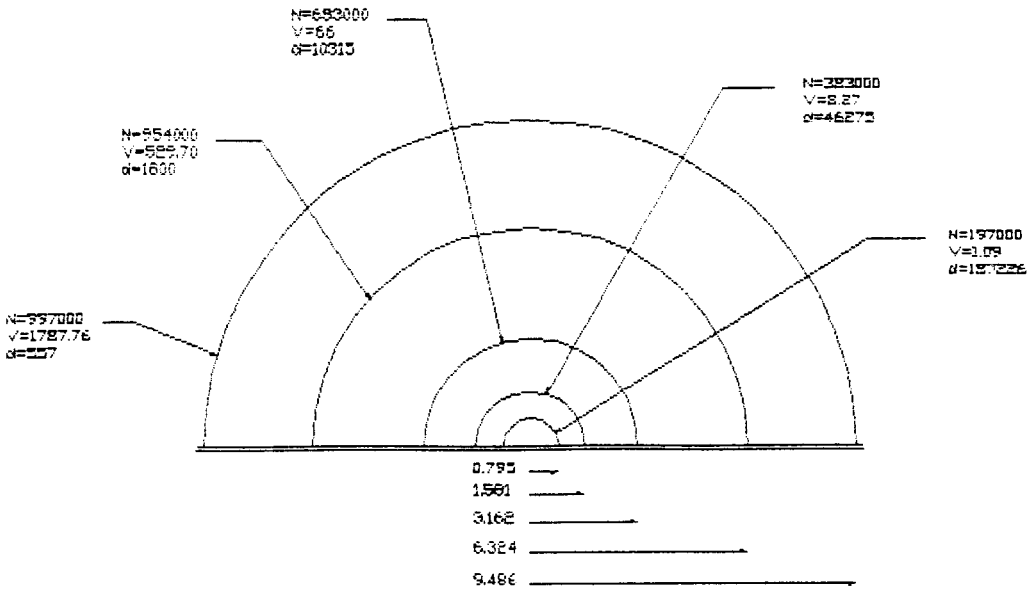
The R.M.S. displacement of a Brownian molecule diffusing away from the membrane in the box is $(2.D.t)^{1/2}$ where D is the diffusion coefficient and t is the time. Thus if a particle diffuses on the average one centimeter in one second, it will require four seconds to diffuse two centimeters and nine seconds to difusse three centimeters. So we can conclude that the radial displacement of a particle diffusing in any direction away from a central point "is not proportional to time, but proportional to the square root of time".

Consider now the instantaneous release of 1.000.000 μCi of a gaseous contaminant from a point source in the center of a large industrial room that has 12 mixing changes of air/hour or 5 minutes/mixing air change. Given 5 min. in an eddy diffusion field (D = 1 m^2/min), the R.M.S. distance from the source containing 68% of the diffusing molecules is calculated from:

$$R.M.S.= (2xDxt)^{1/2} = (2x1 m^2/minx5min)^{1/2} = (10\ m^2)^{1/2} = 3,162\ m$$

Thus, approximately 68% of the gas will be within one R.M.S. distance or a 6,324 m diameter sphere. One would find 95% of the gas within 2 R.M.S. distance or within a 12,648 m diameter sphere with the source at the center.The following table was made by calculating the volume and average concentration from the normal probability density for concentric spheres with radii of 1/4, 1/2, 1, 2 and 3 R.M.S. It shows the average concentration of the various volumes around the source, 5 min. after instantaneous release of 1.000.000 "molecules" of gas.

R.M.S.	Ø 1/2 Sph.	Volume(m^3)	Nø Molecules	C=Nø/Volume
1/4= 0,790	1,581	1,0	197.000	187.226
1/2= 1,581	3,162	8,2	383.000	46.275
1 = 3,162	6,324	66,2	683.000	10.315
2 = 6,324	12,648	529,7	954.000	1.800
3 = 9,486	18,972	1787,7	997.000	557



Repeating this procedure for succesive periods of 5 min. we can see how the affected volumes that contains the majority of the contaminant molecules is growing with time, and in consequence diminishing the contaminant concentration (mass/unit volume)

As mentioned above, actual worker exposure or delivered dose results from the time-weighted integration of contaminant concentration and uptake rate. Obviously, concentration modeling alone will not suffice to answer completely questions of how much toxicant was delivered. Ultimately the models or combination of monitoring/modeling must render an acceptable estimation of dose and be validated against measured worker exposure data in various occupational settings.

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IRPA9
1996 International Congress on Radiation Protection
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Vienna, Austria

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PAPER TITLE MODELING OF THE ATMOSPHERIC TRANS
RADIOACTIVE CONTAMINATION FROM CHERNOBYL ACCIDE
USING ETA MODEL

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MAJOR SCIENTIFIC TOPIC NUMBER 4.2. (see page 7)

ABSTRACT (See instructions overleaf)

MODELING OF THE ATMOSPHERIC TRANSPORT OF RADIOACTIVE CONTAMINATION FROM CHERNOBYL ACCIDENT USING ETA MODEL

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The atmosphere is the main media to transporting and dispersing the radioactive and/or chemical contaminants in accidental situations. The atmospheric models can be used to simulate the transport of contaminants in typical aocidental cases and for realistic meteorological conditions. This report describes an approach in simulating of the Chernobyl accident and similar hypothetical cases. The study is based on an atmospheric model extended by the additional equation that is modeling the transport of a certain radioactive concentration. A step mountain synoptic model, called ETA model (well-known model for weather forecasting), is used to investigate the transport and deposition of radioactive material in the Chernobyl accidental zone. Calculation was done in two steps for five 48-hours forecast periods. The first step is consisted in calculating of meteorological fields over Europe using horizontal resolution of 0,5 degrees. The second step is based on integration the same version of ETA model. The model is completed by the new prognostic equation for contaminant, with high horizontal resolution (about 20 km). The meteorological fields obtained by first step are used for initial and boundary conditions. The results of calculations are discussed by comparison with measurements. It is demonstrated that the model can reproduce certain observed characteristics of deposited material at the earth's surface inside the Chernobyl accidental zone.

MEASUREMENTS AND MODELLING OF ^{137}Cs MIGRATION INTO VARIOUS TYPES OF SOIL

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INTRODUCTION

The behaviour and migration of long-lived radioisotopes is an important part of information in the prediction of the consequences of radioactive contamination in agricultural areas. Therefore, the investigation of radiocaesium migration in soil is a focus of several research all over the world (1,2).

The level of soil contamination in Central and Eastern Europe after the nuclear accident at Chernobyl has been high enough to study the vertical migration of radicaesium under natural conditions. In Hungary, ^{137}Cs measurements have been carried out regularly since the Chernobyl accident at several location to determine the long-term migration of radiocaesium in soil system (3).

In the project presented here, the distribution of ^{137}Cs activity concentration has been measured and analyzed over the period of 1987-1996. On the experimental data collected during these ten years a model has been developed which can be used for long-term prediction of radioisotope penetration.

SAMPLING AND SAMPLE PROCESSING

Approximately 30 * 30 cm^2 areas on plane, grassy, uncultivated fields are sampled at each of five sites. Seven layers extending from the surface down to a depth of 20 cm are excavated at each of 11 sampling times (12, 14, 17, 24, 29, 36, 41, 48, 53, 83 and 118 months) after the accident. The distances between the sampling spots of the same site never exceeded 10 m.

The samples are first dried at 105 °C, then homogenized and the grains exceeding 1.25 mm diameter are removed. After this processing activities of 400 cm^3 volumes are measured by HpGe semiconductor gamma spectrometry in Marinelli geometry.

SOIL CHARACTERISTICS

Five sites have been selected to represent five different soil types, characteristic in Hungary. Several measured soil parameters are given in the Table below.

Identification	black rendzina	humic sandy soil	meadow soil	typical Ramann brown forest soil	brown forest soil with clay illuviation.
Site	Ady-liget	Szada	Domony	Galgamácsa	Dobogókô
pH (KCl)	5.5	7.5	7.3	7.2	4.2
$CaCO_3$ (%)	-	0.5	7.2	4.2	-
org. matter (%)	3.9	1.3	4.6	3.1	5.1
cation exhange capacity (me/100g)	43.0	9.8	26.8	31.0	23.1
changeable K (me/100g)	0.6	0.04	1.12	1.65	2.72
density (g/cm^3)	1.14	1.70	1.45	1.24	1.13

ERROR SOURCES

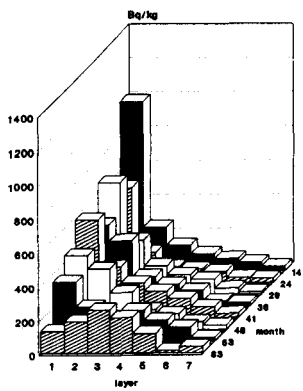
There are many sources of errors and uncertainties in the final results: short-range soil inhomogenities, the roughness of soil surface, the uncertainties in soil layer thicknesses, and errors of the spectrometric measurements.

To improve the error estimation at the last but one sampling three samples were taken and evaluated parallelly at each site. The standard deviations of the three sample sets are in the range of 25-40 %

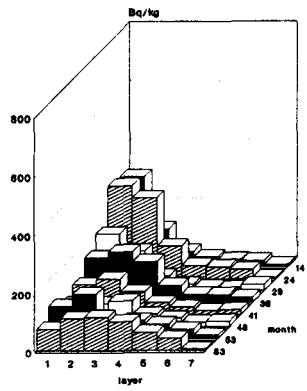
The effect of the sampling uncertainties are decreased by determining the actual weights of the layers, in g/cm^2, and using these values in the evaluation, rather than the nominal thicknesses in cm.

MEASURED CONCENTRATIONS

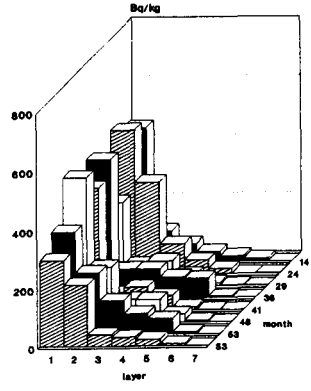
As an illustration, the measured activity concentrations are shown for three types of soil



brown forest soil with clay illuviation (Dobogókő)



meadow soil (Domony)



humic sandy soil (Szada)

MIGRATION MODEL

The physical and chemical processes of the migration are assumed to follow the diffusion-convection model. Let the initial surface concentrations from the Chernobyl and nuclear test fall-outs be a_{Ch} and a_{nt}, respectively, then the concentration a at depth z, at time t after the Chernobyl accident is

$$a(z,t) = a_{Ch}\exp[-\lambda t]\frac{1}{2[\pi Dt]^{1/2}}\exp\{-\frac{[z-ut]^2}{4Dt}\} + a_{nt}\exp[-\lambda(t+\tau)]\frac{1}{2[\pi D(t+\tau)]^{1/2}}\exp\{-\frac{[z-u(t+\tau)]^2}{4D(t+\tau)}\},$$

where
- $a(z,t)$ is the activity concentration (Bq/cm^3) at time t and depth z,
- a_{Ch} is the Chernobyl fall-out on the soil surface (Bq/cm^2),
- a_{nt} is the fall-out of the nuclear weapon tests on the soil surface (Bq/cm^2),
- D is the diffusion coefficient (cm^2/month),
- u is the convection velocity (cm/month),
- λ is the decay constant of ^{137}Cs (1/month),
- τ is the average time between the nuclear tests and the Chernobyl accident (set to 300 months).

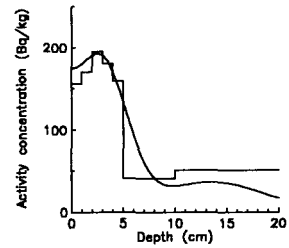
Furthermore, a total reflection at the air-ground interface is assumed.

The best fits of the initial surface activities, the convection velocities and the diffusion coefficients obtained by applying a Monte Carlo technique are given in the next table.

Identification	black rendzina	humic sandy soil	meadow soil	typical Ramann brown forest soil	brown forest soil with clay illuviation
Site	Ady-liget	Szada	Domony	Galgamácsa	Dobogókô
a_{Ch} (Bq/cm^2)	0.76	0.61	0.53	0.70	1.43
a_{nt} (Bq/cm^2)	0.61	0.84	0.76	0.68	1.14
u (cm/y)	0.082	0.042	0.220	0.106	0.424
D (cm^2/y)	0.343	0.053	0.400	0.041	0.433

According to the best fits the ratio of the maximum and minimum convection velocities is about 10, and the diffusion coefficients vary also within a factor of about 10.

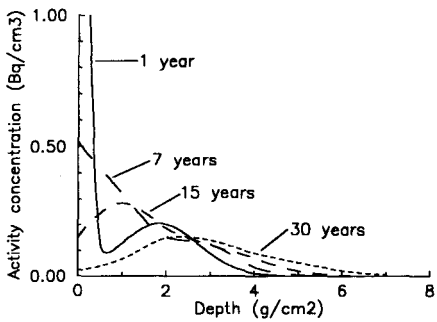
As an illustration, the measured activity concentrations (hystogram) and the fitted depth profile (continuous curve) for brown forest soil with clay illuvation (Dobogókô) are given in the figure for 83 months after the accident.



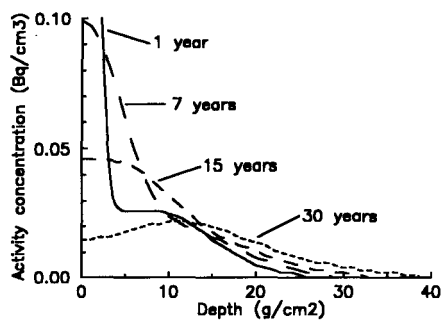
The goodness of fit is somehow characterized by the fact that the initial surface activities are in the range of 0.6-0.8 Bq/cm^2, both for Chernobyl and the nuclear tests, as expected form earlier measurements at these sites around Budapest, i.e. at one of the most contaminated areas of Hungary. Both values are slightly but significantly higher at Dobogókô, where samples were taken at a site with higher precipitation.

MODEL PREDICTIONS

Assuming the correctness of the model for longer time periods, predictions can be made for future times. Activity concentration curves computed for 15 and 30 years are given in the next figures for two types of soil.



Humic sandy soil (Szada)



Meadow soil (Domony)

CONCLUSIONS

The penetration of caesium into the soil is a very slow process, the majority of the activities is still in the top 1-5 cm layer.

According to the measurements carried out in the first ten years following the Chernobyl accident the migration can well be modelled by the diffusion-convection model. Since the uncertainties are relatively large, further measurement series are required to confirm the validity of the model and to improve the parameter estimations.

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ADAPTABILITY OF THE MATHEMATICAL MODEL IN ^{137}Cs LEACH-RATE AND EFFECTIVE COEFFICIENT OF DIFFUSION DETERMINATION FROM THE MORTAR MATRICES

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ABSTRACT

Mortar materials were used in the radioactive waste liquid effluent, containing ^{137}Cs, immobilization processes. As a matrix material, two types of the mortar formulation were used: the first one, prepared in ordinary manner, where the aggregates used were stone particles of average diameter of 2 mm, and the second one, where the polyethylene of low density granulates of average diameter of 2 mm were used. Both sample groups were introduces on leaching in aim to determine their physico-chemical characteristics, mainly ^{137}Cs leach-rates and indirectly coefficients of inspected radionuclide effective diffusion. Leaching tests were performed for 200 days. Measured ^{137}Cs leach-rates were the base parameter in mathematical modelling of the leaching process. From the postulated differential equation, that represents the leaching process, radionuclide diffusion coefficients, as well as deposition coefficients inside the mortar matrices were calculated, using programme packages for nonlinear fitting. When these coefficients were calculated for the samples of the finite geometry, with the sample's surface completely open to contact leachant, e.g., distilled water, numeric values were returned in the mathematical model, and the cumulative leached fractions of the radionuclide, that represents the matrix behaviour due to leaching in the prolonged time periods, were obtained. Differences between the ^{137}Cs leach-rate values, obtained experimentally and by mathematical modelling of the leaching process, were noticed. Mathematical model underestimates the leach-rate values until the completed hydratation of the cement in the mortar matrix and overestimates leach-rate values after the cement hydratation was done. These differences could be explained by adopted approximations, when postulating the mathematical model of the leaching process, neglected saturation concentration of the radionuclide inside the matrix and variations of the deposition coefficients of the radionuclide concerning two types of applied granulates in mortar formulations. Nevertheless, these differences do not exceed 5 percents for the leach-rate values, which permits to postulate opinion on mortar-radwaste matrix behaviour in the prolonged periods of time.

INTRODUCTION

Mortar was used as a matrix material in an immobilization process of radioactive waste material, $^{137}CsCl$ solution. Low density polyethylene, PELD, was used as granulates in preparing the mortar matrices, parallel with the preparation of the matrix mixture samples that were made with stone granulates. PELD was in a granulate form of 2 mm diameter. PELD was used instead of ordinary used granulates, usually little stone particles, in aim to decrease porosity and density of the mortar matrix and to avoid phenomenon of stone particles segregation on the bottom side of the immobilized radwaste cylindric form [1]. Satisfying this aim, we tried to obtain better physico-chemical characteristics of the radwaste mortar mixture forms, especially radionuclide leach-rates. Quality of the final radwaste form should be discussed due to: a radionuclide immobilized in the mortar matrices leach-rates, as well as to the appropriate coefficient of diffusion. Calculating the radionuclide coefficient of diffusion for the investigated mortar matrices, made either in ordinary manner or with PELD granulates, it is possible to predict behaviour of the final radwaste form in the prolonged periods of disposal time, considering the most undesirable disposal conditions, e.g., flooding of the disposal site. When discussing the phenomena that have influence on the radionuclide leach-rate from the concrete matrices: diffusion, dissolution of the matrix and deposition of the radionuclides on the pore walls inside the matrix, the assumption is made that the radionuclides are leached from the surface and interior of the cylindrically shaped sample geometry. In aim to simplify a mathematical modelling of the leaching process of ^{137}Cs from the orthocylindrical sample, we made an assumption that the concentration of the examined ionic type is uniform in every segment of space and time, and equal to the saturation concentration of the ionic form. In this manner, the dissolution factor could be neglected in leaching process consideration, and postulating the following differential equation, Equation 1., which describes the processes that occur inside the matrix material, and whose solutions, for the defined initial and boundary conditions for finite cylindrical geometry, give the radionuclide leach-rate from the matrix-radwaste mixture form [2].

$$\frac{\partial C}{\partial t} = D \cdot \left(\frac{\partial^2 C}{\partial r^2} + \frac{1}{r} \frac{\partial C}{\partial r} + \frac{\partial^2 C}{\partial z^2} \right) - \delta C \qquad (1)$$

where: C-radionuclide concentration in prospected time of experiment in the leachant volume (mol), δ-radionuclide deposition velocity constant (1/s) and D-diffusion coefficient (cm^2/s).
Solution of the Equation 1., gives the possibility of calculating the cumulative ^{137}Cs leached fraction from the sample of finite geometry. Experimentally obtained values of the cumulative leached fraction of the ^{137}Cs were

fitted on adopted mathematical model of the leaching process, which is solution of the Equation 1. Modified programme packages for nonlinear fitting, using method of the least squares, were used, with the parameters of effective diffusion and deposition. Using PELD as granulates in the radwaste mortar mixture, weight of the final, full size immobilized monolith would be significantly lower than one, prepared with stone granulates.

EXPERIMENTAL

Mortar-radwaste mixture samples, that were used in leaching processes, as well as in mechanical strength characteristics examinations, were made using mixture formulation: cement (PC-20 MPa), either stone particle aggregates (0-2 mm) or PELD granulates of 2 mm diameter, mixing additives and water. PELD granulates, have following properties, that underlined their addition in the mortar matrix formulation: density of 0.940 g/cm^3, dilution index of 0.3 g/10 min., breakable temperature of $T_b \cong$ -120°C, resistance on aggressive liquid environment and influence of mechanical forces, e.g., 30 kp/cm^2 under temperature of 60°C. The water used for mortar preparation was doped with ^{137}CsCl solution (pH=1.2). A water-to-cement ratio, W/C, of 0.36 provides density of 2.145 g/cm^3 for mortar matrices prepared with stone granulates and density, from 1.953 to 1.960 g/cm^3 for the mortar matrices prepared with PELD. The mixing period for a 3 litre mortar batch was ten minutes, using a planetary mixing device. Radioactive waste mortar mixtures were poured into plastic molds to harden. After one day, the samples were taken out from the molds and cured in an atmosphere of 65 percents relative humidity and T=20°C for a period of 28 days [3]. Hardened samples all have approximately the same level of bound activity ($A_0 \approx$ 10 kBq). In these investigations, matrix formulation was not optimized, because of the high water-to-cement ratio, W/C and absence of radionuclide sorbents in the experimental matrix formulation. The composition used in the experiments was design to allow easy leaching of ^{137}Cs from the mortar monolith. Solidified orthocylinder shaped samples, H=4.5 cm, with the exposed surface area completely open to contact leachant medium, were used in prospecting physico-chemical characteristics of mortar-radwaste mixture forms. After curing period, samples were placed inside clear plastic beakers, each containing leachant media and introduced to leaching. The leachant media used was distilled water (pH=6.2). Experiments were carried out at a room temperature of T=20 ± 1°C. The leachants were renewed periodically: 5 times in the first month and approximately once per month in the first 200 days, in order to obtain longer periods in which leachants remained in contact with the specimens [3,4]. Radionuclide ^{137}Cs presence in the leachants were measured by a gamma-spectrometer.

RESULTS AND DISCUSSION

Process of radionuclide leaching from the mortar matrices, prepared in described manner, is divided in two stages. The first stage is the initial "wash-out" period, when the radionuclides, immobilized in matrix material, are "washed-out" from the radwaste-mortar monolith surface. Duration of this stage, when the leachants are frequently exchanged, is up to hundred days. The second stage of radionuclide leaching from the matrix material, in which the most of radionuclides are structurally bonded, is a relatively steady-state process [1,2,3]. Measured leach-rate of the inspected radionuclide,^{137}Cs, from the radwaste mixtures prepared with PELD was the same level of order, but significantly lower than ones for the monoliths prepared in ordinary manner. For each treated matrix formulation, effective coefficients of diffusion of radionuclide incorporated in mortar matrix were calculated, presented in the Table 1.

Table 1.-^{137}Cs Diffusion and deposition coefficients for different matrix compositions after 200 days of leaching.

Characteristic	Matrix with stone granulate		Matrix with PELD granulates	
	Experimental	Math. model	Experimental	Math. model
$\frac{\sum a}{A_0}$ (%)	4.51	4.71	3.51	3.69
D_e x 10^9 (cm^2/s)	1.481		1.088	
δ x 10^7 (1/s)	1.414		1.800	

Parameters obtained in this manner, were returned into the appropriate mathematical model and used in calculation of the ^{137}Cs cumulative fraction that could be leached in postulated leaching experiments[2]. In such manner, behaviour of the radwaste-mortar matrix mixture in the prolonged disposal period, under the most conservative and undesirable environmental conditions, could be predicted. Differences between ^{137}Cs cumulative

leached fractions experimentally obtained and those calculated by the mathematical modelling of the leaching process are neglectable, as it is shown at the Figure 1.

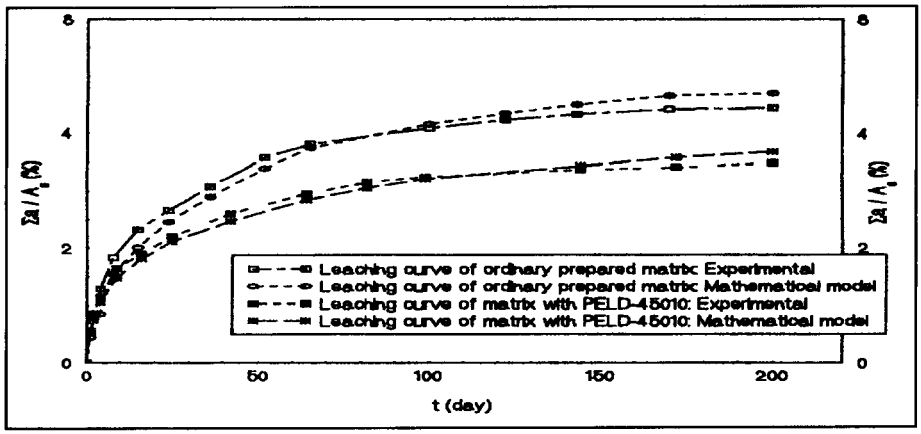

Figure 1. ^{137}Cs cumulative leached fraction from the mortar matrix prepared with PELD-45010 granulates and in ordinary manner,using stone granulates, obtained experimentally and by mathematical modelling of the leaching process.

Good homogeneity, relatively uniform distribution of the granulates in the radwaste-mortar matrix and low percent of the void space inside the bulk material, that could results from the segregation pathways of the stone granulates inside the matrix during matrix hardening and increasing the monolith porosity, as well as absence of the phenomenon of particle segregation on the bottom of the matrix material, may results in better physico-chemical characteristics of the radwaste-mortar matrices prepared with PELD. Taking into account that the full scale solidified radwaste forms prepared with PELD will have at least one fifth lower weight than monoliths prepared with stone granulates, advantages of the PELD granulates in the immobilization processes of the radwaste materials are obvious.

CONCLUSION

In our paper, we have shown that there is a possibility of improving physico-chemical characteristics of the radwaste-mortar mixture, by the PELD granulates addition in the mortar matrix, instead of the stone particles. When mathematical modelling the leaching process, differences between the ^{137}Cs leach-rate values, obtained experimentally and by mathematical model, were noticed. Mathematical model underestimates the radionuclide leach-rate values until the completed hydratation of the cement in the mortar matrix and overestimates leach-rate values after the cement hydratation was done. These differences could be explained by the influences of the adopted approximations, when postulating the mathematical model of the leaching process, neglected saturation concentration of the radionuclide inside the matrix and variations of the deposition coefficients of the radionuclide concerning two types of applied granulates in mortar formulations. Nevertheless, these differences do not exceed 5 percents for the leach-rate values, which permits to postulate opinion on mortar-radwaste matrix behaviour in the prolonged periods of time. Examinations considering optimization of the matrix formulations, by varying the different weight percentage of the PELD granulates presented in matrix formulation, in aim to increase mechanical strength and decrease radionuclide leach-rates should be performed.

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MATHEMATICAL MODELLING OF RADIONUCLIDE BEHAVIOUR IN THE VIENNE RIVER DURING ROUTINE RELEASES OF THE CIVAUX NUCLEAR POWER PLANT

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INTRODUCTION

The Civaux nuclear plant, currently under construction, is located on the Vienne river in west centre of France, a river with a minimal monthly average flow rate of 23 m^3/s. Special attention was given to the prediction of radioactive contamination of the river due to the routine discharge of liquid wastes at low flow during summer. To address this issue, a one-dimensional unsteady model, named CALIDO, was developed to simulate radionuclide transport along the 120 km stretch of the Vienne river, from Civaux to the confluence with the Loire river. It takes into account the processes governing sediment transport and radionuclide exchange between water and suspended matter. The model's aim is to assess the radiocontamination of the three main compartments of an hydraulic ecosystem, the water, the suspended matter and the bottom sediments.

THE CALIDO MODEL

CALIDO is a one dimensional non stationary model taking into account longitudinal dispersion. It calculates the distribution of activity concentration in water, suspended matter and bottom sediments as a function of time and space. With this aim in view, the river is divided into longitudinal sections according to the flow stream. The model calculates both the interactions between the environmental compartments and the linkage of the successive river sections as shown in figure n° 1.

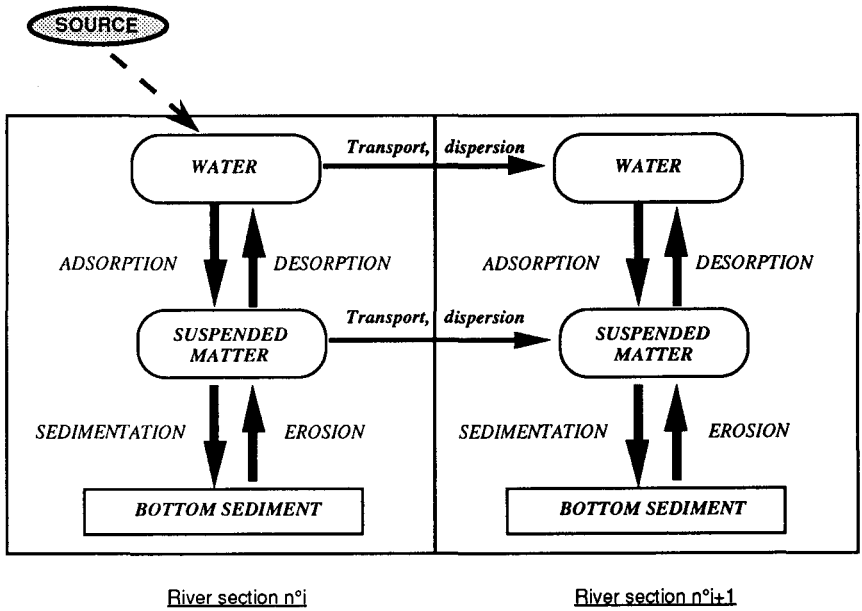

figure n°1 : Processes included in the model and linkage of successive river sections.

The global model is subdivided in three coupled submodels:
- The hydrodynamic submodel assesses river flow, depth and velocity in each section. We use a fully complete 1D hydraulic model named LIDO which solves free surface flow problems in rivers. LIDO requires a very accurate bathymetry to describe the natural form of the river. It calculates hydraulic parameters in sections of a few hundred meters long.
- The sediment model uses the results of the hydraulic model, it simulates the transport and the dispersion of suspended matter in water column. The processes of sedimentation and resuspension are modelized by the classical laws of Krones and Partheniades. Sedimentation and resuspension (which are regulated by two parameters: settling velocity and erosion rate) occur at constant critical shear stresses which have to be calibrated.

- The exchange model describes the exchanges of radionuclides between the dissolved and the particulate phases. The exchanges are modelized using a kinetic equation for a one step reversible reaction which leads to an equilibrium characterized by the distribution coefficient K_d.

CALIBRATION OF THE MODEL

The hydrodynamic submodel : the bathymetry of the Vienne river (data of main section profiles) was achieved and we performed several tracings of the river in order to calibrate the roughness coefficients of each river sections.

The sediment submodel : to obtain site-specific values of the four submodel parameters (settling velocity, erosion rate, critical deposition and erosion shear stress) a concentrated effort at field data collection downstream of Civaux was required :

- during the summer period, the concentration of the suspended matter in the river was measured every 8 hours at a few selected locations along the river. These measurements were used to calibrate the submodel by comparison between measured and calculated results. A field survey of bottom sediments was undertaken at low flow to gain information on the thickness and size distribution of the sediment layer. The vertical profile of the suspended matter was examined at a low velocity location of the river. It allowed us to calculate the settling velocity of particles. Samples were collected weekly for size distribution analysis (measured by laser beam diffraction, using a Coulter particle size analyser). Besides, the organic fraction of the suspended matter was distinguished from the mineral fraction. These analyses were undertaken in order to calibrate the submodel for different granulometric or qualitative classes of the suspended matter.

The exchange model : in order to determine K_d and kinetic parameters for each radionuclide values, we performed both in situ and laboratory experiments. The radioactivity associated with suspended matter was measured in the Loire river during scheduled discharges from low activity waste tanks of the Dampierre nuclear power plant. The obtained results were compared with laboratory experiments aiming at studying the behaviour of some selected radionuclides in Vienne river water.

EXAMPLES OF CALCULATIONS

The first example shows the evolution of activity in water and suspended matter after a constant release in stationary conditions. The term source is a constant release of ^{58}Co with a 100 Bq/m^3 activity over a period of 4 hours. The river flow rate is about 23 m^3/s and the contaminated wave is followed over a period of 3 days from the site of release, Civaux. The results are presented in figure 2.

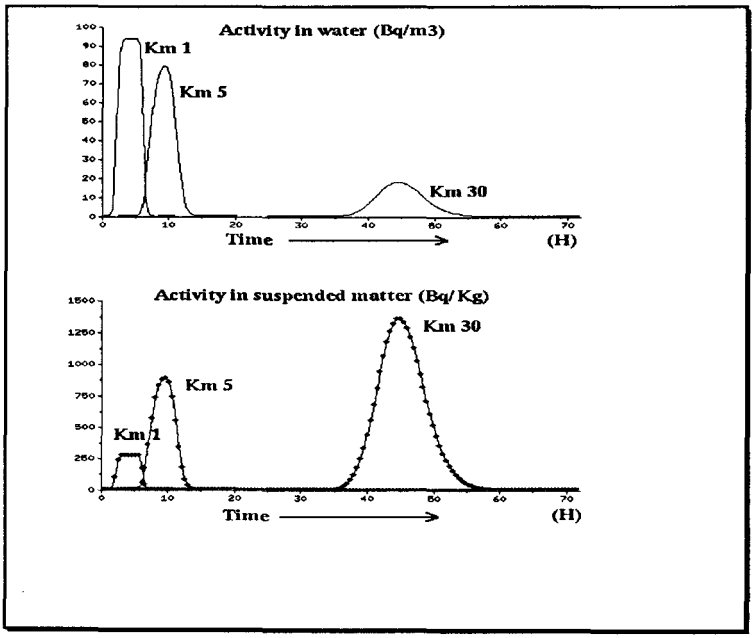

figure n°2: Calculated activities in water and suspended matter as a function of time presented at different distances from the site of release.

As a result of dispersion and adsorption, the concentration in water decreases and, simultaneously, the concentration in suspended matter increases.

The second example is the simulation of the routine functioning of the Civaux nuclear plant over the 1987 hydrologic conditions. Based on the daily river flow rate, a power plant effluent model simulates the daily release of radionuclides. The suspended matter concentrations at Civaux are also computed as a function of the daily river flow rate. Taking the example of ^{58}Co, one of the main released radionuclides, CALIDO model simulates its behaviour in the three compartments of the river during all the 1987 period. With such a kind of non stationary model integrating all the variability of meteorological conditions, it is possible to assess the radiocontamination of the river taking into account all the successive passings of contaminated waves. Figure n°3 focuses on the contamination of the bottom sediments at the end of summer 1987 between Civaux and the confluence of Creuse river. Expressed in Bq/m^2, the main contaminated areas are located just in front of major dams.

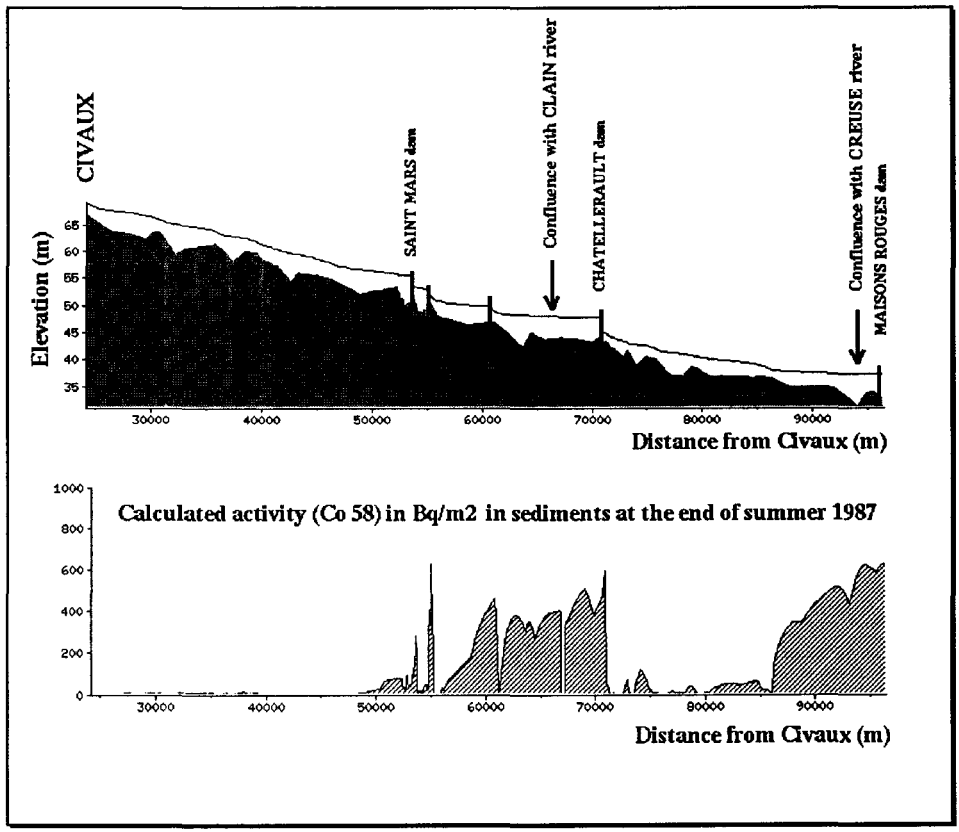

figure n° 3 : Calculated activity in sediments as a function of space at the end of summer 1987, after 9 months of simulated routine functioning of Civaux nuclear plant.

PROSPECTS

CALIDO is currently used with a single class of sediment. We are working at including different classes to integrate all the results of experiments.

The model will be coupled with other biological transfer models (transfers to biomasses) to assess the contamination of the upper level of the river food chain. The radiocontamination of the different compartments is a pathway to the contamination of man.

Properties of deterministic models for prediction of radionuclide concentrations in river systems

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Abstract

A deterministic model was used for predicting the activity concentration of radionuclides in rivers. The model was validated in the framework of VAMP, aquatic working group, river subgroup, where scenarios as Clinch-Tennessee rivers as well as Dnjepr river were provided. This was a good chance to test the predictive power of the model. Some of the results of this exercise are presented.

Description of the model

The model was developed by /Sc 76/ and modified to predict the activity concentration the river Danube/Pr 82/. It is a simple deterministic model, taking into account interaction with suspended material, radioactive decay and sedimentation. The model seems promising, because of the small numbers of parameters needed for evaluation.

The basic equation is :

$$C_{i,j}=C_{(i-1),j}\cdot W_{(i-1)}\frac{1}{W_i}\cdot e^{-(k'+\frac{\lambda}{v})x}(1-E_{i,j})\cdot\frac{1}{1+k_d S}$$

C_{ij}	activity concentration of radionuclide i at site j [Bq/m³]
Q_i	annual release of radionuclide [Bq/a]
K	conversion factor s→a = $3.17\cdot10^{-8}$ [a/s]
W_j	mean annual flow at a point j (m³/s)
k′	depletion factor for sedimentation (m-1)
λ	decay constant (s^{-1})
x	distance (m)
v	flow velocity (m/s)
K_d	K_d-factor (l/kg)
S	amount of suspended material (kg/l)

The scenarios are sufficiently described in the final report of VAMP (in press).

To apply this model to the scenarios some assumptions had to be made:

The discharge of radioactive effluents is continuous and instantaneous mixing with the river or reservoir water takes place.

The activity removed by biological materials can be neglected.

The river system is divided into several parts, representing the boxes of the model

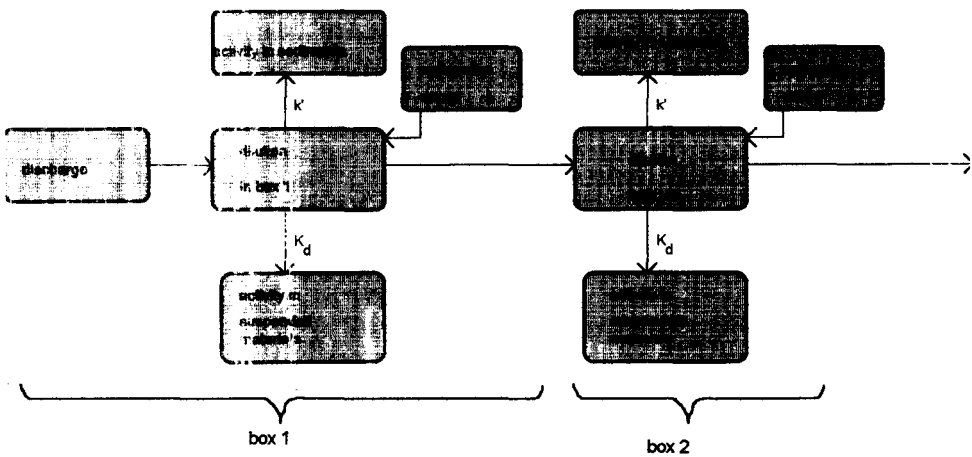
Some parameters as flow velocity, amount of suspended materials of the water body are constant in a box.

There is no resuspension of radioactive material absorbed by the sediments.

Reduction of radioactive materials due to chemical processes are not considered.

The cross section area of the river can be described by mean values of width and depth.

The processes simulated by the model and the mechanism are shown in figure 1:



Large river systems like the Clinch Tennessee rivers or the Dnjepr cascade have to be divided into several segments. The choice of the number of boxes is arbitrary, but the modelist must find segments in which the properties are well known and of interest. As a consequence of using a deterministic box model results, are not time dependent and also the transport of the activity concentration downstream will not be simulated correctly. The model was initially developed to provide yearly averaged values of the activity concentration. Modelling the Clinch Tennessee rivers the experience was made that the model led to reasonable results, when using adequate parameters When monthly averaged values had to be calculated, the model still led to satisfying results, as there are high flow velocities. Modelling the Dnjepr cascade monthly averaged values of the activity concentration were not in a good agreement with the experimental data. The were two reasons: the first was that the same parametrisation as for the Clinch Tennessee rivers could not be used, and the second reason was that the correlation between the calculated and the experimental data was not satisfying. The second one sets up on the big range of the reservoirs of the Dnjepr cascade (each reservoir is longer than 100 km) and on the very slow flow velocities. To improve the time dependence a new term for the dilution was used, which was based on a differential equation. The new term, describing dilution, considers how the activity concentration changes, when the inflow has a different activity concentration as the flowing off. As there was not enough hydrological data (e.g. reservoir depth in each month) the assumption was made , that the capacity of each reservoir was constant.

Choice of parameters

As mentioned above, choice of parameters is very important. The first approach for the Clinch Tennessee rivers was don by using "default parameters". K_d was taken from /IA82/, and the k' as suggested by /ME 79/. Using the same parameters for the Dnjepr cascade and comparing the results with the experimental data, the experience was made that especially for longer distances the calculated values underpredicted the measured values. A calibration with the monthly averaged data of the Dnjepr cascade was carried out and a new set of values was obtained. With this new set of parameters all values were recalculated, what led to an improvement in the quality of the results.

Sensitivity analysis

To identify the most important parameter and variables of the model, a sensitivity analysis was carried out. For each parameter and variable, a range of its value was determined. Regarding variables, the maximum and the minimum of the observed data were taken. For the parameters the maximum and the minimum found in the literature were taken. A sensitivity index SI /Ho 83/ was calculated. The model is sensitive for the parameter when SI is close to 1. The calculations show that the flow rate is a very important parameter for all radionuclides. In addition, for radionuclides as Cs-137, which strongly interact with suspended material and bottom sediments, the depletion factor for sedimentation and the amount of suspended materials are of even more importance.

Table 1: table of sensitivity indexes for Cs-137 and Sr-90, Kiev reservoir

	sensitivity index	
	Cs-137	Sr-90
flow velocity	0.0032	0.0040
flow rate	0.5055	0.6278
suspended material	0.9356	0.4990
Kd-factor	0.3764	0.0151
k'-factor	0.9047	0.5938

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MODELLING THE DYNAMICS OF FISH CONTAMINATION BY CHERNOBYL RADIOCAESIUM: AN ANALYTICAL SOLUTION BASED ON POTASSIUM MASS BALANCE

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INTRODUCTION

The Chernobyl nuclear accident provided a sudden contamination of many European water bodies with radiocaesium, followed by a rapid partial immobilization. As a result, activities and concentration factors of radiocaesium in biota fluctuated strongly over several years before reaching quasi-equilibrium, with patterns significantly differing between organisms. To model these dynamic relaxation processes, mass balance equations for ^{137}Cs in aquatic food chains were developed on the following basis: a) potassium acts as a biogeochemical equivalent of caesium; b) the concentration of potassium in fish and other biota is rather constant; c) the main source of potassium in freshwater fish is the dietary uptake.

MODEL DEVELOPMENT

The model describes the fate of a contamination pulse in the water and in associated food chains. The decline of the activity concentration in the water, which is considered as the secondary source of ^{137}Cs, can be well expressed by a two-component exponential function corresponding to a fast (**F**) and a slow (**S**) component in the immobilization of Cs:

$$\mathbf{A_W(t) = K_W^F \cdot \exp(-\lambda_F \cdot t) + K_W^S \cdot \exp(-\lambda_S \cdot t)}$$

where **t** is time, **A** are activity concentrations, **K** are transfer coefficients, and λ are rate constants.

The food chain considered here includes the following compartments: fish food organisms such as zooplankton and zoobenthos (**Z**), non-piscivorous fish (**NP**), and piscivorous fish (**P**). In every compartment the turnover rate of caesium is considered as constant over time. If potassium acts as a biogeochemical equivalent of caesium, the uptake of Cs in animals is proportional to the ratio of assimilation efficiencies of Cs and K, and the elimination of Cs is proportional to the ratio of turnover rates of Cs and K. Since homeostasis requires that uptake and elimination of K are equal, mass balance yields the following equation for the member **j** of a food chain:

$$\frac{d\mathbf{A_j}}{dt} + \lambda_j^{Cs} \cdot \mathbf{A_j(t)} = \mathbf{T_{j-1}^j} \cdot \mathbf{A_{j-1}(t)}; \quad \text{with} \quad \mathbf{T_{j-1}^j = k_{\lambda j} \cdot \lambda_j^{Cs} \cdot \frac{\alpha_j^{Cs}}{\alpha_j^K} \cdot \frac{C_j^K}{C_{j-1}^K}} \quad \text{and} \quad \mathbf{k_{\lambda j} \equiv \frac{\lambda_j^K}{\lambda_j^{Cs}}}$$

where **A** are radiocaesium activity concentrations, **C** are potassium concentrations, **T** is a complex transfer coefficient, α are assimilation efficiencies, and λ are first order decay rate constants describing the biotic elimination (turnover) of Cs.

An analytical solution of this equation is given by:

$$\mathbf{A}_j(t) = \exp(-\lambda_j^{Cs} \cdot t) \cdot \left[T_{j-1}^{j} \cdot \int \mathbf{A}_{j-1}(t) \cdot \exp(\lambda_j^{Cs} \cdot t)\, dt + Const \right]$$

where *Const* is determined by initial conditions:

$$Const = \left[\mathbf{A}_j\right]_{t=0} - \left[T_{j-1}^{j} \cdot \int \mathbf{A}_{j-1}(t) \cdot \exp(\lambda_j \cdot t)\, dt\right]_{t=0}.$$

When applying this model to a food chain after combination with the water model, solving the equations adds one exponential term for each compartment. For piscivorous fish, the following equation is obtained:

$$\mathbf{A}_P(t) = T_W^P \cdot \left[K_P^F \cdot \exp(-\lambda_F \cdot t) + K_P^S \cdot \exp(-\lambda_S \cdot t) - K_P^Z \cdot \exp(-\lambda_Z \cdot t) - K_P^{NP} \cdot \exp(-\lambda_{NP} \cdot t) - K_P^P \cdot \exp(-\lambda_P \cdot t) \right]$$

where T_W^P is the bioconcentration factor and $\mathbf{K}$ are complex transfer coefficients.

MODEL VALIDATION

The model was tested with post-Chernobyl data from long-term studies in different European lakes [1-6]. For each compartment, a transfer coefficient ($\mathbf{K}$ or α) and a rate constant λ was fitted iteratively to the data. An example from one lake is shown in Fig. 1. The model provided a good description of several important aspects: a) the differences in contamination levels and contamination dynamics between predatory and non-predatory species (trophic level effects); b) the dependence of ^{137}Cs activity in fish on the potassium concentration in the water (bioavailability); c) time scales of contamination and depuration in different fish species (prognostic tool). Evaluation also showed a significant effect of growth rate on the turnover rate (biological half-life) of ^{137}Cs in fish.

CONCLUSIONS

The complex contamination dynamics in aquatic food chains after an accidental release of radiocaesium can be adequately described with a physiological model, using equations that can be solved analytically. These solutions describe the ^{137}Cs activity concentration in every compartment as a series of exponential functions, of which some are derived from the concentration in the water (source pattern), and the others determined by the caesium turnover rate in each food chain compartment.

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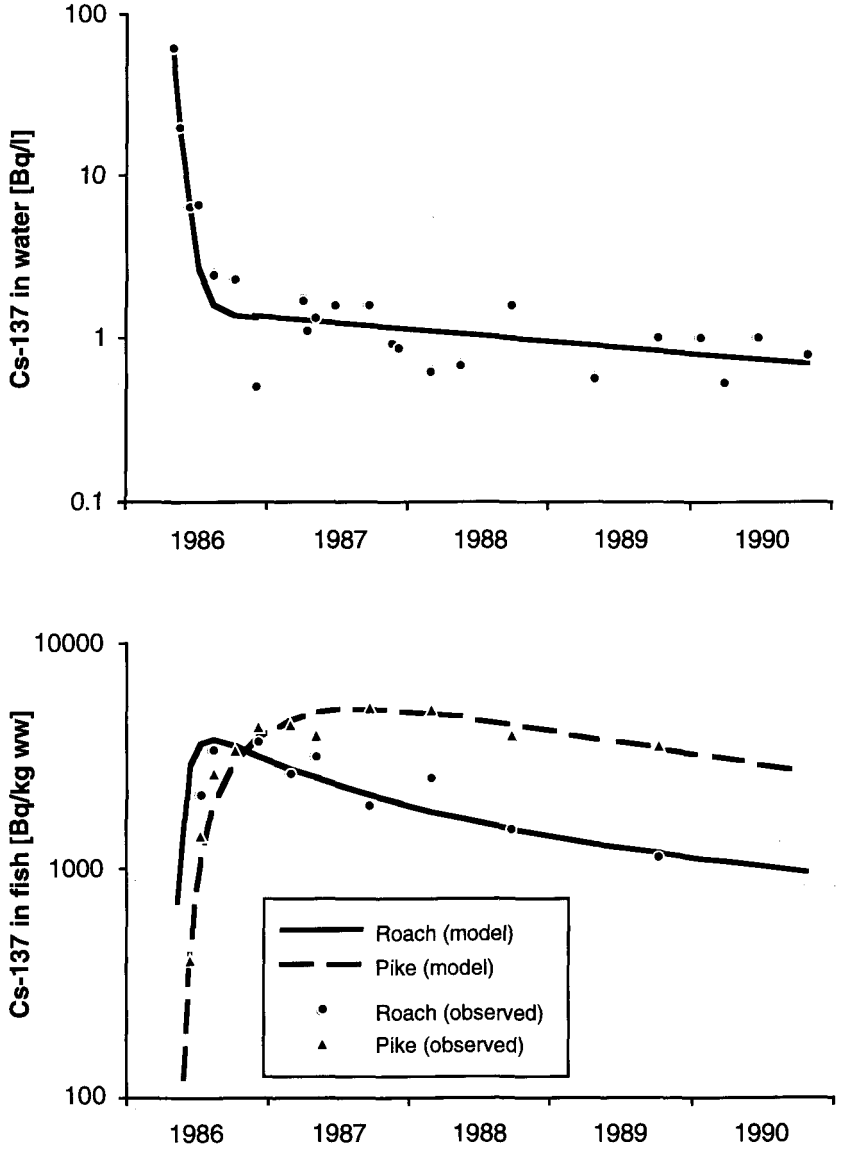

Figure 1. Observed data [1] and modelled values of ^{137}Cs activity concentrations in the water and in different fish species in lake Hillesjön (Sweden) after the Chernobyl fallout in 1986.

MODEL VALIDATION : THE USE OF HUMAN MEASUREMENT DATA

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ABSTRACT

Measurement of radionuclides in individuals provides the most direct method of evaluating the results of modelling procedures used to estimate the uptake and distribution of radionuclides in humans, for both environmental and occupational exposure. This paper describes recent and on-going comparisons of human measurements with model predictions and considers the benefits and difficulties associated with this approach to model validation.

INTRODUCTION

One method which can be adopted to establish the adequacy of the results of modelling procedures used to estimate radionuclide uptake by individuals, is to compare predicted radionuclide contents with measurement data obtained for the population under examination. This paper summarises two studies where such comparisons have been undertaken. In the first, measurement data for inhabitants of Seascale, a village close to the UK Sellafield nuclear reprocessing plant, were compared with model predictions based on methodology utilised to assess the risks of radiation-induced leukaemia and other cancers in children and young persons living in the village[1]. A second study is also in progress to compare plutonium concentrations measured in individuals resulting from atmospheric weapons testing with predicted values based on current ICRP models.

SEASCALE COMPARISON

Human measurement data for the Seascale population are available in the form of radionuclide concentrations in tissues taken at autopsy (plutonium-239+240 and caesium-137), measurements of plutonium in fetal tissues and whole body radiocaesium measurements. The assessment methodology used to determine the radionuclide uptake, and thereby the internal doses received by the inhabitants of the village, was based on estimations of the annual inhalation and ingestion intakes of a range of radionuclides, in association with the most recent dosimetric and biokinetic models. These calculated intakes were derived, as far as possible, from direct measurements of radionuclides in environmental materials, air concentration measurements and habit data. Where measurement data were insufficient, intakes were estimated using a combination of discharge data and environmental transfer models.

The comparison of measured and predicted whole body radiocaesium levels for the period 1957 - 1989 is given in Figure 1 and shows that the radiocaesium contents of individuals are generally in good agreement with the mean radiocaesium contents measured in groups of subjects prior to 1969. The overestimation in whole-body contents for the late 1970s and 1980s is probably due to the pessimistic assumption that all fish consumed by the Seascale residents was locally caught: consumption of foodstuffs of marine origin providing a large contribution to the radiation dose received during these years. An increase in both predicted and measured values can be observed for 1986, the year of the Chernobyl accident.

The autopsy data detailed in Table 1 suggests that the modelling procedure tends to overestimate radionuclide concentrations in the skeleton and liver, but leads to underestimates of the concentrations of plutonium-239+240 in the lung. The reason for this discrepancy is unclear but, may be due to the assumption made in the modelling procedure that the absorption from lung to blood of all inhaled plutonium compounds could be characterised as Type M (moderate absorption rate) as defined in the current ICRP Human Respiratory Tract Model[2]. In reality, it is more likely that the inhaled plutonium would have behaviour intermediate between Type M and Type S (slow absorption rate), which would result in a greater retention in the lung and lower concentrations in the liver and skeleton. The default absorption parameters for Type M material were used in the modelling procedure since this is the more cautious assumption with regard to dose to the red bone marrow, the tissue thought to be at risk when considering leukaemia induction. Table 1 also shows a

comparison of concentrations of plutonium-239+240 predicted for the fetus at full term, with measurements in fetuses following termination. All measurements were below the limit of detection of the technique, giving considerable support to the conclusion that uptake of plutonium is unlikely to have been significantly underestimated in the modelling procedure.

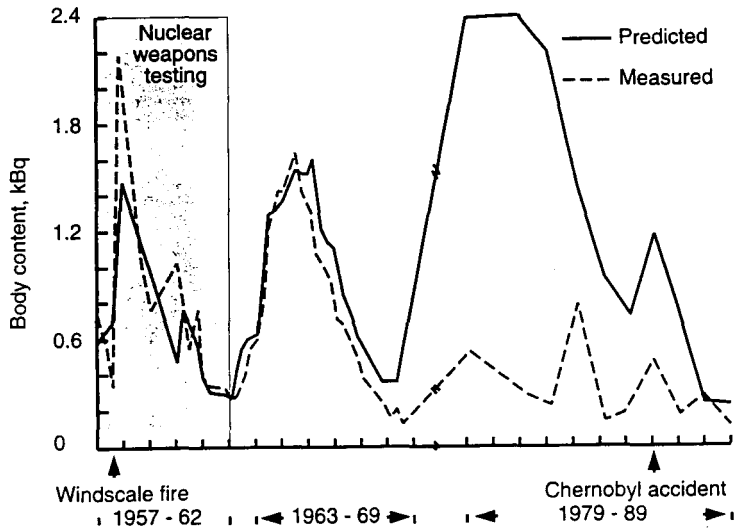

Figure 1: Measured and predicted radiocaesium body contents for 1957-89

	Skeleton		Liver		Lung		Fetus	
Radionuclide concentration	Measured	Predicted	Measured	Predicted	Measured	Predicted	Measured*	Predicted
$^{238}Pu + {}^{240}Pu$ (mBq kg^{-1})	7	153	68	342	12	2	AND <1.5	0.1
^{137}Cs (Bq kg^{-1})	-	-	4	11	-	-	-	-

* AND = Activity not detected, followed by the 95% detection activity

Table 1: Comparison of mean measured with predicted radionuclide concentrations in autopsy and fetal tissues

WEAPONS FALLOUT COMPARISON

Worldwide autopsy measurements of plutonium-239+240 have been collated and median activity concentrations determined on a yearly basis for each location where data are available. In the cases examined, intakes of these radionuclides has arisen from atmospheric weapons testing and not from occupational exposure. Using an adapted version of NRPB's LUDEP (Lung Dose Evaluation Program)[3], which implements the new Respiratory Tract Model[2] and uses the new ICRP biokinetic model for plutonium, coupled with estimations of plutonium intake, activity concentrations for lung, liver, skeleton, lymph node and kidney were predicted. Annual intakes of inhaled fallout plutonium-239+240 were estimated using air concentration measurements of these radionuclides where possible. In the absence of these data, estimations were based on strontium or caesium air concentration measurements by utilising the radionuclide ratios, measurements made at similar latitudes or modelled data. Inhalation of plutonium was the only route of intake considered, this being the predominant exposure pathway.

Two sets of predicted values were calculated based on i) Type S and ii) Type M parameters. The comparison undertaken for one organ, lung, is illustrated in Figure 2. Taken with preliminary comparisons for the other organs, the results indicate that absorption of plutonium from lung to blood lies between Type S and M parameters. This study does appear to suggest that there is reasonable agreement between measured and predicted concentrations, given the inherent uncertainties in the process.

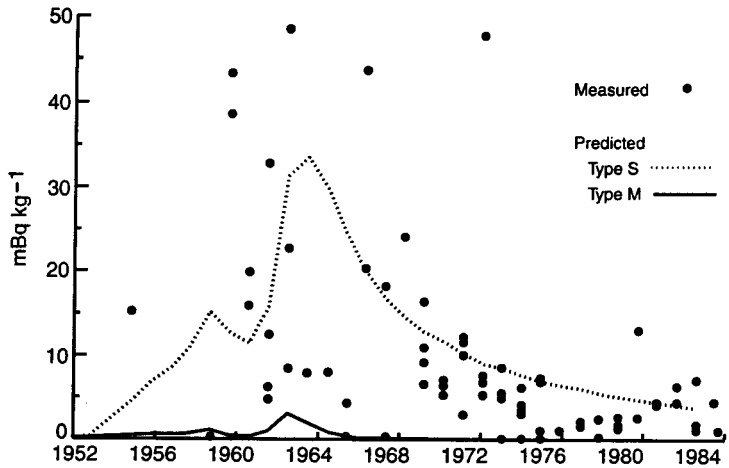

Figure 2: Measured and predicted plutonium lung concentrations

CONCLUSIONS

Human measurement data have been shown to be a valuable source of information to substantiate the data and methods used in model predictions. The Seascale comparison has shown that modelling procedures used to determine the radiation doses tend not to underestimate radionuclide intakes and deposition, and often overestimate them. Preliminary results of the weapons fallout study appear to indicate reasonable agreement between measured and predicted concentrations.

The benefits of using human measurement data for model validation are immediately obvious: it is only measurements of individuals which provide direct evidence of the level of radionuclide uptake by a population. There are disadvantages associated with this approach however, since in order to provide a reliable estimation of the radionuclide uptake by a population it is necessary to have a large number of measurements to ensure that the distribution of radionuclide uptake by the population has been adequately represented. Human data are by their very nature scarce and therefore for the purposes of these comparative studies, the measured radionuclide contents of studied individuals have been assumed to be representative of the population under investigation.

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GUIDE-LINES FOR AN EARLY EVALUATION OF A NUCLEAR ACCIDENT, CALCULATED WITH THE COMPUTER MODEL PARK

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Introduction

For a nuclear accident where large areas are contaminated, it is necessary to predict the exposure of the population as early as possible in order to plan appropriate countermeasures. The radioecological computer model PARK (Program System for the Assessment and Mitigation of Radiological Consequences) [1,2] is part of the German decision support system IMIS (Integrated Measurement- and Information System for the Surveillance of Environmental Radioactivity) [3] for a fast assessment of contaminations and doses. In this paper PARK is used to investigate the dose relevance of the exposure pathways, of ingested radionuclides, and of foodstuffs in relation to the date of the event.

Model Description

The following example is based on an assumed nuclear accident with core melting according to the release category no. 2 (FK-2) of the 'Deutsche Risikostudie Kernkraftwerke' [4]. Major portions of the activity inventory of a 1300 MW_{el} nuclear power plant are released into the environment by a leak in the containment of 30 cm in diameter. Within the scope of this scenario, the entire inventory of noble gases is released one hour after the end of nuclear fission. The respective percent portions are: 40 % iodine, 29 % caesium, 19 % tellurium, 3 % strontium and 0,26% lanthanides and actinides, res..

For approximating the atmospheric transport a dispersion model developed by NRPB [5] is used. The radioactive cloud is reaching the investigated distance 11 hours later, at a distance of approximately 200 km from the reactor, since the IMIS system is designed for long range contamination, and not short range distances. Physical decay and production of daughter nuclides are being considered. The ratio between the fraction of elemental iodine, aerosol bound and organic bound iodine is assumed to be equal.

These assumptions lead to an integrated activity concentration of 10^6 Bq·h/m³ for I-131, $7{\cdot}10^4$ Bq·h/m³ for Cs-137, and $1{,}5{\cdot}10^7$ Bq·h/m³ for total activity at the investigated distance. These values are used as input for PARK which then estimates the effective dose by all exposure pathways without countermeasures and change in living habits. PARK is based on the radioecological model ECOSYS [6], and therefore takes into account the state of plant growth and the effects of food processing. As a result, a total effective dose of about 700 mSv is calculated assuming a dry deposition taking place on July 1st. The figures however are scaled, so that this dose corresponds to a value of 1. In the case of wet deposition, it is assumed that the activity in air is entirely deposited on the soil and vegetation.

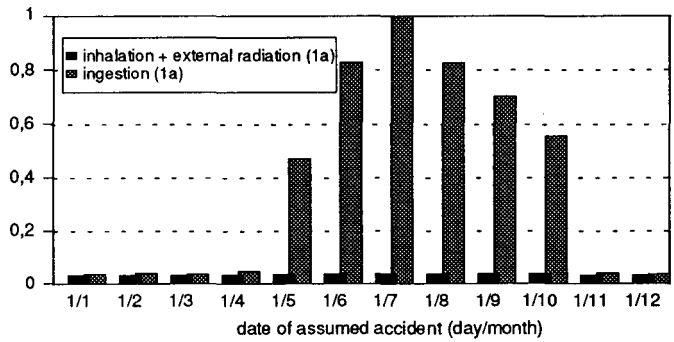

Fig. 1: Effective dose due to inhalation and external radiation (1 year) and from ingestion of contaminated foodstuffs over the first year following the assumed accident at a distance of 200 km, for adults and standard German living habits.

Relevance of Exposure Pathways

Different radiation pathways are considered: external exposure from the cloud and from the surface as well as internal exposure from inhalation and ingestion. Fig. 1 shows the results from the assumed accident occurring

in different months of the year. Ingestion is identified as the most relevant pathway in case of a nuclear accident during the period between May and October, whereas during the remaining months ingestion doses are assessed to be nearly of the same magnitude as those of the external exposure and inhalation.

Relevance of Ingested Radionuclides

Subdividing the ingestion dose into the proportions of radionuclides, only Cs-134, Cs-137 and I-131 are essentially contributing, altogether by more than 90 percent. This result is unexpected, since a set of 30 radionuclides has been included into the calculations [7] but may be understood by regarding the radiological transfer processes and dose coefficients.

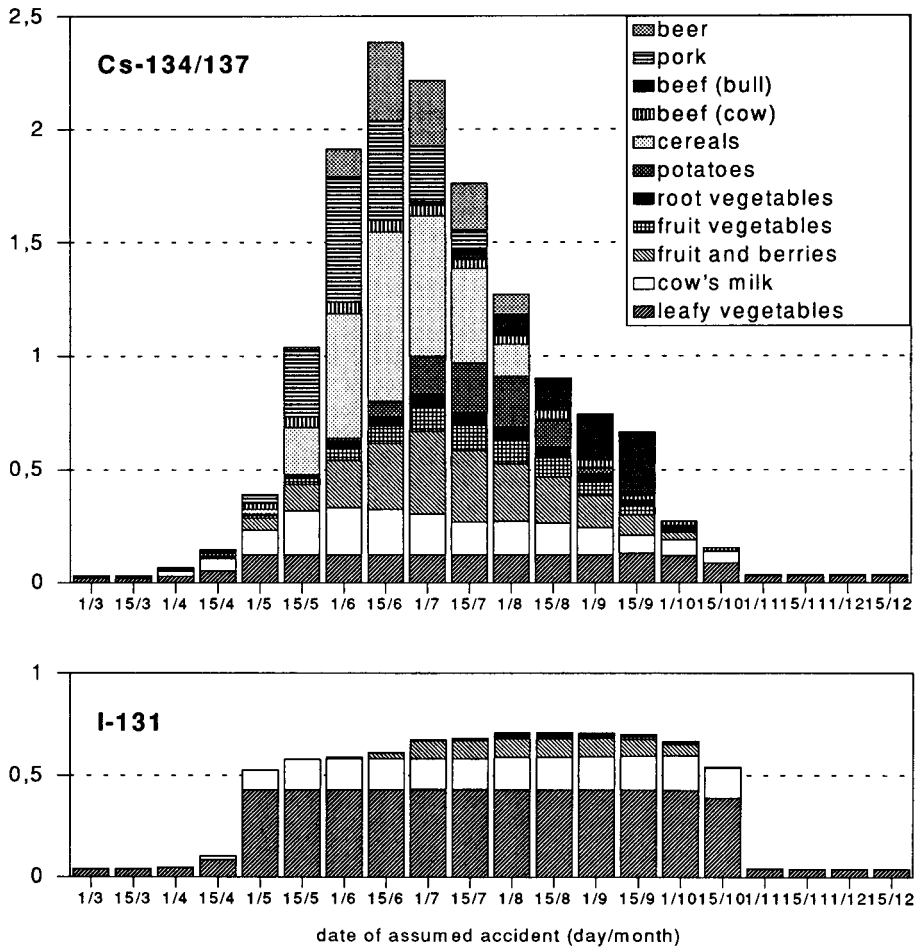

Fig. 2: Effective dose following an ingestion of contaminated foodstuffs during the first year after an assumed nuclear accident at a distance of 200 km. Average consumption rates for German adults are presumed.
***Note:** This diagram does not show a time series but shows each bar as representative of the life time dose related to the given date of the accident*

Relevance of Ingested Foodstuffs

Fig. 2 shows the expected contribution from main foodstuffs to the effective dose from the ingestion of radiocaesium and radioiodine at given dates of wet deposition during the first year. For the purpose of comparison the same scale is used for Fig. 1 and Fig. 2. It is obvious that the dose from ingestion varies by more than one order of magnitude with time of the event [8]. The highest doses can be expected in summer when the vegetation is developed. During this period, the contamination of the plant surface and translocation are predominant compared to the root uptake. Following an accident in spring, the contamination subsequently is reduced by the increased mass of the plant. The dose from the ingestion of leafy vegetables does not vary much between May and October because surface contamination is here the main factor. The dose contribution from other products, however, varies in a wide range and causes the maximum of the total dose during summer. This is most pronounced from products such as cereals, beer (brewn from summer barley), pork (fed with winter barley), beef (fed with corn) and fruits, which are easily contaminated with radiocaesium in summer from wet deposition.

Discussion and Conclusions

Between May and October, scenarios are possible where only limits for contamination of food products are exceeded while the other pathways contribute very little to the total dose. In this situation, countermeasures affecting the agriculture, e.g. restricting the consumption of fresh products, are very important. The results have been verified by using the radiation doses determined for several regions of Germany [9] after the Chernobyl accident which happend in the end of April. With values for the ingestion doses taken from Fig. 2, it is evident that the highest dose contribution results from iodine in combination with leafy vegetables and milk. The dose from caesium was also reduced by countermeasures that were taken. Severe consequences, especially from the caesium contamination, must be faced if such an event is taking place during summer. Our results show that it is not recommendable to rely on the experience from the Chernobyl event only, when assessing consequences from future radiological accidents.

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RADIOCAPACITY:
PROGNOSIS OF POLLUTION AFTER NUCLEAR ACCIDENTS

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INTRODUCTION

All ecological systems have an inherent fundamental peculiarity to accumulate and strongly keep back radionuclides got into them. Maximal ammount of radionuclides in ecosystem, which doesn't yet violates its general trophic functions (productivity, conditioniarity and reliability), can be named as a radiocapacity of a given ecosystem (1).

Let us consider some problems connected with radiocapacity of ecosystems (for example a water body system).

RADIOCAPACITY CALCULATIONS

Water bodies consist of three components: water, soil and biota. Getting into water, radionuclides rapidly and evenly are distribute in it. At the same time radionuclides transfer to the floor of the water body and various live organism (biota). As time goes on, radionuclides which get into a water body in amounts of A, are distributed by its components according to the formula

$$A = B \cdot S \cdot (H + h \cdot K_1 + c \cdot H \cdot K_{2)}), \qquad (1)$$

where B is the critical amount of radionuclides per unit volume of water; S is a surface square of water body; H - its average depth; h - thickness of the sorbing layer of body floor; c - concentration of biota in water body; k_1 and k_2- transfer coefficients of radionuclides into biota and body floor (usually $k_1 \approx k_2 \approx 10^2$-$10^3$). If $h >> c \cdot H$ (it is usually observed), then formula (1) is simplified to formula (2):

$$A = B \cdot S \cdot (H + h \cdot K). \qquad (2)$$

One can see, a term connected with biota is vanished. However, in any water body contaminated with radionuclides, biota plays a role of three functions: accumulation of radionuclides from water; transfering them to the floor of water body; and what's more important - maintenance of such a type of physical-chemical state of floor deposits, which prevent desorption of radionuclides from them.

A firm sorption of radionuclides in floor deposits is observed at pH≥7.0. Under the lower magnitude of pH a desorption process is started, which comes close to its maximum value at pH≈5.0. The decreasing of pH from 7.0 to 5.0 is generaly associated with biota dying off in water body.

Biota state determines a critical value of radionuclides in contaminated water body, such a concentration B, to which biota is still able for normal operating ($B \approx 10^3$-10^4 Bk/l).

RADIOCAPACITY FACTOR OF ECOSYSTEMS

By performing some necessary investigations it is easy to determine S, H, h, and k for any water body. To compare radiocapacity of various water bodies one can use the following unitless quantity:

$$F = \frac{h \cdot K}{h \cdot K + H}. \qquad (3)$$

It can be named as an 'radiocapacity factor'. It shows what part of radionuclides is firmly sorbed by floor deporsits. For the majority of water bodies (exept swamps) F≈0.7-0.9.

The radiocapacity factor F and a critical value B of radionuclides into the ecosystem can be estimated for different types of ecosystems (see table 1)

Table 1. Approximate values for F and B for different ecosystems

Ecosystem	F	B (Bk/km^2)
Desert	0.1	10^{11}
Meadow	0.3	10^{13}
Like, river	0.9	10^{12}
Woodless slope	0.2	10^{11}
Forest slope	0.6	10^{12}

Knowing F and B for different landscapes, one can estimate their possible role as desactivators in the case of radionuclide contamination. All that gives us a possibility to make a forecast when and what kind of countermeasures must be taken.

PROGNOSIS OF POLLUTION OF THE DNIEPER CASCADE WATER BODIES

After the Chernobyl accident radionuclides pollutions of large areas of Byeloruss, Russia and Ukraine take place.Practically, all these areas are arranged about the Dnieper cascade square. Bellow Chernobyl the Dnieper consists of six large water bodies which drain into the Dnieper-Bug lagoon. Water exchange is quite small in these water bodies (close to $3 \cdot 10^2$ volume per year), that allows one to apply the method described above to this system.

Evaluation of the basic parameters of the Dnieper water bodies is presented in table 2.

Table 2. Factors of radiocapacities of the Dnieper cascade water bodies.

Water body	S (km^2)	H (m)	h (m)	k	F
Kiev	920	4	0.1	100	0.7
Kanev	690	4	0.1	50	0.6
Kremenchug	2250	6	0.1	800	0.8
Zaporozhye	570	4	0.1	100	0.7
Dnieper	410	8	0.1	230	0.7
Kakhovka	2150	8	0.1	280	0.7
Factor of radiocapacity of the whole cascade					0.9994

Calculations in three last columns are done for ^{137}Cs. We notice, the factor F for the whole cascade is much higher (about 0.9994) than for each water body in separate. It implies that floor deposits sorb and firmly keep back 0.9994 part of initial ^{137}Cs radionuclides during

water transportation over the cascade from the Kiev water body to the Kakhovka one. Only 0.0006 part arrives to the Dnieper-Bug lagoon. Estimations have been done in 1988 (2).However, seven years later the differences between radionuclides contaminations of Dnieper cascade water bodies are approximately the same, as they were forecasted soon after Chernobyl accident.

CONCLUSION

The notions of `radiocapacity of ecosystems' and the `factor of radiocapacity' are determined. It is shown, that analysis of ecosystems in radiocapacity terms allows one to forecast for a long time the distribution of radionuclides over the polluted areas and determine countermeasures in all regions susceptible to risk. For example, the Dnieper cascade water bodies is observed.

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ESTIMATING AIR QUALITY CLOSE TO RADIUM-BEARING WASTE PILES

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INTRODUCTION

In the last decades, with the advent of regulatory air pollution programs came the need for practical tools to estimate transport and diffusion of pollutants from continously emitting sources. The Gaussian plume model has become widely accepted for this need. Computer codes based on it and applicable to single sources were further extended to multiple sources and area-sources (1-4). Such models are still in wide use despite their age and limitations. In the last few years, recommendations have been made to improve models of this type (5,6). The present work has the objective to propose an adaptation of the Gaussian model to produce reasonable estimates of radon air concentration above or in the vicinity of well-characterized ground level area-sources.

The FLURAD-AREA model developed in the present work is basically similar to many other sector-averaged models. It consists of replacing each area-source by a set of equivalent point-sources and summing up the contributions of these points. The formula used to determine the average atmospheric concentrations due to an elementary point source is derived from the Gaussian plume formula, assuming that each direction in a wind sector centered on the source is equally probable over a long period of time. But the new model differs from the previous ones in the treatment of the plume advective speed. In former models, the wind speed that appears in the dispersion formula is held constant throughout the computation and assumed to be equal to some reference wind speed (measured at a reference height of 10 m for instance). This seems quite unrealistic in case of a ground level source, since the average advective speed of the pollutant is obviously very low at small distances from the source and much greater at larger distances, due to the vertical extension of the plume. In the new model, this is taken into account by considering that, for each point-source, the average advective speed of the emitted pollutant increases with distance from the source. The new model is believed to give more realistic results that the former ones, particularly at short distances from the sources where the wind speed effect is of great importance.

BASIC FEATURES OF THE MODEL

The model utilizes a user-specified grid of terrain elements which describes the local environment around the release zone. Each rectangular element in the grid can be characterized by terrain roughness, height of vegetation and radon fluence rate. It can be subdivided into several tens of small area sources which are assimilable to point sources. The contributions of all the sources are summed up to determine the radon-in-air concentration at the grid nodes.

The basic equation in the model is the well known sector-averaged equation for a ground-level continuously emitting point source. Wind direction is assumed to be specified in sectors of angle ω. For any period of averaging, it is assumed that all wind directions within a given sector occur with equal probability. The average concentration at the sampling height h, for a particular combination of meteorological conditions at any receptor point at distance ρ within the given sector due to a point source with emission rate Q is:

$$\chi_{ij}(\rho) = \frac{0.798}{\alpha_j(\rho)} \frac{Q}{\bar{u}_i \sigma_j(\rho) \omega \rho} \exp\left(-\frac{h^2}{2\sigma_j^2(\rho)}\right) \qquad (1)$$

where $\bar{u}_i$ is the wind speed at z=10 m representative of the wind speed class i; $\alpha_j(\rho)$ a ventilation factor, to be defined later, that does not appear in the original equation; $\sigma_j(\rho)$ is the standard deviation of the vertical concentration distribution for a ground-level emission. The vertical standard deviation is a function of the distance ρ from the source and the atmospheric stability class j.
Equation (1) is evaluated to determine the effect of each equivalent point source on a receptor (at a grid node) for each possible combination of atmospheric stability and wind speed. The relative

frequency of occurrence of each such meteorological combination is then used to weight the calculated concentrations; the resulting weighted concentrations are summed over all meteorological combinations and all sources. Thus the long-term average radon concentration at the receptor is obtained by:

$$\bar{\chi} = \sum_i \sum_j \sum_n F_{ij}(d_n)\, \chi_{ij}(\rho_n, d_n) \qquad (2)$$

where d_n indicates the wind direction sector in which transport from a particular source (n) to the receptor occurs; ρ_n is the distance from the particular souce to the receptor; $F_{ij}(d_n)$ denotes the relative frequency of winds blowing into the given wind direction sector d_n, for a given wind speed class i, and atmospheric stability class j.

The relative frequencies $F_{ij}(d_n)$ can be derived from hourly meteorological data, the stability classes being defined for instance on the basis of the Turner's objective criteria. If, however, specific meteorological data are lacking, one may use simplified regional data, for instance the frequencies of occurrence of the stability classes and their associated mean wind speeds at 10 m, all wind direction being confounded. The approximation then is that the frequency of weather categories is independent of wind direction. This may be a reasonable assumption for many sites.

TREATMENT OF THE ADVECTIVE SPEED

In the original model, the advective speed of the substance being dispersed was held constant thoughout computation and arbitrarily assumed to be equal to the wind speed $\bar{u}_i$ measured at the reference height z=10 m. In the new present model, the advective speed is now $\alpha_j(\rho)\bar{u}_i$, where $\alpha_j(\rho)$is a ventilation factor which is the ratio of the true average wind speed in each vertical cross-section of the plume, to the wind speed at 10 m formerly considered. Due to the large gradient of wind speed in the atmospheric layer adjacent to the ground, the ventilation factor is much less than unity at small distances from the source where the plume is still close to the ground, and increases at greater distances with the vertical extension of the plume.

It was shown by Smith and Singer (7) and by Horst (8) that the average horizontal speed of the pollutant molecules is approximately equal to the wind speed at the height $0.6\,\sigma_j(\rho)$ above ground-level. If $u_{ij}(z)$ is the mean vertical profile of wind speed for the wind speed class i and the stability class j, then the ventilation factor in the formula (1) is the following function of the distance ρ:

$$\alpha_j(\rho) = \frac{u_{ij}\left[0.6\,\sigma_j(\rho)\right]}{u_{ij}(10)} \qquad (3)$$

In this equation, the wind speed profiles above and beneath the top of vegetation can be respectively approximated by the classical log-linear formulas taken from Businger (9) and the exponential function from Meroney (11):

$$u_{ij}(z) = \kappa(u_*)_i \left[\ln\frac{z-d}{z_0} - \Psi\left(\frac{z-d}{L_j}\right)\right] \qquad (z \geq H) \quad (4)$$

$$u_{ij}(z) = u_{ij}(H)\exp\left[-\beta\left(1-\frac{z}{H}\right)\right] \qquad (z < H) \quad (4bis)$$

where κ is the Von Karman's constant; $(u_*)_i$ the friction velocity; H the height of vegetation; d the zero plane displacement height; z_0 the roughness length; L_j the Monin-Obukhov length that characterizes the atmospheric stability class j according to the correspondance shown by Golder (10); Ψ a corrective term whose expression is given in reference (9); β a factor between 1 and 3 depending on the density of the canopy foliage .

INFLUENCE OF SOIL COVER

The pattern of radon-in-air concentration above an area source of limited horizontal extension is very sensitive to the characteristics of the atmosphere boundary-layer close to the ground. Each sampling point situated above the source is at close distance from any point of the ground-level area source. Consequently, the advective speed $\alpha_j(\rho)\bar{u}_i$ to be used in equation (1) is much smaller than the wind speed $\bar{u}_i$ measured at 10 m. The concentrations above the source that are derived from the new model are thus much higher than those derived from the former model in which the advective speed was assumed to be $\bar{u}_i$ throughout calculation. The difference between the new concentrations and the old ones are strongly dependent of the nature and characteristics of the vegetation cover which determine the wind speed profiles in the first meters above ground level.

The concentrations of radon above a source of given strength are the smallest for a well ventilated area such as a flat surface with only low vegetation. In these conditions, the dampening of wind speed in the lowest layers of the atmosphere is relatively limited and does not induce the build-up of high concentrations. On the contrary, if the source is situated in a ill-ventilated woody area, the wind speed is strongly attenuated close to the ground, which results in the development of higher concentrations above the source itself. The influence of the vegetation cover, such as described in the new model, corresponds to the intuitive feeling that the radon concentration pattern should depend on the site ventilation.

The main difficulty encountered in the implementation of the new model is to characterize the boundary-layer over a terrain of variable roughness. The boundary layer properties and particularly the vertical wind speed profiles given in the literature as representative of low level boundary-layer properties over the land are those observed over fairly extensive flat plains with no more cover than homogeneous forest or low vegetation. The relevant surface properties are frequently affected by the patchiness of the vegetation cover and the presence of individual obstacles ranging from bushes and hedges to woods and buildings, and also to some degree by the natural topography of the landscape. While this spatial inhomogeneity does not affect very much the medium-range dispersion of pollutants from high stacks, it becomes of prime importance when the source is at ground level and the receptor very close to the ground at a short distance from the source.

By applying the new model to the typical case of a square source of 100 m x 100m, which emits radon with an homogeneous fluence rate of 1 Bq m^{-2} s^{-1}, it appears that the yearly averaged radon-in-air concentration above the source itself can reach values of a few tens of Bq m^{-3}. The concentration rapidly decreases as one moves away from the border of the source and becomes quite negligible beyond a few hundred meters from the source. This confirms the usefulness of trying to predict the radon air concentrations in the near-field of the emitting sources, which is the only place where the excess of concentration over background may be significant. Due to the difficulties and limitations aforementioned, the new model is only a tentative one. Further work is needed to improve the model and validate it by comparing predicted and measured values above and close to well-defined sources.

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Sorption/Desorption phenomena in uranium mill tailings : Study of the distribution of radionuclides on pure solid phases and application to the modelling of source term.

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Abstract

The mining and milling of the uranium raw material are processes which leave substantial volumes of uranium mill tailings. These radioactive waste products may have an environmental impact in far future. The health hazards occur in a number of pathways via radioactive gas emanation, external exposure, radioactive dust and through contamination of surface and ground waters. For long term consequences, modelling has to be performed to predict future exposures. Of course, the necessity of using predictive models requires a basic understanding of the relevant physical, chemical and geochemical properties of the system components.

In a previous paper [1], we have reported data obtained on uranium tailings deposited in Lengenfeld (Germany) 20 years ago. The solid phases (determined by XRD) are mainly quartz, dolomite and kaolinite. The lack of data concerning the mechanisms responsible for the retention of radionuclides on these solids has led us to perform batch experiments on pure solid phases. In this paper, we present the influence of the solution composition (pH, ionic strength) on the sorption of radium onto a solid phase (kaolinite). The variations obtained have allowed to define the mechanisms controlling the sorption and to determine the associated constants. The introduction of these values in geochemical codes based on thermodynamic equilibrium will allow to predict the mobility of radium in presence of these phases.

Key words

Uranium mill tailings, radium, distribution coefficient, kaolinite, ion exchange

Introduction

Uranium mining and milling have left behind great amounts of tailings containing both radionuclides (radium and uranium) and toxic elements (As, Ni,..) which may be released in the geosphere. The control of the release rate of potential contaminants from the wastes is the most important objective of long term management of these residues. Therefore, the knowledge of the original ores, the milling and extraction processes and the chemical and mineralogical characteristics of the tailings are required in order to predict the migration of the contaminants throughout the tailings.

Among numerous mineralogical phases, clayous phases (smectite, kaolinite) are quite important because of their high sorption capacity. Few data concerning the radium sorption onto solids phases are available in literature [2] and in most of cases, the radium behaviour is assumed to be quite similar to those of divalent cations (strontium, baryum...). This paper is devoted to the study of the sorption of radium onto kaolinite as a function of sodium concentration and pH in order to elucidate the mechanisms responsible of the radium sorption.

Experimental part

The studied solid was kaolinite provided by Sigma. This solid was purified in order to saturate all the sorption sites by Na^+ cations. The study of Na^+ sorption on this kaolinite by L. GORGEON [3] has led to determine a Cation Exchange Capacity (CEC) of 4.2×10^{-2} meq.g^{-1} whereas the cobaltihexamine method has indicated a CEC value of 2.8×10^{-2} meq.g^{-1}. This difference was due to the presence of two reactionnel sites as we could see in our results. The specific area of the purified kaolinite was estimated to 13.2 m^2.g^{-1} $\pm$ 0.1 m^2.g^{-1} and the exhangeable cation was sodium. The solution of radium was obtained by dilution of tracers in sodium chlorure (226×10^3 Bq of ^{226}Ra solution provided by Amersham). All other chemicals were reagent grade.

All the sorption experiments were performed in batch at 22°C. No attempt to exclude air was made. The ratio volume of solution to mass of clay was fixed to 20 ml.g^{-1}. The solution containing radium was added to the solid and then the mixing was shaken (3-D shaking). After one day (result of kinetic study), the slurry was centrifugated (4900 rpm, 30 mn). The pH was then measured in the solution with a Sentron pH-meter and

the radium concentration in solution was determined by gamma spectrometry with a high resolution hyperpure germanium spectrometer.

Radium distribution coefficients D were calculated as the amount of radium sorbed on the solid divided by the amount of radium remaining in solution :

$$D_{(ml.g^{-1})} = ([Ra]_{solid}/[Ra]_{solution}) \times V/m \qquad (1)$$

Results and discussion

Time dependence

For different sodium concentrations, the sorption has been studied as a function of time (Figure1).

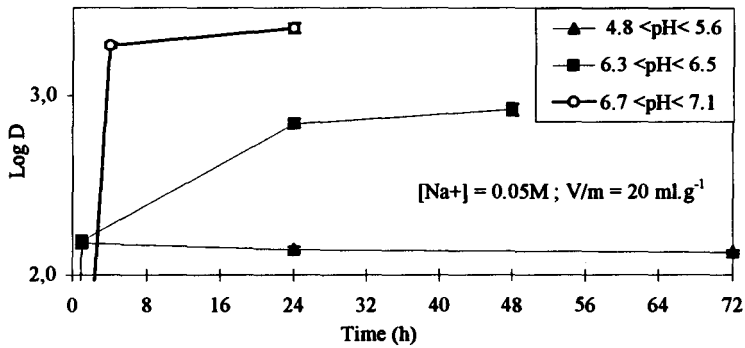

Figure 1 : Time influence on the radium sorption on kaolinite

A two steps kinetics is observed. The first step (rapid) is attributed to the sorption onto the surface, while the second often continuing for days is attributed to a diffusion process of the species through the first hydrated layers of the solid [4]. To avoid a possible alteration of clay for long settling time and considering the low variations of the amounts sorbed after 24 hours, all the experiments have been conducted with an equilibrium time of 24 hours.

pH dependence

The variations of the amounts of sorbed radium as a function of pH for different sodium concentrations are presented on Figure 2.

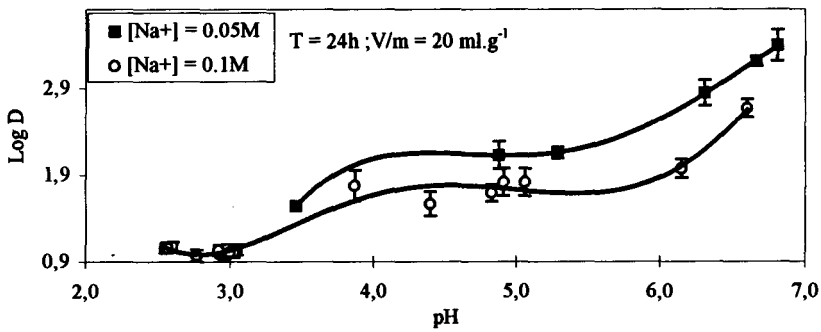

Figure 2 : pH influence on the radium sorption on kaolinite

For the two sodium concentrations, the distribution coefficient increases with pH excepted a plateau between pH 4 and 5.5 for which logD is constant. A possible explanation of this shape is a competition between H^+, Na^+ and Ra^{2+} at low pH. For pH>4, only competition between Na^+ and Ra^{2+} occurs and the distribution coefficient remains constant with increasing pH. For pH>6, D increases sharply. A similar curve has already been mentioned elsewhere [3] and has been attributed to another kind of sorption sites on kaolinite.

[Na^+] concentration dependence

The ionic exchange mechanism between radium and sodium can be written:

$$2\{X^-Na^+\} + Ra^{2+} \Leftrightarrow \{(X^-)_2Ra^{2+}\} + 2Na^+ \qquad (2)$$

$\{X^-Na^+\}$, $\{(X^-)_2Ra^{2+}\}$ = sorbed species

$$\text{and } K_{Na/Ra} = [(X^-)_2Ra^{2+}]\,[Na^+]^2 / [X^-Na^+]^2\,[Ra^{2+}] \qquad (3)$$

$K_{Na/Ra}$ = selectivity coefficient between sodium and radium for sorption on kaolinite

Combining (1), (2) and (3), we obtain the relation :

$$\log D_{Ra} = \log K_{Na/Ra} + 2\log CEC - 2\log[Na^+] = \text{Constant} - 2\log[Na^+] \qquad (4)$$

This model predicts a linear variation of D with the electrolyte concentration. The sorption of radium as a function of sodium concentration has been studied in a pH range 8 to 9. At high pH value, the H^+ concentration in solution is widely lower than Na^+ concentration and the sorption sites are occupied by sodium. The data expressed as $\log D = f(\log[Na^+])$ are plotted on Figure 3.

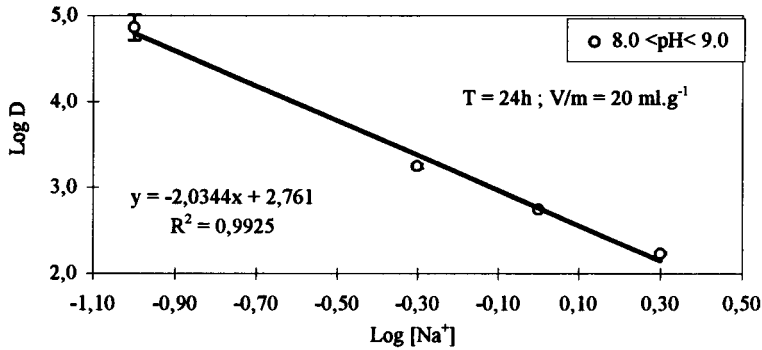

Figure 3 : Influence of Na^+ concentration on the radium sorption on kaolinite

The straight line obtained has a slope of -2.0344 (R^2 = 0.9925), which confirm the cation exchange mechanism between radium and sodium. The selectivity coefficient $K_{Na/Ra}$ can be estimated by the intercept of the curve and the CEC value measured by L. GEORGON [3]. A very high value (3.2×10^5 $g.ml^{-1}$) is obtained corresponding to a stronger affinity of radium than sodium for the sorption sites on kaolinite.

This study has allowed an estimation of the Na/Ra selectivity coefficient. Moreover, the strong influence of pH on the cation sorption onto kaolinite has been clearly observed, due to the competition between Na^+, Ra^{2+} and H^+ for sorption sites.

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METHODOLOGY OF ENVIRONMENTAL IMPACT STUDY AND OPTIMIZATION OF RELEASES
Application to the Study Centre of CADARACHE

FRENCH ATOMIC ENERGY COMMISSION
by
JC CARIES (Health Physics Department)

OBJECTIVES :

- Impact study.
- Optimization of releases.

PRINCIPLES :

- Activities in the air and the water computed independently.
- The dose error contains more information than the dose itself ; it is representative of the knowledge evolution of the site.
- Computations of sensitivity and global error are essential in the modelling - experimentation interaction.
- Assessment of doses in the organs.
- Information feedback of the data-processing tools of a CEA subsidiary company (CISI) : modularity and flexible operation.

PROCEDURE :

- Source term : Spectrum of the radioelements.
- Impact study (per station).
  - * computations of transfers to both food and man ; sensitivities and error computations,
  - * main parameters with regard to the value : dose + error (d+Δd)
- Surveillance plan of the site.
  - * computations of the number of stations to be studied for a fixed percentage of (d + Δd),
  - * reduced equation of transfers , mapping and meshing of the site,
  - * exposures and food activities computations in every mesh, and filing.
- Optimization of releases (critical station)
  - * determination of the boundary level of exposures,
  - * successive iterations with the source term of the impact study and tests with different spectra of the source term.

RESULTS : Liquid transfers : Δd = (7 to 10)×d - Atmospheric transfers : Δd = (3 to 7)×d

	AIR				WATER					
	^{85}Kr	^{3}H	^{131}I	^{125}Sb	^{3}H	^{60}Co	^{137}Cs	^{90}Sr +Y	^{241}Am	^{239}Pu
Authorized releases (1978) Bq/year (1)	555.E12		18,5.E9		1,85. E12	3,7.E9			0,37 .E9	
If d+Δd=10 μSv/year multiplying factor of (1)	(if 100% ^{3}H) 0,3		22		65	24			0,3	

Maximum exposure due to present releases from the Centre :d < 1μSv/year - *Present tritium releases from the Centre: some per cent of the authorized releases*

OPTIMIZATION: *α emitters* Objective: release << 0,3 × authorization by:

- improving analyses to avoid adding up detection limits
- retaining the solid particles of the effluents in the installations.

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IRPA9

1996 International Congress on Radiation Protection

Proceedings / Volume 3

Vienna, Austria, April 14-19, 1996

Organised by the Austrian Association for Radiation Protection

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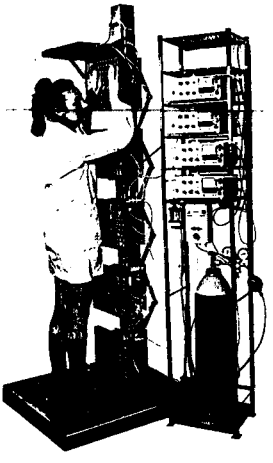
Body Contamination Monitors - State of the Art and Future Development

Rainer Gerlach, Rados Technology GmbH

History

Since the introduction of large-area proportional counters in the mid-1960s, it is possible to efficiently test large parts of the body for contamination. Initially, simple hand/foot monitors were used with these detectors, but already by the end of the 1960s there were personnel contamination monitors for also measuring large parts of the body. These devices were constructed according to the materials available and with simple analogue electronics. Some of these devices are still in use today, more than 25 years later. Now personnel contamination monitors are a constituent part of monitoring controlled areas. Since their introduction they have been subject to massive on-going development in almost all sectors. Detectors can be manufactured with almost any dimensions to match requirements and with appropriate electronics are also suitable for simultaneous alpha measurement. The geometrical arrangement of the detectors has been improved so that almost all areas of the body are measured. The application of the extremely powerful computer techniques that are available nowadays has brought about substantial improvements in many respects: background subtraction, comprehensive service and calibration support, user guidance through the combination of modern sensor technology with monitor displays and speech output, networking and data storage.



Body Contamination Monitor built in 1970

Classification and Application

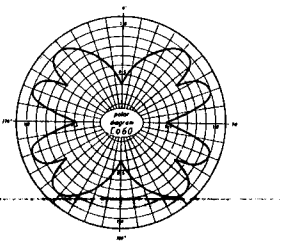
Personnel contamination monitors can be classified into three categories: **Hand/foot clothing monitors:** These just monitor the hands and feet simultaneously. The clothing must be measured with moveable clothing probes. The application extends to areas with a low throughput of personnel. **Pre-Monitors:** These also measure parts of the clothing simultaneously with the hands and feet, but normally only have a simple measurement geometry. The field of application covers checks within the controlled area with the objective of suppressing the spread of contamination. **Exit Monitors:** These measure the complete body with an optimum measurement geometry and are installed at the exits of controlled areas with high throughputs of personnel. The highest demands are placed on these monitors

Basic principles

In almost all countries there are laws and directives which stipulate the requirements for the monitoring of controlled areas. Here, the limits for the contamination monitoring are defined. This is mainly based on just the type of radiation or, as in Switzerland, is differentiated according to nuclides. The legal regulations are supplemented by standards such as the DIN standards in Germany or the IEC at an international level. Added to this, come recommendations from radiation protection commissions. Internationally the most important for personnel contamination monitors is IEC 1098 which was issued in 1992 ("Installed Personnel Surface Contamination Monitoring Assemblies for Alpha and Beta Emitters"). It regulates the minimum requirements on this kind of product and prescribes a range of tests with point and large-area sources.

- The determination of the uniformity of sensitivity is found with alpha and beta sources which are used for measuring the sensitivity distribution at specified measurement points within the measurement area of 15 x 10 cm^2 for hand detectors and 30 x 10 cm^2 for foot detectors. The sensitivity profile for the complete detector array is recorded for the body measurement (only beta). The horizontal profile based on a polar graph and the vertical profile based on an axial scan are recorded. An elliptical phantom with a 95 cm circumference and a 35 cm major axis is used as a basis.

- The sensitivity for the determination of the detection limits for the hand and foot monitors is made with large-area sources of 15 x 10 cm^2 (hands) and 30 x 10 cm^2 (feet). The mean of the axial and horizontal scans is used as the basis for the detection limits of the body measurement. Computational rules are also given for the determination of the detection limits in the quoted standard. However, a draft ISO standard (ISO 112929): "Determination of the lower limits of detection and decision for ionising radiation measurements")[1] already exists. In modern monitors with automatic adaptation of the measurement period the appropriate formulae in this draft standard are already taken into account.



Polar Response Diagram according to IEC 1098 (Split -Delta Geometry)

Development and Problems in the Measurement Geometry

The fact that IEC1098 makes the measurement geometry a decisive quality feature due to the requirement for recording the polar and axial scan diagrams indicates how important geometrical properties have become. In actual fact the capability of a monitor to detect contamination reproducibly over as much of the body as possible mainly depends on the size and arrangement of the detectors. After all, the detector response is very dependent on the distance to the source. This particularly applies to low-energy radiation and K capture radiation.

Whereas formerly monitors monitoring the front and back of a person simultaneously in one measurement phase were common, monitors measuring the front and back sequentially have become more established due to a major problem with the one-step method. The advantages are obvious - in contrast to one-step monitors, the distances are noticeably shorter and more reproducible. In addition, the sides of the body can be measured with good sensitivity. With the one-step method this is not possible even with the important nuclides of ^{60}Co and ^{137}Cs due to the large distances. Substantially better results are also produced for the head. Forearm detectors are significantly easier to integrate in a two-step monitor. An optimum sensitivity distribution is obtained with a triangular geometry with mean vertical division (Split-Delta from Rados). The sensitivity profile of this geometry is also tolerant against different sizes of person.

Also, modern geometries represent a compromise in detection characteristics, operating convenience and, not unimportant, costs. Improvements can still be achieved in the area of the shins, on the upper side of the feet and in the region of the upper arms, but with substantial extra expenditure. The measurement geometries now achieved have partial suitability for the detection of alpha contamination. Admittedly, a non-defined sensitivity profile is produced due to the extreme dependence of alpha radiation on distance, but, for example due to the split-delta geometry, large parts of the body and clothing come close enough to the detectors to enable good alpha detection.

Detectors

The market is currently dominated by large-area proportional counters for continuous operation. There are a number of reasons for this: They have high alpha and beta sensitivities with a low background, they can be produced economically in almost any dimensions and they can be easily repaired on site. Their main disadvantage is that they need a continuous supply of counting gas. Alternative sealed proportional counters only achieved limited success, because they exhibit practically no alpha sensitivity, have a noticeably lower sensitivity to low-energy radiation and, due to restricted dimensions, the devices constructed with enclosed detectors produce more dead zones. Added to this, much effort is required to replace the window foils so that they have to be more protected, giving a further reduction in sensitivity.

In principle, large-area plastic scintillation detectors are suitable, but there are still many fundamental problems to be solved. In particular, the detector material must be designed very thin (1 to 2 mm) so that it exhibits a low background with high beta sensitivity. Reading the light out from these types of plastics economical is still a problem, but if this can be achieved for a reasonable expenditure, then this type of detector would be a very interesting alternative to the proportional detector.

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[1] DIN 25482 is applied in Germany

Interference Effects and their Treatment

The measurement is disturbed by a number of external influences - varying background count rate, xenon and radon clouds, xenon and radon attached to the clothes or hair or contamination on the detectors. In contrast, in modern devices electromagnetic interference is no longer a problem.

~~The most important disturbing influence is the varying background count rate. It is caused by natural~~ radiation (cosmic rays, building materials, etc.) and by artificial radiation sources (active piping, contaminated surfaces, etc.). Added to the normally slow changes in the background are those, for example, due to the large queues in front of the monitor. Often the field is also directional. There are a number of methods of background subtraction. Many of the methods are too slow or lead to extended downtime periods. A method specially developed for application in personnel contamination monitors operates with an exponential filter and two sliding median values. The two median values are formed from different numbers of pulses so that they have different response times. During the background measurement the signal from the exponential filter is checked for whether it is located within specified limits about the median values. Based on a range of criteria, the system decides which of the median values is regarded as the present background. With this method the monitor can adapt quickly to short-term changes in the background without downtimes and without sacrificing the required accuracy.

Xenon and radon clouds mainly behave as other background variations and are taken into account by the subtraction method. This is not the case with radon or xenon activity clinging to clothing. It only appears when the person enters the monitor. The large-area characteristic of this activity may give a clue, but is not sufficient to be used as a criterion in the classification as background, because contamination may also cover a large area. It is just these types of contamination that must not under any circumstances remain undetected.
The alpha channels also exhibit a background effect. This is situated in the region of 0.1 to 0.5 pulses/s, so that usual background subtraction methods cannot be used. Therefore, for the alpha background subtraction, methods are used which are specially developed for low pulse rates.

Data Processing and Networking

Whereas with the first generations of personnel contamination monitors the measurement data was lost after termination of the measurement, it is now possible to link the monitors distributed in the plant with a central computer where the measurement data can be collected. Practice shows that important conclusions can be drawn from this data, enabling for example, the origin of the contamination in the system or the presence of hot particles to be deduced. From this, appropriate measures for improving the radiation protection can be derived. The possibility of accessing information from the devices centrally also represents a significant alleviation in the workload for the servicing personnel so that it also increases the availability.

Modern Body Contamination Monitor RTM860TS with Split-Delta Geometry and powerful PC-Electronics

Future Prospects

Although personnel contamination monitors have now achieved a high technical standard, continual development will still take place in the future. Modern computer technology offers opportunities of improving the service friendliness and also the possibilities of networking and data evaluation have not yet been exhausted. An example here is the coupling with dosimetry. The latest developments show that progress can be achieved in the field of background subtraction. It is still to be seen whether the large-area proportional counters in use today can be replaced by other technologies. In any case however, refinements in the measurement geometry and the signal evaluation will produce further improvements in the measurement characteristics.

"HOT PARTICLE" INTERCOMPARISON DOSIMETRY*

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ABSTRACT

Dosimetry measurements of four "hot particles" were made at different density thickness values using five different methods. The hot particles had maximum dimensions of 650 μm and maximum beta energies of 0.97, 046, 0.36, and 0.32 MeV. Absorbers were used to obtain the dose at different depths for each dosimeter. Measurements were made using exoelectron dosimeters, an extrapolation chamber, NE Extremity Tape Dosimeters™**, Eberline RO-2 and RO-2A™ survey meters, and two sets of GafChromic™ dye film with each set read out at a different institution. From these results the dose was calculated averaged over 1 cm^2 of tissue at 18, 70, 125, and 400 μm depth.

Comparisons of tissue-dose averaged over 1 cm^2 for 18, 70, and 125 μm depth based on interpolated measured values, were within 30% for the GafChromic™ dye film, extrapolation chamber, NE Extremity Tape™ dosimeters, and Eberline RO-2 and 2A™ survey meters except for the hot particle with 0.46 MeV maximum beta energy. The results for this source showed differences of up to 60%. The extrapolation chamber and NE Extremity Tape™ dosimeters under-responded for measurements at 400 μm by about a factor of 2 compared with the GafChromic™ dye films for two hot particles with maximum beta energy of 0.32 and 0.36 MeV which each emitted two 100% 1 MeV photons per disintegration. Tissue doses determined using exoelectron dosimeters were a factor of 2 to 5 less than those determined using other dosimeters, possibly due to failures of the equipment.

INTRODUCTION

Previous investigators used various methods to determine dose from hot particles including extrapolation chamber measurements (1-4), exoelectron dosimeter measurements (3,5), and radiochromic dye film measurements (2,4,6-9). The difficulty with such measurements arises from the extremely non-uniform dose distributions on contact with the particles (2,7). The results from radiochromic dye film dosimetry have shown differences of up to 40% for different imaging systems reading the same irradiated film (8).

METHODS AND MATERIALS

To evaluate the precision of appropriate systems for measuring dose from hot particles, an intercomparison study was carried out at Brookhaven National Laboratory (BNL) by researchers from four institutions. Five different methods from four institutions (Table 1) were used to measure doses from four hot particles with different beta- and gamma-emission characteristics (Table 2). Films exposed by the BNL group were read at the National Institute of Standards and Technology (NIST), and films exposed by the UBIRM group were evaluated at UBIRM. An empirically determined rule of thumb was used to convert the Eberline RO-2 and RO-2A™ measurements to dose averaged over 1 cm^2 at 70 μm depth (4).

The Tm-170, Sc-46, and Yb-175 sources were cut from foils into three-dimensional slab sources. The Co-60 particle was spherical (Table 3).

Sources were mounted on styrofoam blocks to minimize backscattered beta particles, and for containment, they were covered with a thin radiation-resistant cover of Kapton™ with a thickness of 13 μm and a density of 1.4 g/cc. Sources were placed on each dosimeter for a period that did not exceed the saturation level of the dosimeter.

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*This work was supported by the U.S. Nuclear Regulatory Commission.

**Certain commercial products are mentioned by name for informational purposes. This does not imply endorsement by the researchers nor that they are the best or only products available for the purposes described.

Exposure times ranged from 5 s to several days. Absorbers were inserted between the source and the dosimeter to obtain doses near tissue thicknesses of 18, 70, 125, and 400 μm. Doses were corrected for decay to a reference time and computational methods were used to obtain dose at the density thicknesses of interest averaged over 1 cm^2.

Table 1. Dosimeter(s) Used by Each Research Institution

Research Institution	Dosimeter	Descriptive Reference
Brookhaven National Laboratory (BNL)	GafChromic™ Dye Film	2,4,7
University of Birmingham (UBIRM)	GafChromic™ Dye Film	2,4,7
Pacific Northwest Laboratory (PNL)	Exoelectron Dosimeter	3,5
Yankee Atomic Electric Company (YAEC)	Extrapolation Chamber NE Extremity Tape™ Dosimeter Eberline RO-2 and RO-2A™ Survey Meters	4 a 4

a Formerly called Vinten™ dosimeter, consists of 10 mg/cm^2 TLD crystals adhered to a backing material.

Table 2. Isotopic Characteristics of the Hot Particles Used in the BNL Dosimetry Studies[a,b]

Isotope	Half-Life (days)	Maximum β Energy (MeV)	β Abundance (%)	γ Energy (MeV)	γ Abundance (%)
Tm-170	129	0.97 0.88	76 24	0.084	3.3
Yb-175	4.2	0.47 0.35 0.071	87 3.3 11	0.396 0.283 0.114	6.5 3.1 1.9
Sc-46	83.8	1.48 0.357	0.004 100	1.12 0.889	100 100
Co-60	1902	1.48 0.317	0.12 100	1.332 1.173	100 100

a Weast, R.C. (Editor), *CRC Handbook of Chemistry and Physics*, CRC Press, Inc., Cleveland, OH 1976.
b NCRP Report 58, "A Handbook of Radioactivity Measurements Procedures," National Council on Radiation Protection and Measurements, Bethesda, MD, 1978

Table 3. Dimensions and Densities of Particles

Particle	Density (g/cc)	Thickness (μm)	Length (μm)	Width (μm)
Tm-170	9.4	260	440	400
Sc-46	2.9	130	460±74[a]	330±42[a]
Yb-175	7.0	130	480	340
Co-60	8.4	210±11[a,b]		

a Average (±1 σ) from the batch of particles neutron activated at the same time.
b The Co-60 source was spherical.

RESULTS

The results showed agreement within 30% between the GafChromic™ dosimeters, extrapolation chamber, NE Extremity Tape™ dosimeters, and Eberline RO-2/2A™ ion chamber for 70 and 125-μm tissue depths (Table 4). The best agreement was for the Tm-170 particle, while the worst was for the Yb-175 particle; the reason for the latter is

not understood. The good agreement between the Eberline RO-2/RO-2A™ and the other dosimeters was surprising. Unfortunately, the detector saturated while measuring the Sc-46 particle. The significant under-response of the exoelectron dosimeters compared with other techniques also is not understood.

Table 4. Doses to 1 cm² of Tissue at Selected Depths Derived from Interpolation of Measured Values

Hot Particle	1 cm² Depth (μm)	BNL/NIST GafChromic™ (Gy/s)	UBIRM) GafChromic™ (Gy/s)	PNL ExoElectron (Gy/s)	YAEC Extrapolation (Gy/s)	YAEC NE Extremity Tape™ (Gy/s)	YAEC RO-2/2A™ Thumb Rule (Gy/s)
Tm-170	18	1.2E-03	1.3E-03	2.4E-04	1.3E-03	1.3E-03	
Tm-170	70	9.8E-04	1.1E-03	1.4E-04	1.1E-03	1.2E-03	1.1E-03
Tm-170	125	1.0E-03	9.6E-04	1.1E-04	8.9E-04	1.0E-03	
Tm-170	400	5.6E-04	5.9E-04	4.7E-05	4.9E-04	5.6E-04	
Yb-175	18	7.0E-03	1.2E-02	9.7E-03	1.2E-02	1.2E-02	
Yb-175	70	7.5E-03	5.3E-03	3.5E-03	8.2E-03	8.7E-03	6.7E-03
Yb-175	125	4.8E-03	6.5E-03	2.2E-03	5.6E-03	6.4E-03	
Yb-175	400	1.7E-03	1.9E-03	3.3E-04	1.4E-03	1.3E-03	
Sc-46	18	1.1E-01	1.0E-01	1.4E-01	1.1E-01	7.2E-02	
Sc-46	70	7.2E-02	6.6E-02	3.8E-02	6.6E-02	5.1E-02	
Sc-46	125	4.2E-02	3.8E-02	2.1E-02	4.0E-02	3.5E-02	
Sc-46	400	8.4E-03	5.7E-03	1.1E-03	6.9E-03	5.7E-03	
Co-60	18	3.3E-03	3.2E-03	1.1E-03	3.7E-03	2.8E-03	
Co-60	70	2.2E-03	1.8E-03	2.3E-04	2.1E-03	2.0E-03	2.5E-03
Co-60	125	1.8E-03	1.3E-03	1.3E-04	1.2E-03	1.4E-03	
Co-60	400	5.7E-04	6.2E-04	5.0E-05	3.8E-04	2.2E-04	

CONCLUSIONS

Doses from four hot particles with maximum beta energies between 1 and 0.32 MeV were measured using five different dosimeters. With the exception of the exoelectron dosimeter, the different methods gave good (±30%) agreement for dose averaged over 1 cm² at 70 and 125 μm tissue depths.

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"HOT PARTICLES" IN THE LUNGS OF PEOPLE FROM THE CHERNOBYL ACCIDENT WHO DIED OF ACUTE RADIATION SICKNESS

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INTRODUCTION

Although the "Hot Particles" (*"HP"*) problem is known since the middle of the 1950s there is no common opinion concerning its threat to human health up to date. There exist extreme points of view - from absolute denial of the possibility, as well as any significance of "HP" intake into the lung to obvious exaggeration of "HP" danger (e.g. pulmonary tissue injury, carcinogenic effect) (1-3).

MATERIAL AND RESULTS

Lungs of 27 persons who were irradiated with a high dose-rate of uniform whole body gamma- and beta-irradiation (its range varied from 3.7 to 13.7 Gy) and died of acute radiation sickness within two first months after Chernobyl accident were investigated. Intakes of radionuclides, including "hot particles", by inhalation did not add a significant fraction to the total dose because of short-term exposure, as well as did not influence clinical picture and outcome in these 27 cases.

At the same time this unique material allows to consider "HP" problem at least in two aspects. The first - phenomenological aspect - deals with human lung particles intake, localization, number, size, radionuclide composition, behavior and so on. The second - medical aspect - arises from the necessity to assess the potential "HP" hazard for witnesses who survived after the accident.

Depending on individual location within the next few minutes and hours after the accident all of 27 victims were subdivided into two groups: The first one consisted of 18 persons (the nuclear reactor and turbines operators mainly) who were working indoors the 4th power unit at about 50, 100 and 200 meters' distance from the explosion epicenter. The second group included 9 persons (the firemen, guard and 1 railroader) who were staying outdoors at a distance up to 100, 200 and more meters. The minimal distance from the epicenter was equal to 35 m (the 1th group), the maximal - 1000 m (the 2nd group).

Pulmonary tissue samples, obtained from the central, peripheral and lung root areas of each lobe, were Formalin-fixed, dehydrated, paraffin-embedded and then serially cut at $3 \div 4$ μm. Sorted out by means of autoradiography (RM-1 X-ray film, Russia) the sections with radioactivity were exposed to histoautoradiography test (A-2 emulsion, Russia) and stained with hematoxylin and eosin.

Two types of emitting particles were detected: mixed alpha- and beta-emitting particles, as well as pure beta-emitting particles. The former "hot particle" type dominated. Any special examination of the "HP" radionuclide composition was not performed because it is well known from the literature (4). Being repeated every 2 years (1986, 1988, 1990 and 1992) our histoautoradiography investigation of the same material has confirmed those data. For instance, in 1990 and 1992 histoautographs the incidence of pure beta-emitting particles was found to be markedly decreased, however some of them (^{90}Sr or ^{106}Ru) were still observed.

The most particles were revealed in the lower lobes (central and lung root areas). All "HP" were localized into alveolar macrophages, which filled up some alveolar sacs and bronchioles, lying on alveolar walls and entering their thickness. Such histological picture is evidence of macrophages mediated "HP" mobility (redistribution) all over the lung leading to dose realization not only in strongly limited pulmonary structures, as it is considered, but through a bigger lung volume. On the other hand this fact illustrates the process of radioactivity elimination (sputum macrophages mediated "HP" pulmonary clearance).

We failed to detect "HP" size directly in histoautographs with the use of negative imprint "weakening" technique because of both "hot" and non-"hot" particles overloading of the macrophage-carrier. Another approach (the particle geometrical diameter measuring in 10 macrophages from 10 casual pulmonary tissue fields) has shown the named index to keep within 0.2-1.0 μm.

The maximal "HP" incidence was found in cases which formed the first group. The number of inhaled particles seems to depend on individual patient's location within the period following the accident (i.e. on aerosol clouds spreading) rather than the distance from the explosion epicenter and duration of staying at NPP. Indeed, 2 cases with the similar distance from the explosion epicenter ($35 \div 40$ m) and duration of staying indoors the 4th power unit (about $20 \div 40$ minutes) but different premises location were found to differ in their "HP" amount. The former had more than 20 "hot particles" per histological section (2 cm^2, $3 \div 4$ μm), the latter - 1 "HP" in a hundred of similar sections. Very few, if any, "hot particles" were present in the lungs of the second group. Like the first one there was no association between the "HP" incidence and the distance from the accident epicenter. Attention needs to be paid to respirator using (3 cases) which produced sufficient lung protection.

It must be noted that inferences drawn from present study are correct only for persons staying at NPP within the next hours after the accident. Some time later aerosol distribution becoming uniform resulted in risk "HP" inhalation for many witnesses.

Even in cases with the maximal "HP" loading of the lung, their number per macrophage appear not to exceed some units. Therefore macrophage activity value seems to correlate with one particle activity. For alpha-emitting particles it was found to keep within $5 \cdot 10{-}6 \div 8 \cdot 10^{-5}$ Bq (DTD CR-30 TASTRACK estimate; data presented be A.Marenny). Taking into account fuels' alpha- and beta-emitting particles ratio it could be considered that "HP" total mean activity was equal to $0.3 \div 5.0$ Bq (data presented by V.Kutkov).

"HP" short-term exposure in these 27 cases does exclude any biological effect development.

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EQUIPMENT AND STANDARDS FOR SURFACE CONTAMINATION MEASUREMENTS

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INTRODUCTION

The measurement of the surface contamination level with alpha and beta emitters is made generally by using contaminometers, equipped with large area detectors; they must satisfy some technical requirements, imposed by the International Standards [1], the most important being their response as function of the contamination level of the surface, R, [s^{-1} /(Bq cm^{-2})]. On this purpose such contaminometers must be developed and a wide variety of large area alpha and beta sources must be used for their calibration. These sources must also fulfill some requirements [2] regarding their activity, particle emission, uncertainty, uniformity. The present paper presents the equipment and methods used for the preparation and standardization of alpha and beta sources, as well as the main characteristics of a portable, alpha-beta contaminometer, developed in our Laboratory.

1. PREPARATION AND STATEMENT OF THE STANDARD SOURCES ACTIVITY

The most widely used large area standard sources are prepared by the adsorption of the radioactive material in a thin aluminium oxide layer obtained by the anodization of the surface of an aluminium plate [3]. The particle emission is measured directly by using large area windowless proportional counters [4], but their activity can not be stated, as the preparation procedure is not quantitative. In the case of thin alpha sources, the activity may be calculated from the particle emission with 2Πsr geometry from the relationship

$$A = 2E_{\alpha} \qquad (1)$$

where A is the activity [Bq], E_{α} is the 2Πsr particle emission [s^{-1}]. In the case of beta sources, due to self absorption and backscattering, relation (1) is no more valid. This is the reason for which, a long period, the well-known producing laboratories certified only the particle emission, but not the activity; only recently, an indirect method for determination of the activity from the emission value was developed by Janssen et al [5].

We adopted the following methods for the preparation of the sources . The alpha sources were prepared electrolitically from ^{241}Am, using stainless steel discs as supports; a maximum active diameter ϕ =22mm was obtained. The emission in a 2Πsr geometry is measured, and the activity is calculated according to the relation (1). The beta sources were prepared by the quantitative dispensing of standard solutions of ^{35}S, ^{147}Pm, ^{204}Tl, ^{90} (Sr+Y), on thin paper foil[6], 3mg.cm^{-2} thick. Equal mass drops, about 10 mg each were dispensed in each square centimeter marked on the foil; after drying, the paper was sealed between plastic foils 3mg.cm^{-2} thick and fixed on aluminium plates with a thickness of 3 mm; an uniformity better than 7.5% on the whole surface was found. The sources are recognized by the Romanian Bureau of Legal Metrology (RBLM), obtaining the certificate nr.5353/1991. The solutions were standardized absolutely by using the 4ΠPC proportional counter-γ coincidence equipment, 4ΠPC or efficiency tracer method [7], recognized as secondary standard of Romania [8,9]. The activity of the sources, A[Bq], was raced calculated as:

$$A = a.m \qquad (2)$$

where a is the activity concentration of the solution [Bq. g^{-1}], m is the mass of standard solution[g]. A maximum expanded uncertainty of ±(2-4)% for a 99% confidence level was obtained. The 4ΠPC characteristics were fully described in the paper [10].

2. MEASUREMENT OF THE PARTICLE EMISSION IN 2Πsr GEOMETRY

2.1. Windowless 2Πsr proportional counter.

Alpha and beta sources with a diameter inferior to 30 mm are measured directly in the superior half of the 4Π proportional counter, working in the regime of alpha or beta detection. The saturation backscattering corrections for ^{90}(Sr+Y) is 7% and is ensured by using the 3mm Al support; for other radionuclides, this correction is less than 1% and may be neglected.

For the large area sources emission measurement, we applied two parallel methods. One of them is based on the use of the 2Πsr windowless proportional counter and consists in the followings. During the preparation of the large area source, 3-4 sources with a maximum diameter ϕ=26 mm were prepared quantitatively from the same solution by using the same materials (paper and plastic foil). Their emission values, $E_{\beta i}$ [s^{-1} /2Πsr] were measured in the 2Πsr windowless proportional counter, and their individual efficiencies $\varepsilon_{\beta i}$ [s^{-1} / Bq] were calculated as:

$$\varepsilon_{\beta i} = E_{\beta i} / A \qquad (3)$$

The mean efficiency, ε_β and its standard deviation $S\varepsilon_\beta$ were calculated from $\varepsilon_{\beta i}$ values. The backscattering correction factors, f, were applied and the large area source emission was then calculated by using the relationship $E_\beta = f.\overline{\varepsilon}_\beta A$ (4) Table 1 represents some experimental values which allowed to verify the validity of the method. It was recognized by the RBLM as a technical Procedure [11] ($f\overline{\varepsilon}_\beta$) is the mean efficiency of a radionuclide.

Table 1

Radionuclide	Solution concentration a ,[Bq/g^{-1}]	$\overline{\varepsilon}_\beta$, [s^{-1}/Bq]	$S\overline{\varepsilon}_\beta$,%	f	$(\overline{f\varepsilon_\beta})$
(Sr+Y)-90	36560	0.4447	±1.5	1.07	0.4851± 1.1
	7165	0.4651	±0.9		
	1580	0.4460	±1.6		
	954	0.4579	±1.4		
Tl-204	88600	0.4143	±0.4	1.00	0.3985±4.7
	4964	0.3610	±3.5		
	1890	0.4203	±1.2		
Pm-147	21050	0.1305	±4.0	1.00	0.1243±4.1
	1972	0.1257	±4.1		
	1006	0.1168	±4.5		

The second method is presented at point 2.2

2.2. Large area 2Πsr proportional counter

A multiwire proportional counter, working with pure methane flow at the atmospheric pressure, 250x250 mm area, with an aluminized Mylar window, thickness 0.85 mg/cm was used for the direct measurement of the particle emission of the large area sources. It is placed in a 5cm lead shielding and is provided with a 4mm plate fixed at 5mm distance from the window, where the sources to be measured are placed. The background counting rates are 0.1s for alpha operation and 10^{-1} for beta. When a 2400 V high voltage. Two main characteristics of the counter were followed : the response of the detector and the edge effects. The response is defined both as the ratio between the counting rate, and emission E_β, R_1=N/E [s^{-1}/s^{-1}/2Πsr] or the ratio between the counting rate and activity R_2=N /A [s^{-1}/Bq]. In the case of alpha sources, seven ^{241}Am sources, prepared in our laboratory, with the activity in the interval (19-17000)Bq and four ^{239}Pu of the same type delivered by the LMRI-France with the activities interval (1.8-3600)Bq were used. The obtained values of R_1 and R_2 are presented in the table 2.

Table 2

Radionuclide	$\bar{R}_1 \pm \bar{S}_{R1}$	$\bar{R}_2 \pm \bar{S}_{R2}$
Am-241	0.748±1%	0.374±1%
Pu-239	0.712±3%	0.356±3%

In the case of beta sources, several sources of ^{137}Cs, ^{90}(Sr+Y), ^{204}Tl, ^{147}Pm with different dimensions, from point to (150 x 100) mm and separately with (220 x 180)mm, (250 x250)mm for ^{90}(Sr+Y) were used. The obtained results are presented in the table 3.

Table 3

Radionuclide	Source dimension	$\bar{R}_1 \pm \bar{S}_{R1}$	$\bar{R}_2 + \bar{S}_{R_2}$
Cs-137	point	0.94+3%	0.446+1.0%
(Sr+Y)-90	<(150 x 100) mm	0.94+3%	0.457+5.0%
	(250 x 250) mm	0.80+3%	0.403+6.0%
Tl-204	< (150 x 100) mm	0.95+3%	0.358+6.0%
	(220 x 180) mm	0.86+3%	0.335+6.0%
Pm-147	< (26 x 26)mm	0.83+3%	0.111+5.0%
	(150 x 100) mm	0.74+3%	0.092+7.0%
	(220 x 180) mm	0.64+3%	0.086+7.0%

One may notice the reproducibility of the results and the absence of edge effect for dimensions of the sources less than 150 x 150 mm. The same study, of the edge effect was made with a point ^{90}(Sr+Y) source measured at different distances from the center of the counter. The results are in good agreement with those presented in the table 3.

The system was certified by the RBLM certificate numbers 1826/1992, 1837/1993.

3. THE PORTABLE CONTAMINOMETER

A portable contaminometer, provided with two multiwire proportional counters (for alpha and beta detection) having an active area of 50 cm^2 and a window thickness of 0.85 mg.cm^{-2} working in regime of propane flow was developed. It was calibrated with large area alpha and beta sources presented at the points 1 and 2 and meets all the requirements of the IEC 325/81 document. The contamination level ranges are : for alpha contaminants ($8.4 \cdot 10^{-2}$ - $8.4 \cdot 10^{2}$)Bq/cm^2 and for beta contaminants ($0.2\text{-}0.67 \cdot 10^{3}$)Bq/cm^2 . It may be used continuously for a 20 hours period and is provided with an alarm system, announcing contamination levels superior to a preestablished threshold.

CONCLUSIONS

1. The equipment, standards sources and measurement methods developed in our laboratory offered us the possibility to solve completely the problem of the measurement of the contamination of the surfaces with alpha and beta emitters.

2. The traceability at both national and international level of the equipment, standards and methods used was once again demonstrated.

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IRPA9

1996 International Congress on Radiation Protection

April 14-19,1996

Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
(Instructions for preparation on reverse)

Abstract No.
Receipt
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PAPER TITLE TESTS OF SURFACE CONTAMINATION MONITORS IN ALPHA AND BETA RADIATION STANDARD FIELDS AT THE CALIBRATION LABORATORY OF SÃO PAULO

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

The need for effective monitoring of surface contamination is quantified in terms of activity per unit area; this quantity is used to specify the derived limits, that are incorporated in the national regulations of radiation protection. These regulations recommend the calibration of all health physics instruments every 12 months. In the case of surface contamination monitors, a special set-up and a method were developed at the Calibration Laboratory of São Paulo, using standard alpha and beta radioactive sources. At São Paulo the monitors are being calibrated using ^{241}Am, ^{14}C, ^{36}Cl and ^{90}Sr + ^{90}Y sources, and the instrument efficiency is determined for each case for alpha and beta radiation. Moreover, all instruments are being normally tested also with ^{241}Am sources of different activities (response linearity) and with ^{137}Cs, ^{244}Cm, ^{233}U, ^{238}Pu and ^{239}Pu sources (energy dependence). Instruments of different manufacturers and types were used for this study.

Work partially supported by CNPq - Conselho Nacional de Desenvolvimento Científico e Tecnológico, Brazil

SEMICONDUCTOR DETECTORS FOR DETERMINATION OF RADIONUCLIDE CONTAMINATION IN THE SUBSTANCES AFTER THE CHERNOBYL ACCIDENT

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1. INTRODUCTION

The development of rapid and sensitive devices for the monitoring systems of α-radinuclides in the environment is of a paramount importance, especially after Chernobyl accident. Alpha-spectrometry methods based on Si-semiconductor detectors proved to be among the most reliable ones up to now. Different kind of detectors have been developed and fabricated by authors for that purpose by means of the surface-barrier as well as of the planar technology. The description is presented for the Multichannel-Alpha-Radiometer-Spectrometer ("MARS-16") designed on the basis of the developed detectors for rapid measurements of the concentration of alpha-radionuclides in the environment (soil, fluids, air) and food. Strip-detector option of the "MARS-16" could be used for the high accuracy position determination of the alpha-radioactive contamination.

2. EXPERIMENTAL RESULTS

To provide alpha-spectrometry systems with high quality semiconductor detectors different technologies for the fabrication of the planar detectors were studied. The main physics-technology limitations for the production of silicon planar detectors with different topological configurations were determined.

The detectors were fabricated on the base of n-type silicon plates with diameter of 3 inch, <100> orientation and ~ 1 kOhm·cm resistivity. The plates were oxidized in water vapour with an addition of the 3-chlorine-ethylene at the temperature of 850-1000°C. Vapour-borned Si-oxide as well as a many layer dielectric with a Si-nitride were used for dielectric layers.

p^+-n junction was made by ion implantation of boron at energy E = 40 keV, ohmic contact on the rear side was made by implantation of phosphorus at energy 60 keV. The rear side and p^+-n junction of the detectors were metalized by 0.3 μm thick Al.

Planar detectors on the base of high resistivity silicon with area of 1 cm^2, 16 cm^2 and passive-mosaic detectors with total area of 24 cm^2 were developed. The application of passive-mosaic structures allows to produce the detectors with large active area for integral alpha-activity measurements. The energy resolution of fabricated detectors with area of 16 cm^2 and sensitive layer thickness of 250 μm for 5.5 MeV α-particles was 70-90 keV at the bias voltage of 50 V and leakage current of 60-100 nA. For the detectors with area of 1 cm^2 the energy resolution was 25-30 keV at bias voltage 20-60 V and leakage current 20-40 nA. Such detectors were used for express analysis of radionuclides in samples from various regions

contaminated during Chernobyl accident.

In particular, large area Si-detectors (up to 16 cm^2) have been used in the Multichannel-Alpha- Radiometer-Spectrometer ("MARS-16") designed mainly for rapid measurements of the concentration of alpha-radionuclides. 16 Si-detectors are housed in a vacuum chamber equipped by special preamplifiers with low noise for detectors with large capacitance. The vacuum chamber is inserted into a crate CAMAC together with amplifiers, strobe mixer, analog unit and crate controller linked to PC IBM-486. The developed software allows for analysis of spectra to extract values of alpha-concentrations by means of thin and thick samples methods (1). Fig.1 represents the example of the alpha-spectrum measured by one of "MARS-16" channels for the thin probe prepared by means of radiochemical separation of Pu-isotopes using ^{236}Pu as a tracing element.

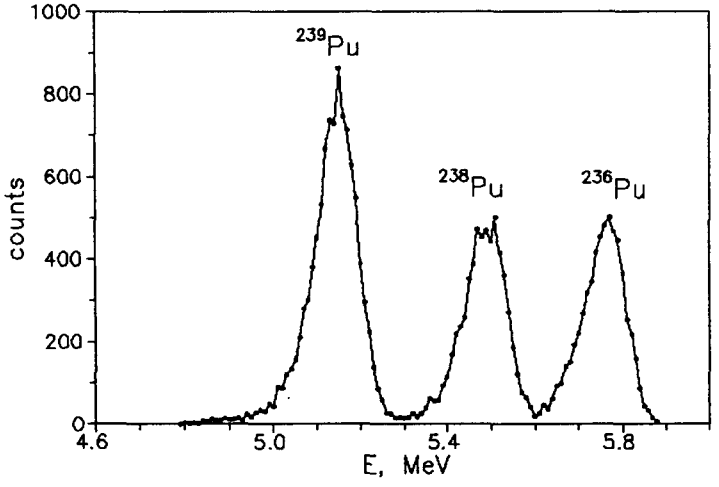

Fig.1. Alpha-spectrum measured for the thin probe prepared by means of radiochemical separation of Pu-isotopes.

The remarkable feature of "MARS-16" is an ability to provide selective alpha-radioactivity monitoring avoiding time consuming and expensive procedure of radiochemical preparation of samples. Fig.2 represents the example of an alpha-spectrum measured by "MARS-16" directly from the contaminated soil sample (histogram). Dashed lines show the contribution of various alpha-emitting isotopes, while the solid line represents their sum, as it follows from the Monte-Carlo simulation. We could easily reach the sensitivity to $^{238,239}Pu$ concentrations (at 20% statistical error) up to 20 Bk/kg without radiochemical sampling procedure and measuring time ~ 20 hours. Results obtained by means of thin and thick sample methods generally are in a good agreement , although in some cases the thick sample method gives higher concentration values.

Earlier unknown features for alpha-radiometry are introduced by strip-detectors widely used in high energy physics. Extremely good energy resolution (up to few keV for minimum ionizing particles) combined with high position determination accuracy even at very high counting rates

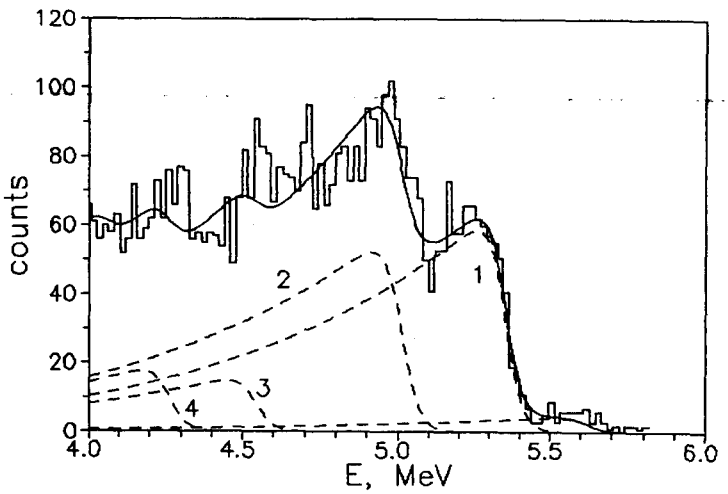

Fig.2. Alpha-spectrum measured for thick sample of soil. Dashed lines correspond to the contributions of different isotopes with distinguished energy of α-radiation: 1 - ^{238}Pu; 2 - ^{239}Pu; 3 - ^{234}U; 4 - ^{238}U. Solid line is their sum.

(up to hundred kHz for the total area of strip-detector) characterizes a strip-detector as an exceptional device for alpha-radiometry.

The design of the 128-channel Si strip-detector with submicron position sensitivity for alpha-radiometry purposes has been developed. Some information on the characteristics of fabricated and studied strip-detectors could be found in the Proceedings of this Conference. High accuracy of the 2-dimensional position determination is realizable by means of double-sided strip-detectors. The position accuracy for both "X" and "Y" coordinates could be essentially improved by a fabrication of the strip-detector with special strips configuration and exploring the method of submicron position sensitivity (2).

ACKNOWLEDGMENTS

This investigation is partly supported by International Atomic Energy Agency under the research contract N7200/R2/RB.

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CALCULATION OF FLUENCE-TO-DOSE EQUIVALENT CONVERSION COEFFICIENTS FOR NEUTRONS TO BE USED FOR CALIBRATION OF PERSONAL DOSIMETERS

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ABSTRACT

Conversion coefficients from neutron fluence to the ICRU slab dose equivalent for personal dosimeter calibration were calculated using the MCNP-4 code and the cross section library based on JENDL-3.1. Influences given by the changes of neutron quality factor and phantom size were also investigated.

INTRODUCTION

The international Commission on Radiation Units and Measurements(ICRU) has recommended in its report No.47(1) that the dose equivalent at the depth of d in a 30×30×15cm ICRU tissue slab phantom, ICRU slab dose equivalent, $H_{si}(d,\alpha)$ (α: incident angle), should be used as the quantity for calibration of personal dosimeters. Though the scope of that report focuses on photons and electrons, the consistent calibration procedure requires the same quantity for neutrons as well. Conversion coefficients from neutron fluence, Φ, to $H_{si}(d,\alpha)$ for a broad parallel beam of neutrons incident in angle α are needed to apply the quantity $H_{si}(d,\alpha)$ to calibration of personal dosimeters. There are only a few published reports(2-4) on $H_{si}(d,\alpha)/\Phi$. To improve the reliability of the conversion coefficients, the calculation should be made with different techniques and nuclear data. In addition, neutron quality factors, Q_n, that mean the average quality factors for the neutron-induced heavy charged particles, have been recalculated based on the new data on stopping powers and ranges for protons and alpha particles(4), thus the influence of this alteration on $H_{si}(d,\alpha)/\Phi$ needs to be investigated. To clarify the contribution of neutrons and secondary photons to the dose in the phantom, it is important to compare the values of the conversion coefficients calculated for the phantoms with different size. This paper presents the calculated values of $H_{si}(d,\alpha)/\Phi$ at depths of 0.07, 3, and 10mm in the range from thermal to 20MeV in energy and from 0°(normal incidence on the surface of the phantom) to 75° in incident angle, and the influences of the alteration of Q_n and of the difference of the phantom size on the conversion coefficients are also described.

METHOD

The Monte Carlo code MCNP-4 was used in the calculation. The surface crossing estimators were employed to calculate spectral fluences of neutron and neutron-induced photon at depths of 0.07, 3, and 10mm on the mid-axis within the ICRU tissue slab phantom, irradiated by a broad parallel beam of neutrons incident in angle α. This type of estimator would be more effective than volume-type ones to evaluate the fluences that vary significantly with depth in the phantom. The estimator was a disk with a radius of 2cm, which was chosen by test calculations. The number of particle histories was selected dependently on energy and incident angle so that one relative standard deviation could be maintained within 1%.

The continuous energy cross section libraries employed in this work are FSXLIB-J3(5), generated from JENDL-3.1 evaluated in Japan(6), for neutrons, and MCPLIB, based on data from ENDF, for photons. Since $S(\alpha,\beta)$ data for treatment of chemical binding effect in thermal neutron scattering are not available for ICRU tissue, the data for hydrogen bound in light water were adopted in the calculation.

The absorbed dose was calculated by the kerma approximation. The kerma factors were taken from the data of Caswell *et al*(7) for neutrons and of Hubbell(8) for photons. The neutron quality factors used in the calculation of dose equivalent are the data of Schuhmacher and Siebert(9). In order to avoid the round error due to the structure of energy bins, we did not use energy bins, but each particle was weighted by the kerma factor and Q_n corresponding to the particle energy on the estimator.

RESULTS AND DISCUSSION

Figure 1 shows the conversion coefficients for normal incidence, $H_{sl}(d,0°)/\Phi$, as a function of incident neutron energy. $H_{sl}(d,0°)/\Phi$ increases significantly with energy in the range between 10keV and 1MeV due to a rapid augmentation of kerma factor in this energy region. No significant energy dependence of $H_{sl}(d,0°)/\Phi$ is observed in the low energy region (below ~10keV). This is because the contribution of slowing down neutrons to the dose is predominant in this energy region. $H_{sl}(d,0°)/\Phi$ increases with depth in the low energy region, and the values at 10mm depth are roughly two times higher than those at 0.07mm depth. The contribution of secondary photons to $H_{sl}(d,0°)/\Phi$ is also plotted in Figure 1. The secondary photons give 20-30% of dose equivalent in the low energy region, whereas they give quite a small contribution in the high energy region.

A variation of the conversion coefficient at 10mm depth with incident angle, α, is illustrated in Figure 2 as the angular dependence factor, defined as the ratio of $H_{sl}(10,\alpha)/\Phi$ to $H_{sl}(10, 0°)/\Phi$. As is apparent from this figure, $H_{sl}(10,\alpha)/\Phi$ depends significantly on the incident angle, and the angular dependence factors in the low energy region vary slightly due to the same reason in the above.

The calculated values of $H_{sl}(10,\alpha)/\Phi$ were compared with the data of the previous works(2,3). As a whole, the present results agree with the data of other authors within 15%, even though the cross section library and the estimator were different from theirs. In detail, the present values are about 5% less than those of the previous works in the high energy region, and the discrepancies increase up to about 15% with decreasing energy. Major differences in the method between the previous works and ours are cross section library, estimator, and the treatment of kerma factors below 0.025eV. They used the cross section libraries based on ENDF/B-IV, volume-type estimators, and the 1/v-dependent kerma factors below 0.025eV, whereas we used the library based on JENDL 3.1, surface crossing estimators, and the constant factors below 0.025eV though the data source is the same. The influence of each of these differences is now being studied.

The conversion coefficients $H_{sl}(d,\alpha)/\Phi$ using the revised Q_n (4) was calculated for normal incidence. Q_n was reestimated to be smaller than the previous one. Consequently, this reduces the conversion coefficients as seen in Figure 3. A large descent appears at several tens of keV.

Influence given by the change of the phantom size was investigated. A comparison of the values of $H_{sl}(d,0°)/\Phi$ calculated for a 40×40×15cm phantom with those for the 30×30×15cm one shows that the enlargement of the phantom leads to 3-10% growth of the conversion coefficients in the range below 10keV. A further comparison of each component of neutrons and secondary photons demonstrates that secondary photons induced in the phantom, rather than neutrons, have an important contribution to dose for neutrons below about 10keV.

CONCLUSIONS

$H_{sl}(d,\alpha)/\Phi$ for personal dosimeter calibration was calculated using the MCNP-4 code and the cross section library based on JENDL-3.1. The present results of the $H_{sl}(10,\alpha)/\Phi$ agree with data of other authors within 15%, even though the cross section library and the estimator were different from theirs. It is found that the recalculated Q_n reduces $H_{sl}(d,\alpha)/\Phi$ in all energies considered. A study on the influence given by the change of the phantom size makes it clear that secondary photons induced in the phantom, rather than neutrons, have an important contribution to dose for neutrons below about 10keV.

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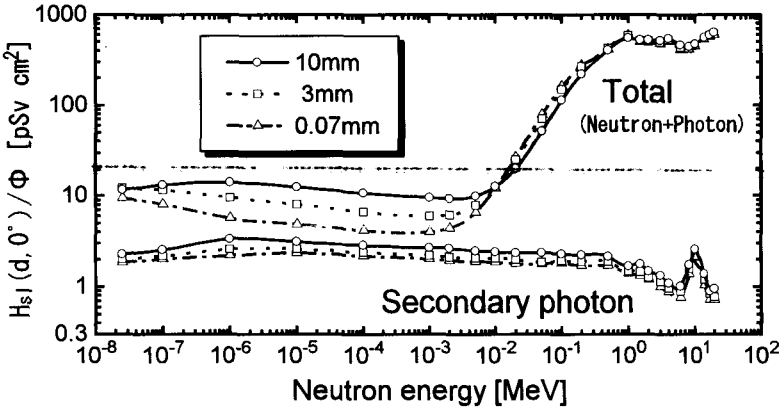

Figure 1. Fluence-to-ICRU slab dose equivalent conversion coefficients, $H_{sl}(d, 0°)/\Phi$, for a broad parallel beam of neutrons incident normal to the phantom, and their component of secondary photons.

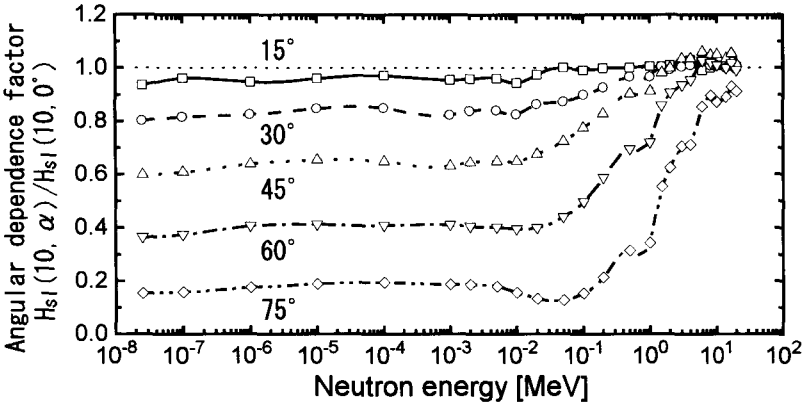

Figure 2. Angular dependence factors of ICRU slab dose equivalent.

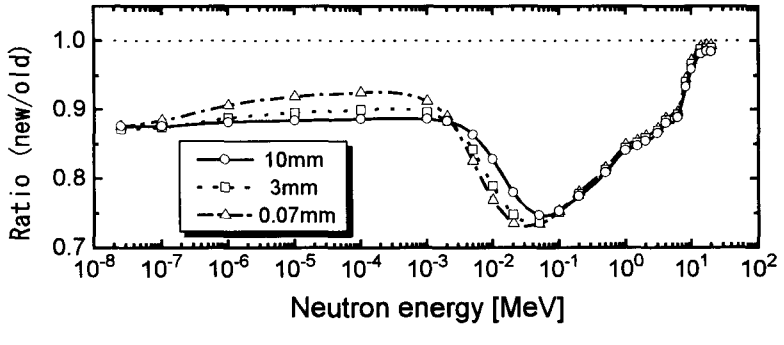

Figure 3. Ratios of $H_{sl}(d,0°)$ calculated with the new(4) and old(9) neutron quality factors.

APPLICATION OF ION-EXCHANGE MEMBRANE SOURCE TO PREPARATION OF CALIBRATION SOURCE SIMULATING FILTER PAPER FOR DUST MONITORS

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ABSTRACT

Ion-exchange membrane sources were applied to preparation of beta-ray calibration sources simulating filter paper for dust monitors. Concerning the influence of backscattering on counting efficiencies of detectors, the similarity between the filter paper and the prepared sources is discussed. The sources were proved to be more suitable for the realistic calibration of dust monitors.

INTRODUCTION

A calibration source for dust monitors consists of an active layer and a backing material(for example, aluminum or stainless-steel). Usually, the backing is thick enough to prevent the penetration of beta-rays backscattered by structural materials behind the filter paper. The sources can be standardized in surface emission rate[1)] but not in radioactivity. However, the actual filter is relatively thin and penetrable for the beta-rays. When the radioactive aerosols collected on the filter are measured, the backscattering radiation influences the counting efficiency of detectors. Taking account of the backscattering effect, the calibration source requires (a)the similarity in material to the filter paper and (b)the standardization in radioactivity. Taking these matters into consideration, the filter-simulated sources with ion-exchange membrane(hereafter, FSS) were prepared. This paper presents the preparation method of the calibration sources, and the discussions on such characteristics of the sources as self-absorption, release of radioactivity and the effect of backscattering.

EXPERIMENTS

The ion-exchange membrane sources were prepared by soaking ion-exchange membranes into the solution in which radionuclides and their carriers were mixed, washing and drying them.[2)] Three types of membrane sources(about 0.7, 3 and 6 mg/cm^2 in the thickness) were prepared for each radionuclide of ^{60}Co, ^{137}Cs and ^{90}Sr+^{90}Y. The activities of ^{60}Co and ^{137}Cs membrane sources were determined by the manner of relative measurement using a well-type HPGe detector. For the ^{90}Sr+^{90}Y membrane source, its activity was determined with ^{85}Sr as a tracer. The sources consist of protective film(polyester), the ion-exchange membrane source(polyethylene), adhesive(acrylic family) and backing material(polyester). The total thickness of FSS was approximated to that of filter paper (12.7 mg/cm^2). Experiments were made without the protective film and with two types of detectors, a plastic scintillation detector and a GM tube. The counting loss of beta-rays by the protective film(0.28 mg/cm^2) was about 2 % for ^{60}Co and less than 1 % for ^{137}Cs and ^{90}Sr+^{90}Y.

Since the active layer of FSS is about 0.7 to 6 mg/cm^2 thick, beta-rays are self-absorbed in the active layer. For evaluation of the self-absorption, an experiment was made. A thin ion-exchange membrane source(about 0.7 mg/cm^2) and several polyester membranes were layered. The total thickness of the layer was approximated to 12.7 mg/cm^2. The membrane source was inserted into any position of the polyester layer, the total thickness being kept constant. The beta-rays were measured for each position of the membrane source. Then, an absorption curve for each radionuclide was obtained as a function of source position. The correction factor of self-absorption was estimated by integrating the absorption curve in respect of the source position.

The radioactive aerosols are assumed to be collected on the surface of actual filter paper. In

order to compare the FSS with the actual filter paper, the calibration sources of which radioactivity was adhered on the surface of filter paper(hereafter, RAS) were prepared.

RESULTS AND DISCUSSION

Table 1 gives the counting efficiencies of plastic scintillation detector and GM tube measured with the RAS and the FSS. As seen in Table 1, the detectors can be calibrated with the FSS as well as the RAS. However, as for $^{90}Sr+^{90}Y$, the counting efficiencies for the FSS are systematically somewhat smaller than those for the RAS. The correction factors of self-absorption for each radionuclide with the plastic scintillation detector are shown in Fig. 1. The correction of $^{90}Sr+^{90}Y$ FSS is 2 % at most(in the case that the active layer is about 6 mg/cm^2 thick). Hence, the reason for the smaller efficiency is considered to be possible overestimation of the activities of $^{90}Sr+^{90}Y$ FSS.

Measurements with a dust monitor include the backscattered beta-rays by structural materials behind the filter paper. By using the FSS, the influence of backscattering was evaluated for three types of filter paper support. One of them is a stainless-steel mesh plate usually attached to the dust monitors. The others are stainless-steel and aluminum plates of 2 mm in thickness. The counting efficiencies were measured with these different supports and without supports. Although there are some structural materials behind the supports, the influence of backscattering is considered to be small because of sufficient distance between the support and the material. For $^{90}Sr+^{90}Y$, the efficiencies increase to 20 % or more using the stainless-steel plate, and to about 8 % using the typical mesh plate, in comparison to the efficiencies without supports. These results indicate that the influence of backscattering behind the filter paper is not negligible for more realistic calibration of dust monitors.

The counting efficiencies of plastic scintillation detector for the sources with a thick backing are shown in Fig. 2. The thick backing can prevent the penetration of beta-rays backscattered behind itself. The activities of the sources are assumed to be twice the surface emission rate measured by a windowless 2π-proportional counter. In the figure, the efficiencies for the FSS are also shown. The mean beta-ray energies of ^{60}Co and ^{137}Cs referred to the NCRP Report[3], and that of $^{90}Sr+^{90}Y$ was derived from the spectrum by a plastic scintillation spectrometer. The FSS makes the backscattered beta-rays penetrate through itself, and is standardized in radioactivity. The counting efficiency for the FSS includes the influence of backscattering, and the efficiency for actual filter paper as well. The differences in thickness and reference quantity of calibration sources reflect on the calibration of dust monitors.

When an unsealed source is handled in routine calibration works, the release of radioactivity from the source may lead to accidental contamination. Therefore, the RAS is not suitable to the routine calibration works. The ion-exchange membrane source is free from the contamination because the release of radioactivity is extremely small. With the protective film, the FSS becomes much safer against the contamination.

CONCLUSION

The FSS can be used for the calibration of dust monitor in consideration of the influence of backscattering. Although the active layer of FSS is several mg/cm^2 thick, its self-absorption can be easily estimated due to its simple structure. Especially, this source is more suitable for the realistic calibration if the main radionuclides are fixed in nuclear facilities. Moreover, the FSS can be safely handled in the routine calibration because of negligible release of radioactivity. This study confirmed that this type of source was useful for the calibration of dust monitors.

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Table 1 Counting efficiency of detectors for several sources.

Detector	Radio-nuclide	RAS*1	FSS*2		
			0.7 mg/cm^2	3 mg/cm^2	6 mg/cm^2
Plastic scintillation detector	^{60}Co	0.128±0.001	0.128±0.001	0.125±0.001	0.125±0.001
	^{137}Cs	0.217±0.001	0.213±0.002	0.218±0.003	0.212±0.002
	^{90}Sr+^{90}Y	0.272±0.002	0.252±0.005	0.261±0.005	0.265±0.004
GM tube	^{60}Co	0.159±0.001	0.157±0.002	0.154±0.001	0.155±0.001
	^{137}Cs	0.257±0.001	0.249±0.003	0.248±0.003	0.252±0.002
	^{90}Sr+^{90}Y	0.308±0.002	0.284±0.006	0.289±0.005	0.293±0.004

*1 Source of which radioactivity is adhered on filter paper.
*2 Filter simulated source with ion-exchange membrane.

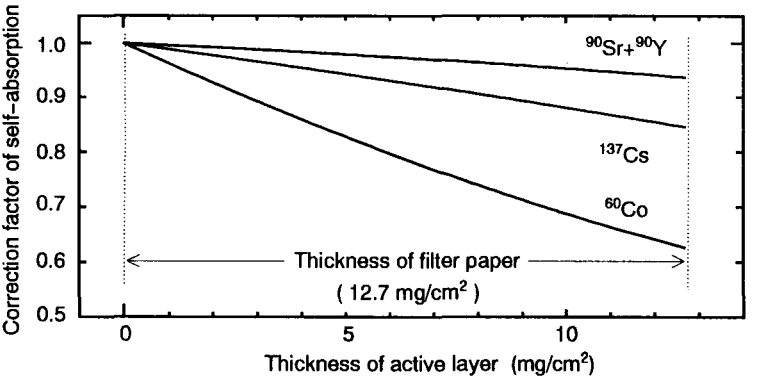

Fig. 1 Correction factor of self-absorption for each radionuclide with plastic scintillation detector.

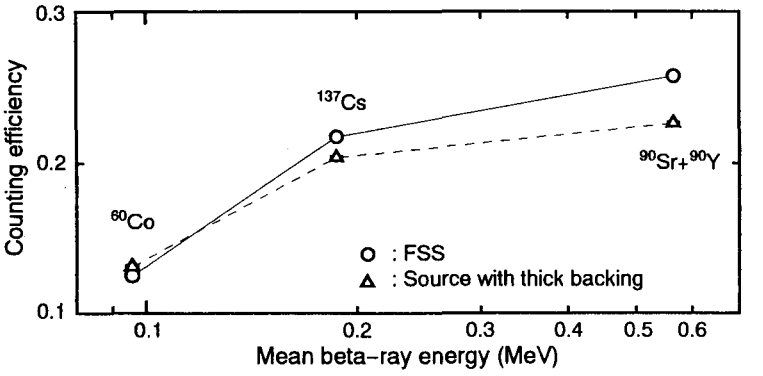

Fig. 2 Counting efficiency of plastic scintillation detector for two types of source.

Mitigation of radiation dose through deposition to indoor surfaces

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Introduction

When an accident occurs that involves releases to the atmosphere of toxic materials the first action of the civil defence authorities is to urge people to go indoors and close the windows and doors. A review of measurements of Indoor/Outdoor concentration of airborne pollutants (e.g. Alzona et al. (1979)) reveals that most substances of outdoor origin have a lower concentration in indoor air, but a more detailed knowledge of the magnitude and mechanism of this reduction is disireable.

Since inhalation dose is directly proportional to air concentration (for a given particle size) the dose reduction factor, DRF, can be defined theoretically as the ratio between the indoor pollutant concentration, C_i, and the outdoor pollutant concentration, C_o, integrated from the start of the cloud passage, t_o, to infinity. Considering the house to be a single box with an air exchange with an infinite outdoor volume a simple differential equation can be derived by equating the change in indoor concentration per unit time with the difference between the production (ingression from the outside) and loss of particles (indoor deposition). This equation can be solved analytically. The solution this equation shows that the time integrated DRF equals the equilibrium ratio for constant outdoor pollutant concentration:

$$DRF = \frac{\int_{t_0}^{\infty} C_i(t)dt}{\int_{t_0}^{\infty} C_o(t)dt} = f\frac{\lambda_r}{\lambda_r + \lambda_d}$$

where f is the fraction of particles in outdoor air penetrating the building envelope, λ_r is the air exchange rate and λ_d is the deposition constant. This is a general result for all shapes of passing clouds, as emphasized by Roed et al. (1991), when it is assumed that λ_r and λ_d are constant. The air exchange rate, λ_r, may change if the weather conditions change, but in all circumstances the value used in modeling will be an average value, which again will give an average value for the dose reduction factor. The deposition constant, λ_d, will vary with the particle size, but for each size class and type of pollutant the expression will be valid. Some authors , e.g. Engelmann (1992), have given a more optimistic expression for the DRF by including a term describing ventilation of the house right after the passage of an rectangular cloud and thus reducing or removing the contribution to dose from the 'tail' after passage, but this factor has been found to be to unrealistic to be included in the model.

Experimental

The indoor deposition is the least well known parameter determining the indoor inhalation dose. Very little information exists on the depostion of particles larger than 0.5 μm in houses. In order to improve our understanding of the mechanisms governing the indoor-outdoor air activity ratio Imperial College and Risø National Laboratory have developed a particle tracer technique, where silica particles are labelled with dysprosium and used as tracers for indoor depostion experiments. The idea is to disperse the particles in a real house and measure the decrease in tracer concentration by taking consecutive air filter samples. During the experiment, the air-exchange rate is measured and the deposition constant, λ_d was then to be found by subtracting the air exchange rate, λ_r, from the decay constant, λ_t. The air-exchange rate was measured by releasing SF_6 gas into the test room and monitoring the decrease in concentration by gas chromatography. The absolute decrease with time of both the tracer gas and the particle concentration is proportional to the concentration and the decay will thus follow an exponential curve if the experimental conditions are constant during the test. Both these decay constants are found by linear regression. For all the four houses studied the deposition velocity, v_d, has been calculated using the geometric surface, S, to volume, V, ratio. That is, no contribution from the surface of furniture, etc. has been included in the surface area of the furnished rooms. Such measurements would be difficult to make in an objective and reproducible way.

Results and discussion

Results and a detailed description of the individual experiments in the four houses have been given by Roed et al. (1991), Byrne (1995) and Lange (1995). Despite the differences between the test conditions in the various houses, the results are in good agreement, with increasing deposition velocity for increasing particle size and degree of furnishing. Table 1 shows the average results for furnished and unfurnished rooms. As tests only

were made with furniture in the room during the Jersie experiment there is actually only 'unfurnished' results from three houses. In all experiments, the deposition velocity was highest under the furnished conditions. In Table 1 the deposition can also be seen to increase with particle size as predicted by deposition theory for supra-micron particles, but the actual deposition velocities exceeds those predicted by the theories that only includes gravitational settling by a factor of 2 to 5. When the results were compared with the measurements of indoor deposition presented in Roed & Cannell (1987): Table 2 good agreement was observed. For Be-7 which is associated with particles in the size range of 0.5 to 1.0 μm, Lange (1994), an average deposition velocity of ~~0.71 x 10^{-4} ms^{-1}s was found. The indium particles had a AMAD of 0.5 to 0.7 μm, which is close to that of Be-~~ 7, and the deposition velocity has been found to be 0.61×10^{-4} ms^{-1} for unfurnished rooms and 0.82×10^{-4} ms^{-1} for a furnished room on average. Roed and Cannell found a v_d of 3.1 to 3.9×10^{-4} ms^{-1} for Ce-144. This corresponds to the v_d of 4 or 5.5 μm particles in Table 1 and this would be reasonable size for cerium as it belongs to refractory group of release products as described in Rulik et al. (1987).

Size [μm]	GSD [ ]	Avg. v_d Unfurnished [$10^{-4}ms^{-1}$]	Avg. v_d Furnished [$10^{-4}ms^{-1}$]
0.5	1.60	0.61±0.08(2)	0.82±0.08(6)
2	1.48	1.13±0.16(5)	1.36±0.50(5)
3	1.20	1.33±0.37(2)	2.25(1)
4	1.07	2.42±0.17(5)	3.11±0.6(5)
5.5	1.18	3.03±0.04(2)	3.24(1)

Table 1 Measurements of indoor Deposition Velocities in four houses. The first two columns show size and geometric standard deviation, GSD, of the test aerosol. The last two columns gives the average deposition to all surfaces measured in three different test houses. The numbers in the parentheses give the number of tests for each condition.

A data set was selected where the correlation coefficients were better than 0.95 for the tracer aerosol decay curves. Twelve results from unfurnished houses and fifteen results from furnished houses were chosen. Experiments where small mixing fans were operated during the test have been included in these data sets. A power regression and a linear regression have been made for the data, expressing the deposition velocity as a function of the particle size. Average deposition velocity was chosen rather than the deposition constant in order to take the different surface to volume ratios of the test rooms into account. The correlation coefficient is given in the parentheses:

$$v_d = 1.23(d_p)^{0.65} \; (r = 0.96)$$

where v_d is the average deposition velocity to all surfaces and d_p is the particle diameter. In both the unfurnished and the furnished rooms the power regression had the best correlation coefficient, i.e. 0.96 compared to 0.90 and 0.95. They found that the deposition velocity increased linearly with the particle size in the particle size range investigated. Since the formulae presented in equations (2.3) to (2.5) are purely empirical, it must be emphasised that they are not valid outside the particle size range investigated, i.e. 0.5 to 5.5 μm.

Present models of the effect of sheltering during releases of radioactive materials to the atmosphere consist of a single factor giving a common dose reduction factor, DRF, for all nuclides. A value of 0.5 is currently used in probabilistic accident consequence assessment codes, Brown (1989) (except noble gases for which DRF = 1.0, i.e. no reduction in inhalation due to indoor residence). In order to provide a more realistic model that takes into account properties of the released material the empirical formula for indoor deposition is used together with equation (2.2) to calculate dose reduction factors, DRFs. equation (2.6) shows the derived formula:

$$DRF = \frac{\lambda_r(U_{wind}, \Delta T)}{S/V * 1.23(d_p)^{0.65} + \lambda_r(U_{wind}, \Delta T)}$$

where the deposition constant, λ_d, is found by multiplying the average deposition velocity with an average S/V ratio for the buildings in question. In the review by Engelmann (1992) surface to volume ratios were summarised for a number of different buildings: 1.74 m^{-1} for apartment buildings/houses, 1.3 m^{-1} for office buildings and 0.66 m^{-1} for industrial buildings. These values do not include contributions from furniture and equipment in the room. The average S/V of the test rooms was 1.69 m^{-1} and a value of 1.7 m^{-1} has been used in the model calculations shown in Figure 1.

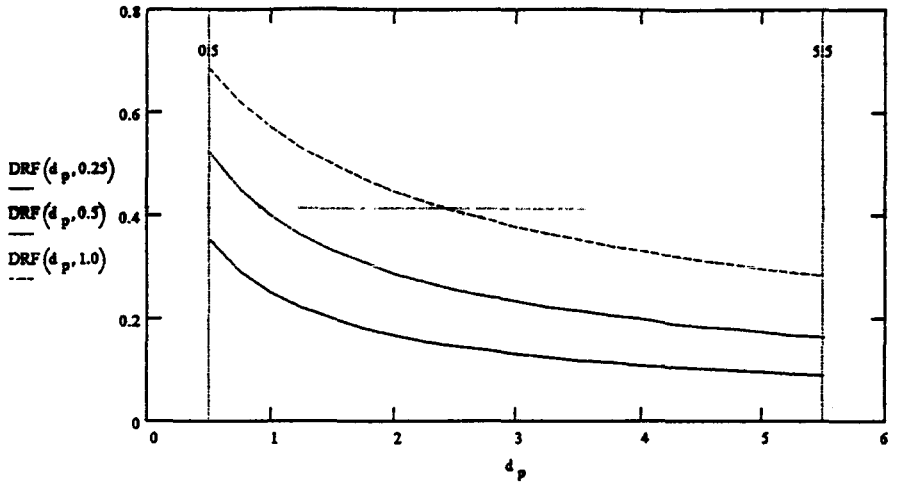

Figure 1 DRF's as function of particle size and air-exchange rate. The number after 'd_p' in the left margin of the figure is the air-exchange rate for that line style. A surface to volume ratio of 1.7 m^{-1} have been used corresponding to a medium sized living room. The vertical lines indicate the valid particle size range, i.e. 0.5 to 5.5 mm.

The air-exchange rate can be expressed as a function of the weather conditions, outdoor temperature and wind speed. The temperature difference over the building envelope can be expressed as the difference between the outdoor temperature and the indoor temperature, typically about 21 °C. The air-exchange rate was determined as a function of these parameters for a typical Danish houses by Kvisgaard et al. (1988). Engelmann (1992) quoted several references and used an equation similar to equation (2.3) for calculation of air-exchange rates for houses in the USA from data on wind speeds and temperature difference. In the Figure the DRF is plotted as a function of particle size for three different values of the air-exchange rate. An assumed surface to volume ratio of 1.7 m^{-1} was used in these calculations. It can be seen that the DRF decreases significantly with particle size. The three values for the air-exchange rate used (0.25, 0.5 and 1.0 h^{-1}) represents a low, medium and high value and the DRF varies between 0.1 and 0.7 or factor of seven depending on the particle size for these air-exchange rates. These relatively large variations in DRF clearly justifies a more detailed model for the DRF.

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CAIRS-ALGADE PERSONAL ALPHA DOSIMETER

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INTRODUCTION

The CAIRS-ALGADE Personal Alpha Dosimeter (PAD) is a portable, lightweight personal monitoring system designed to measure individual exposures to radon and thoron progeny and Long-Lived Radioactive Dust (LLRD).

The CAIRS-ALGADE Personal Alpha Dosimeter consists of a CAIRS track etch detectors (dosimeter head) mounted in an ALGADE individual air sampling system designed to be worn on the belt of the individual to be monitored.

The PAD is worn by the individual for a period of one month. At the end of the month, the CAIRS dosimeter head is removed from the PAD and returned to the CAIRS National Laboratories in Saskatoon, Saskatchewan to be processed. Exposure results are supplied to the respective clients on a monthly basis. We will briefly discuss the CAIRS-ALGADE Personal Alpha Dosimeter, its operation and its capabilities for measuring radon and thoron progeny and LLRD.

ALGADE INDIVIDUAL SAMPLER AND CHARGER

The ALGADE individual sampler is an air pumping system designed to draw air through the CAIRS track etch detector (dosimeter head). A battery operated, centrifuge pump is enclosed in a durable polycarbonate box designed to be worn on the belt of the individual who is to be monitored.

The individual sampler is powered by a small rechargeable battery which, when fully charged, will operate properly for periods in excess of 10 hours. The sampler requires charging times of approximately 14 hours. The ALGADE individual sampler is designed to generate a nominal air flow rate at 4 l/h.

The following is a list of the technical specifications of the ALGADE Individual Sampler.

- A moulded polycarbonate box which houses the system;
- A centrifuge pump driven by a dc motor;
- An electronic module embodying:
  - a nickel-cadmium rechargeable battery, 1.2 V; 1.2 Ah;
  - a constant current charger;
  - a switch controlling the motor operation;
  - a light emitting diode illuminated when the battery is charging;
  - a magnet controlling the start-up of the charger;
- Operating time: in excess of 10 hours;
- Battery charging time: 14 hours;

  Dimensions: 94 x 79 x 63 cm;
- Weight: 230 g (excluding measuring head);
- Relative humidity: 0 to 100%
- Operating temperature: -10°C to 45 °C.

The ALGADE individual samplers are recharged using an ALGADE charging unit. The individual charging unit consists of a polycarbonate box forming a cell into which the individual sampler is placed. The charger includes an electronic card which generates a high frequency alternating current which feeds a coil. This coil supplies by induction the energy necessary to recharge the battery in the individual sampler. It also includes a magnetically operated switch which gates the power

supply to the electronic card when the sampler is placed in the cell and controls an hour-meter which makes it possible to measure the operating time of the sampler.

CAIRS DOSIMETER HEAD

The CAIRS Dosimeter Head is designed and manufactured in Canada. The dosimeter head is an adaptation of an integrated measuring head originally developed by the Atomic Energy Commission of France (CEA). The dosimeter head is designed to detect the presence of radon and thoron progeny and Long-Lived Radioactive Dust (LLRD).

The CAIRS measuring head is in essence, an alpha particle spectrometer capable of detecting separately, without electronics, the 5.99 MeV and the 7.69 MeV alpha particles from radon progeny (Po-218 (RaA) and Po-214 (RaC')) and the 8.78 MeV alpha particle from thoron progeny (Po-212 (ThC')). The alpha particles are detected by the damage they create on a cellulose nitrate film.

Spectroscopic separation of the alpha particles is achieved by using a three-channel collima-tor. Each channel contains an energy-absorbing piece of mylar whose thickness is chosen specifically for the alpha particle it is designed to identify. The design is such that the three alpha particles of interest hit the cellulose nitrate film when their dE/dx (energy loss) is at a maximum (the "Bragg peak" in the dE/dx versus E curve). They produce easily-identifiable holes in the film after etching.

The following is a list of the technical characteristics of the CAIRS dosimeter head:

- High density polyethylene end cap, barrel, collimator and barrel holder;
- Filter used: 25 mm diameter, 1.2 μm pore size;
- Film used: Kodak cellulose nitrate LR-115 Type II film;
- Absorbers used: Mylar absorbers;
- Diameter: 37 mm;
- Height: 43 mm;
- Weight: 21 g.

PAD OPERATION

In the operation of the CAIRS-ALGADE Personal Alpha Dosimeter, the ALGADE sampler draws air through the CAIRS dosimeter head at a nominal air flow rate of 4 l/h. Any attached radon and thoron progeny and LLRD in the air will become trapped on the filter inside the measuring head.

As the radon and thoron progeny and LLRD decay, alpha radiation is emitted. Some of the alpha radiation given off from the filter will travel up the three-channel collimator and pass through the absorbers attached to the collimator. Any alpha particles making it through the absorbers will strike the detector film located on top of the collimator. The detector film (LR-115 Type II) used in the dosimeter head is sensitive to alpha radiation, provided that the alpha particles have an energy of approximately 2.7 MeV.

To obtain the desired energy discrimination between the different alpha particles, each collimator is fitted with a mylar absorber of a specific thickness. The different thicknesses have been chosen so that the alpha particles of interest emerge from the mylar absorber with an approximate energy of 2.7 MeV. Other alpha particles will either be stopped by the absorber or pass through with an energy greater than the optimal 2.7 MeV required to leave clear uniform tracks on the film. All of the energy discriminated alpha particles emitted by the radon and thoron progeny are registered as tiny lines of damaged molecules on the film.

The tracks on the film are then enlarged by etching the films for 90 minutes in 2.5 normal sodium hydroxide (NaOH) solution at a temperature of 60 °C. This produces tiny holes or tracks in the opaque film which are approximately 5 μm in diameter. The enlarged tracks can then be counted using a microscope by placing a light source below the detector film. The LLRD deposited on the dosimeter head filter is counted together with a paired blank separately using a scintillation detector and an automatic sample changer system.

LONG-LIVED RADIOACTIVE DUST DETECTION

The CAIRS-ALGADE Personal Alpha Dosimeter also measures the presence of Long-Lived Radioactive Dust (LLRD). Any traces of LLRD present on the filter inside the CAIRS dosimeter head will remain there for extended periods and can be detected using conventional alpha radiation detectors.

Following a minimum waiting period to allow all radon and thoron December 8, 1995 progeny to decay, the gross alpha activity on the dosimeter head filter is measured using a Zinc Sulphide (ZnS) scintillation detector connected to a Ludlum Scalar/Ratemeter and a Nuclear Chicago sample changer. Each measurement of gross alpha activity on the dosimeter head filters is accompanied by a paired blank measurement to determine the corresponding number of background counts measured by the ZnS detector.

CAIRS PERSONAL ALPHA DOSIMETRY PROGRAM

The CAIRS-ALGADE Personal Alpha Dosimeter is worn daily by the individuals to be monitored. At the end of each working day, the PAD is placed into a charging unit overnight to recharge the ALGADE sampler battery. Radiation safety staff at the client work site are responsible for taking some of the necessary measurements required to determine the average air flow rates through the PADs.

At the end of each month, the CAIRS exposed dosimeter heads are removed from the ALGADE individual samplers and replaced with new dosimeter heads. The exposed dosimeter heads, together with PAD assignment and air flow rate information, are then returned to the CAIRS National Laboratories for processing. Monthly dosimetry reports are generated by CAIRS and distributed to the client.

CAIRS DOSIMETER HEAD TESTS

The CAIRS Dosimeter Heads are regularly tested in the CAIRS Calibration Chamber to ensure proper performance and accuracy. The tests involved the exposure of several dosimeter heads to radon progeny in the CAIRS Calibration Chamber. The dosimeter head results were compared to the CAIRS Reference Method (alphaNuclear alphaSMART 771 System) and manual Kusnetz Method samples.

The average percent difference between the CAIRS dosimeter head results in the CAIRS Reference Method results was 9%. The average percent difference between the CAIRS dosimeter head results and the Kusmetz Method results was 6%.

CAIRS also regularly participates in radon proficiency programs sponsored by the U.S. Environmental Protection Agency and the U.S. Department of Energy Environmental Measurements Laboratory.

BIOLOGICAL DOSIMETRY APPLIED TO TREATMENT WITH ^{131}RADIO-IODINE IN THYROID CANCERS

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This study had 2 objectives :
Firstly to compare the number of unstable chromosomal anomalies (dicentrics, rings and fragments) obtained by the method of conventional cytogenetics with the number of translocations revealed by in situ hybridization (FISH) and secondly to estimate the mean whole body dose after treatment with 3.7 GBq (100 mCi) of ^{131}I.

MATERIALS AND METHODS

Dose effect curves in vitro were generated by the method of counting unstable anomalies and by the FISH method, using peripheral lymphocytes of healthy donors. Both curves were fitted by the linear-quadratic model. Conventional Cytogenetics was carried out on slides containing metaphase spreads stained by Giemsa. Only complete metaphases (46 centromeres) were scored for dicentrics, rings and fragments under a light microscope (1).

Hybridization in situ was carried out using a modified procedure of Pinkel et al (2). The technique was applied to chromosome 4 by means of a specific probe. Chromosome 4 was chosen because of its large size which facilitates the scoring of translocations. The FISH slides were read under a fluorescent microscope using blue light emission and filters for green and red fluorescence.
The pair of chromosome 4 had been labelled by the green fluorescence of Fluorescein isothiocyanate (FITC) and the other chromosomes of the genome counterstained with propidium iodide.
Scoring of dicentrics and translocations was done on a patient blood sample 24 hours before and then 4 days after the administration of ^{131}I.

The number of dicentrics was directly compared to the standard curve.
Since chromosome 4 represents 6,23 % of the total genome, the number of translocations obtained by FISH was extended to cover the whole genome by the following formula (3).

$$F_h = 2\, f_s\, (1 - f_s)\, F_b$$

were F_h is the translocation fraction found by hybridization, f_s the conterstained fraction of the genome and F_b the transloction fraction observed by G-banding which is set equal to 1.

PATIENTS

Patients had been treated with 3.7 GBq (100 mCi) of ^{131}I. The mean total body dose was 0.54 Gy (95 % CI : 0.45 - 0.61 Gy) by chromosome 4 painting. A close relationship was found between total body retention of ^{131}I at day 4 after ^{131}I treatment and the estimated dose by conventional cytogenetics and chromosome 4 painting. In contrast, no relationship was found between the uptake of ^{131}I in thyroid remnants and distant metastases and the dose estimated by biological dosimetry.

RESULTS

The ratio of translocations to dicentrics induced by ionising radiation is supposed to be 1, because a dicentric chromosome as well as a chromosomal translocation is the result of 2 successive events. However, we found that induced translocations were 3-8.4 times more frequent than dicentrics for each dose of irradiation ; this agrees with earlier findings (3).
When comparing the number of chromosomal aberrations found by FISH or conventional cytogenetics with the dose effect curve obtained by in vitro, exposure of normal peripheral lymphocytes, the total body dose was

estimated to be significantly higher than the estimation based on MIRD (4). Some aberrations (dicentrics and translocations) were found before ^{131}I treatment and no known previous exposure to irradiation may explain this.

DISCUSSION

The estimated mean total body dose is 2 to 4 times higher than that based on MIRD calculations (0.13 Gy) (4). In fact, MIRD calculations were derived from individuals with normal thyroid function and normal metabolic activity. Thyroid cancer patients are hypothyroid at the time of ^{131}I administration. The hypothyroid status decreases the renal clearance of radioiodine and thus increases the whole body dose which can explain the discrepancy between the MIRD estimation and the values found by biological dosimetry.

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RADIATION INDUCED CHROMOSOME ABERRATIONS AND INTERPHASE DNA GEOMETRY

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SUMMARY

Ionizing radiation induces DNA double strand breaks (DSBs) and their interaction and illegitimate recombination produces chromosome aberrations. Stable chromosome aberrations comprise inter-chromosomal events (translocations) and intra-chromosomal events (inversions).

Assuming DSBs induction and interaction is completely random and neglecting proximity effects, the expected ratio of translocations to inversions is F=86, based on chromosome arm lengths.

We analyzed the number of translocations and inversions using G-banding, in 16 lymphocyte cultures from blood samples acutely irradiated with γ rays (dose range: 0.5Gy - 3Gy). Our results give F=13.5, significantly smaller than F=86. Literature data show similar small F values but strongly spread.

The excess of inversions could be explained by a "proximity effect", it means that more proximate DSBs have an extra probability of interaction. Therefore, it is possible to postulate a special chromosome arrangement during irradiation and the subsequent interval.

We propose a model where individual chromosomes show spherical confinement with some degree of overlapping and DSBs induction proportional to cross section. We assume a DSBs interaction probability function with cut-off length = 1μm. According to our results the confinement volume is $\approx$ 6.4% of the nuclear volume. Nevertheless, we propose that large spread in F data could be due to temporal variation in overlapping and spatial chromosome confinement.

INTRODUCTION

Ionizing radiation damages DNA and produces chromosome aberrations, generally recognized in metaphase using conventional staining, G banding or fluorescence in situ hibridization (Fish).(1-3)

The primary mechanism for the production of exchange-type chromosomal aberrations after irradiation of resting lymphocytes is the pairwise interaction of two double-strand breaks (DSBs) an their illegitimate recombination.(4)

The simplest aberrations to score in metaphase spreads are dicentrics, which are interchromosomal events, and centric rings, which are intrachromosomal events. These are unstable aberrations and they do not survive after cell division. The stable counterparts are reciprocal translocations and pericentric inversions, respectively, and they are used particularly in retrospective dosimetry due to the fact that they remain long after overexposure to ionizing radiation. The correspondence between unstable and stable aberration production has been generally confirmed.(5)

The observed ratio of interchromosomal interactions (dicentrics or reciprocal translocations) to intrachromosomal events (centric rings or pericentric inversions) indicates that it is smaller than predicted assuming complete randomness, it is there are more centric rings or pericentric inversions than expected.

The observed excess of intrachromosomal events may be due to a proximity effect (6): two DSBs induced in a short distance have an extra probability of pairwise interaction and recombination. Furthermore, there is evidence of chromosome localization during interphase (7).

Assuming that the excess of intrachromosomal events is due to the above mentioned facts, we propose a lymphocyte DNA spatial arrangement during interphase, where individual chromosomes are confined to spherical domains which interact among them under several constrains and we estimated, according to our experimental data, the size of the chromosome domain.

MATERIAL AND METHODS

The number of radiation induced translocations and pericentric inversions were obtained from 16 lymphocyte cultures of both sex donors, which are part of our calibration curve data set for low LET radiation using stable chromosome aberrations. Blood samples were acutely irradiated with γ rays from a Co^{60} source (Picker C4M60) with doses ranging from 0.5Gy to 3Gy (mean dose-rate: 0.5Gy/min). Heparinized whole blood from each donor was irradiated and cultured using micromethod for 48 h in 10 ml of RPMI 1640 medium containing 20% fetal calf serum and PHA (0.15 mg/ml). Colchicine was added to the cultures after 47 h of incubation at 37°C and air dried metaphase spreads were prepared.

Translocations and inversions were identified by G banding, according to Seabright technique, modified.(8) All metaphases were photographed and analyzed using enlarged prints. We observed 325 reciprocal translocations and 24 pericentric inversions in 1040 banded metaphases. The ratio of interchromosomal to intrachromosomal events is F = 13.5

MODEL

Let F represent the ratio of translocations to pericentric inversions. Assuming that DSBs are produced at random on a chromosome arm, proportional to its length in base pairs following a Poisson distribution and assuming, in addition, that pairwise DSBs interactions and recombinations are completely random and neglecting proximity effects, it is possible to estimate a theoretical value of F. (9-11)

$$F = \frac{\sum_{i=1}^{46}(L_i + S_i)(T - S_i - L_i)}{2\sum_{i=1}^{46} L_i S_i} = 86$$

where: L_i = length in megabases of *i*th chromosome long arm.

S_i = length in megabases of *i*th chromosome short arm.

T_i = length in megabases of average diploid set weighted for both sexes.

representing the numerator the average number of reciprocal translocations and the denominator the average number of pericentric inversions.

The convention for translocation counting assumes complete reciprocity. It is important to point out that when it is said that the theoretical F value neglects proximity effects, it means that the individual chromosomes are considered freely distributed in the nuclear volume filling fractions of this volume just according to their lengths.

The obtained value F = 86 is significantly higher than our experimental result F = 13.5 an those obtained by other authors considering unstable and stable chromosome aberrations and using radiation of different linear energy transference (12).

A plausible explanation for the observed small F values is that DSBs produced on the two arms of the same chromosome are in fact and on average more proximal. It implies additional chromosome confinement in domains during interphase, when DSBs are induced and recombined. The proposed spherical domain volumes are smaller than the nuclear volume. We accept several experimental an theoretical considerations (13) which suggest that the probability g(r) that two induced DSBs separated by a distance *r* will interact and produce an exchange event is the cutoff form:

g = constant if $r \leq h$ and g = 0 if $r > h$

Here h is a maximum interaction distance and its value is ≈ 1μm (14). It is interesting to consider that even if two DSBs are produced at a distance more than h apart and are possible to move to get in contact, they may be restituted before interacting and, anyway, it is possible to consider $g \approx 0$ for $r \geq h$.

Then, assuming chromosome spherical domains, two DSBs produced in both arms of the same chromosome have an extra chance of interaction than DSBs produced in different chromosomes because they will more probably lie within 1μm each other.

Now, we propose to analyze DSBs interactions assuming a spherical domain maximum packing model with 12 nearest neighbors in contact, corrected by nuclear surface effect. Moreover, we postulate that the DSBs production is proportional to the geometric cross section , defined as (chromosome length)$^{2/3}$

Based on the above described assumptions the theoretical relationship between reciprocal translocations and pericentric inversions results:

$$F = \frac{\sum_{i=1}^{46} (L_i + S_i)^{2/3} \alpha^{2/3} \beta \delta}{2 \sum_{i=1}^{46} (L_i + S_i)^{4/3} \frac{L_i S_i}{(L_i + S_i)^2}}$$

where: α = chromosome average length in megabases (140.3)

β = average number of nearest neighbor corrected by surface effect (12 * 0.58)

δ = correction factor due to maximum interaction distance (1μm)

Assuming a cell nucleus diameter d = 8μm and a maximum interaction distance = 1μm we calculated that spherical chromosome domains in contact have approximately an average diameter = 2μm. Under these conditions results F = 10.7.

If we consider higher domain diameter, they get some degree of overlapping and consequently the probability of translocations and F values increase. The total overlapping condition corresponds to a domain diameter equal to the nuclear diameter = 8μm and results F = 86.

According to our experimental results, F = 13.5 indicates that domains overlap giving a domain volume = 6.4% of the nuclear volume, with a diameter $\approx$ 3.2μm.

Literature data show strongly spread F values, even considering experiments using the same radiation quality. The present model allows to represent such dispersion just varying the domain diameter but keeping fixed the maximum interaction distance (δ). We proposed that in vivo chromosome domain dimension could vary temporarily during interphase, offering different scenarios to ionizing radiation.

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CORRELATION OF THE LET-DEPENDENT TLD RESPONSE WITH BIOLOGICAL EFFECTS IN VARIOUS RADIATION FIELDS

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ABSTRACT

Investigations were made to correlate biological and physical parameters in order to develope a biologically calibrated dosemeter system. Irradations of cell cultures and TLDs were performed. For this purpose a heatable irradition chamber was developed wherein the irradiation of TLDs together with cell cultures is possible. The heating (37°C) is necessary for the cell cultures during long time irradiations.
For measurements of the TL-response in correlation with LET-dependent biological effects the peak height ratio method developed for determination of the average LET in mixed fields was used. The genotoxic endpoints investigated are micronuclei, sister chromatid exchange, dicentric chromosomes and single strand breaks. These parameters will be correlated with the physical parameter of the peak height ratio. Irradiations were performed in Co-60- and thermal neutron radiation fields.

INTRODUCTION

One of the most important problems in radiation protection concerns determination of the biological effectiveness of ionising radiation. The present radiation protection model is based on the correlation of the physical parameter LET and the Q-factor which correlates with the biological effectiveness. From the physical point of view the absorbed dose as well as the LET and therefore the equivalent dose can be obtained with high accuarcy (1). On the other hand a lot of biological parameters as individual sensitivity of cells of different propands or stress cannot be excluded if the damage caused by ionising radiation is evaluated. To yield more detailed information of radiation action in tissue interest in measurements of genotoxic effects in cells has increased in the last decade. Several methods have been introduced for biological dosimetry (2, 3). For radiation protection purpose a biological calibrated dosemeter system would be of particular interest because of improved information of radiation action. The developement of such a dosemeter system is based on thermoluminescent dosimetry. Combined irradiation of cell cultures and TLDs performed in different radiation fields allows the correlation of different biological and physical parameters. Using this calibration, measurements with TLDs also in mixed radiation fields will allow predictions of genotoxic damage in tissue.

IRRADIATION CHAMBER

In order to perform these combined irradiations of cell cultures and TLDs several requirements have to be met. Since cell cultures need a constant temperature of 37°C for optimal growth a heatable waterfilled irradiation chamber was developed (fig. 1). It consists of a zylindrical waterfilled box containing a heating wire, a thermocouple and a slide-in-unit for the cell-cultures. The thermocouple is positioned in the middle of the box, the heating wire as far as possible from the cells to reduce inhomogenity of the radiation field through scattering. The TLDs are attached at the backside of the cell culture bottle. An external controller allows a temperature regulation with an accuracy of ± 1°C.

DETERMINATION OF PHYSICAL PARAMETERS

Measurements of the absorbed dose and the LET are performed using LiF:Mg,Ti TLDs. Analysing the peak 5 maximum the absorbed dose can be obtained with an accuracy of about 5%. For further analysis of the TLD glowcurves the High Temperature Ratio (HTR)-Method is used (1). The HTR correlates with the distribution of the energy deposition. For evaluation of the LET the HTR of the glowcurve is compared with the HTR of a Co-60 irradiation normalized on peak 5. Using a LET-calibration of the high temperature ratio the average LET of unknown mixed radiation fields and the quality factor can be determined. Figure 2 shows the LET-dependence of the HTR used for the LET-calibration.
Irradiations with a Co-60 source and with thermal neutrons were performed. Since the thermal neutron component is superposed by a gamma component the pair method (4) is used to seperate the gamma and the neutron fraction of the mixed field. For this purpose TLD-600 and TLD-700 are irradiated.

BIOLOGICAL PARAMTERS USED FOR INVESTIGATIONS

Different genotoxic endpoints show individual sensitivities against irradiation with ionising radiation. Besides micronuclei (MN) and dicentric chromosoms (DIC) which are rather sensitive for ionising irradiation, sister chormatid exchange (SCE) and single strand breaks (SSB) have been investigated. Tests have confirmed that

SCE are not sensitive for Co-60 irradiation (5). Analysing the SSB also no significant increase was determined after irradiation with Co-60. Therefore only micronuclei and dicentric chromosomes were used for gentoxic investigations. The measurements were performed with diploid human fibroblasts and the chinese hamster cell line V79.

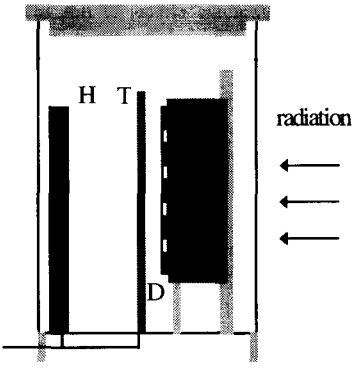

Figure 1: Scheme of the irradiation chamber for cells and TLDs; H ... heating wire, T ... thermocouple, D ... dosemeters, C... cell culture

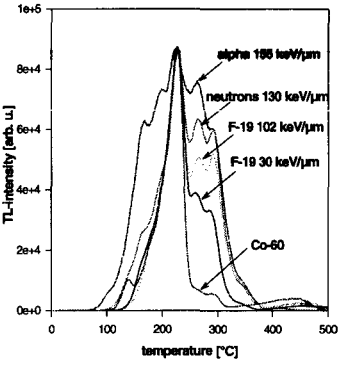

figure 2: TL-glowcurves showing the LET dependent high temperature region, glowcurves are normalized on peak 5 maximum.

RESULTS

Co-60 irradiations were performed with an irradiation source of the University Clinic of Radiation Therapy (11. 04. 86: 9821 Ci). A dose characteristic was measured in order to obtain the lower detection limit and the dose dependence of the genotoxic effects micronuclei and dicentric chromosomes. Table 1 gives the determined micronuclei induction and dicentric chromosomes and the high temperature ratios after Co-60 irradiations of fibroblasts. An increase of both gentoxic endpoints was obtained. Figure 3 shows the dose dependence of micronuclei for both irradiation series. It can be seen that the lower detection limit for micronuclei is about 0.5 Gy. Furthermore an increased MN-rate at 100 mGy compared with 200 and 400 mGy was determined for one irradiation series. The more detailed investigation of these results, especially of the dicentric chromosoms of series 2, is still in progress. The dose characteristic of fibroblasts will be compared with results obtained with V79.

Although the LET of Co-60 is 1 after ICRP 60 (6) it can be seen that the HTR increases with higher Co-60 doses. Figure 4 shows the dependence of the HTR on the absorbed dose. This is due to the supralinearity of the dose characteristic of LiF. Each peak of the glowcurve shows a different supralinear increase (7). Comparing peak 5 and the high temperature peaks 6 and 7 the ratio varies therfore not only in dependence on the LET but also with absorbed dose. Furthermore comparing figure 3 and 4 it can be seen that the dependence of the gentoxic parameter and the HTR on the dose is rather similar. This situation cannot be explained by the model of LET but efforts are made to describe this situation on the basis of microdosimetry.

absorbed dose [mGy]	MN/500	DIC	HTR	absorbed dose [mGy]	MN/500	HTR
0	2.5	1	-	0	7.1	-
94 ± 4	4.7	1	1.26 ± 0.06	87 ± 5	7.9	1.30 ± 0.03
196 ± 2	2.2	0	1.40 ± 0.06	172 ± 11	9.4	1.52 ± 0.09
381 ± 8	2.7	2	1.42 ± 0.13	349 ± 14	11.6	1.53 ± 0.16
611 ± 3	6.4	2	1.78 ± 0.24	550 ± 5	31.7	1.86 ± 0.21
819 ± 51	7.8	1	1.75 ± 0.07	696 ± 29	15.7	1.79 ± 0.08
970 ± 20	6.4	3.7	1.77 ± 0.17	935 ± 34	25.0	1.80 ± 0.13
2008 ± 187	8.7	7.9	2.36 ± 0.16	1874 ± 209	28.7	2.43 ± 0.07

table 1: Results of the dose response of micronuclei and dicentric chromosomes and the HTR after Co-60 irradiation. Evaluation of DIC for series 2 was not finished. Values given for absorbed dose as well as the HTR were determined with TLD-700. For each dose measurement three dosemeters were used.

Irradiations with thermal neutrons were performed at the TRIGA Mark II reactor of the Atomic Institute of the Austrian Universities. The thermal neutron flux is $1.4 \cdot 10^6$ n/cm²s, the gamma doserate is about 60 mGy/h. An increase of the micronuclei induction rate for an irradiation with about 50 mSv neutron dose and 500 mGy gamma was obtained with large uncertainty. Detailed investigations on the genotoxic effects of this mixed radiation field are under progress.

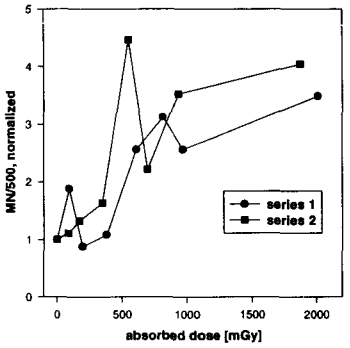

figure 3: Dependence of the MN induction with absorbed dose after Co-60 irradiation. The values are normalized on the controlling group (0 Gy). Lines are only to guide the eye.

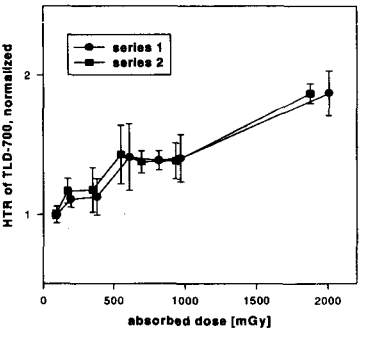

figure 4: Dependence of the HTR measured with TLD-700 after Co-60 irradiation. Results are normalized on the absorbed dose of 100 mGy. Lines are only to guide the eye.

CONCLUSION

For a biological calibration of a TLD-system combined irradiations of TLDs and cell cultures have to be performed using radiation sources with different biological effectiveness. For this purpose an irradiation chamber was developed which meets the proposed requirements. The dependence of the biological and physical parameters on the absorbed dose of a Co-60 irradiation was measured. An increase of the gentoxic parameters micronuclei and dicentric chromosoms as well as an increased HTR was obtained. Comparing the increase of the biological and physical parameters it can be seen that the increase with dose is rather similar. These effects cannot be explained using radiation protection models as LET. Therefore microdosimetric investigations are made to explain these results.

Since the gentoxic investigations show a great uncertainty the results have to be considered. Furthermore the increased MN-rate at 100 mGy for one of the irradiations will be studied in more detail. The irradiation of fibroblasts and V79 with a mixed thermal neutron - gamma radiation field showed an increase of the micronuclei induction rate with a large uncertainty which has also to be considered. Furthermore irradiations with protons of different energy are planned.

ACKNOWLEDGEMENT

We thank Doz. Binder and Dr. Schmidt of the University Clinic of Radiation Therapy, AKH, for their support concerning the Co-60 irradiations.

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PERSISTENCE OF UNSTABLE CHROMOSOME ABERRATIONS IN MEDICAL STAFF OCCUPATIONALLY EXPOSED TO IONISING RADIATION

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INTRODUCTION

Chromosome aberrations in cultured human lymphocytes may serve as biological indicators of occupational radiation exposure. From earlier investigation during the history of radiation protection it is well known that the frequency of dicentrics, rings and acentric fragments remained stable for some time after irradiation (2). *In vivo* analyses of chromosome aberrations in peripheral blood lymphocytes of persons occupationally exposed to ionising radiation have shown various aberration frequencies. With this study we want to highlight the persistence of chromosome aberrations (CA) expressed as chromosome breakage in human peripheral blood lymphocytes in medical staff occupationally exposed to ionising radiation.

MATERIAL AND METHODS

The group of 46 subjects, all medical staff occupationally exposed to ionising radiation was divided into five groups. Analyses for chromosomal aberrations were repeated after 3-4 months (8 subjects), 6-7 months (12 subjects), 8-9 months (6 subjects), 11-12 months (10 subjects) and after 24 months or less (10 subjects). Lymphocytes from whole blood culture were examined for the presence of chromosome aberrations. Whole blood cultures were obtained according to routine protocol using F-10 culture medium (Gibco) containing 20 % foetal calf serum, phytohaemagglutinin and antibiotics. The cultures were incubated at 37^0 C for 48 hours. Colchicine was added during the final three hours of cultivation. Cells were harvested by centrifugation after three hours reincubation, then swelled in 0.075 M Kcl and fixed in 3:1, methanol-acetic acid fixative. Slides were stained by Giemsa method for chromosome aberrations.

RESULTS AND DISCUSSION

The examinees were divided in five groups (Tables 1-5). In the first group, in 25% of examinees obtained after 3-4 months remained unchanged, in 37.5% a decrease, and in 7.5% a increase of unstable aberrations were observed. In the second group, in 25% of examinees an increase and a decrease were observed and in 50% of examinees the finding remained unchanged. In 66.7% of subjects who were analysed 8-9 months after exposure a decrease of unstable chromosome aberrations was observed. In a fourth group, in 50% of examinees a decrease was observed, whereas in 40% the finding remained unchanged. In 60% of subjects investigated after 24 months or less (the fifth group) the finding remained unchanged. Regardless of the results, all examined subjects, except five (*), continued to work. As indicated by personnel monitor film badge, two subjects received the exposure dose that exceeded currently recommanded dose limits (**). The results point out the importance of chromosome aberration analyses for the detection of radiation induced damage, as well as the significance of implementation of protective measures. The individual differences in radiosensitivity as well as the frequency of cytogenetic aberrations depend on physiological conditions of each subject and on variability in DNA repair and misrepair processes (3). As time dependent changes in dicentrics may be subject to individual variations, it may be difficult to extrapolate from one case to another (1).The present results do not allow definite conclusions on the effects of ionising radiation during the occupational exposure, but are enough intriguing to motivate further investigations.

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Table 1.

	FIRST SAMPLING						3 - 4 MONTHS LATER					
Subject	Chroma-tid break	Chromso-me break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)	Chroma-tid break	Chromso-me break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)
1	2		2	1			5		1	1		
2	2	3	2	2			5	1	5	1		
3	5	1	5	1			1					
4*	2	1	6	2			1		1	1		
5	2	1	2	1			1	3	4			1
6	1		3	1			1	1				
7	4		1	1			1	1	1	1		
8*	3	1	2	1			1		4	1	2	
9		2	5	1			1		3	1		

Table 2.

	FIRST SAMPLING						6 - 7 MONTHS LATER					
Subject	Chroma-tid break	Chromso-me break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)	Chroma-tid break	Chromso-me break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)
10	1	1				1	4	2	4	1		
11	2	7	1	1				2	3	1		
12		2	3	1			1	5	5			
13	1	5	5				2	3				
14*	1	1	2		1		1	2	2	2		
15		6	2	1			3	3	3	1		
16*	4	2			1		1	1	1			
17		2	3	1				1	2			
18	3	2	1	1			3	1	2			
19	3	5		1				1	2		2	
20	1	1	3	2			1	2	1			
21*	2	3	1	1			2	2	2		1	

Table 3

	FIRST SAMPLING						8 - 9 MONTHS LATER					
Subject	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)
22	2	2	24	5				18	4	3		
23				1		1	1	6				
24	2	3	4	4			2	1	3			
25	2	4	2	1				2				
26	4	2	4	1				4				1
27	1	2	2	1				6	2	1		

Table 4.

	FIRST SAMPLING						11 - 12 MONTHS LATER					
Subject	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)
28	3	6	1	1			3	9	1	2		
29	2	3	1	1			2	1	1	1		
30**		1	3	1			3	9	2			
31		18	4	3			5	8		3		
32	2	4	2				1	1				
33	1	6					1					
34	1	1		2			2	6	3			
35	1	1	2	1			3	1				
36	2		1	1			1	4	1			
37	1	2	4	1			4	2	2			

Table 5.

	FIRST SAMPLING						≥ 24 MONTHS LATER					
Subject	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)	Chromatid break	Chromsome break	Acentric fragment	Dicentric	Ring	Chromosome interchanges (tetraradial)
38	7	5	3	1				5	3	1		
39	1	3		1			1	4	1			
40		1	1	1			3	2	1	1		
41	3	5		1				1	2		2	
42	1	2			1		2	1	4	4		
43**	1		4	1			1	2	2	1		
44	2	3	4	1			7	5	2			
45	2	4	2				1	1				
46	1		1	1				2	3	1		
47	1		1	1			3	1				

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PAPER TITLE Induction of in-vivo blood chromosomes aberrations by low-level radiation from nuclear fallout

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MAJOR SCIENTIFIC TOPIC NUMBER 2.3 (see page 7)

INDUCTION OF IN-VIVO BLOOD CHROMOSOMES ABBERATIONS BY LOW LEVEL RADIATION FROM NUCLEAR FALLOUT

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In some alpine regions of Austria the surface deposition of the fallout following the nuclear accident in Chernobyal reached levels up to 100 kBq/m² of ^{137}Cs. Parts of these regions can be characterized as upland ecosystems with high transfer-factors resulting in significant contamination of the local food production. Most affected are persons who are working in seasonal agricultural production at elevated sea levels during the summer time. This group still receive high Cs-burdens due to consumption contaminated nutrition and to the elevated gamma background.

The various doses from external gamma radiation and internal radiation (Cs, K, and radon decay products) in the sesonal working places in the mountain regions as well as in the valley residences of selected members of the group affected, have been assessed for each person seperately. These data allow to draw a dose-relationship for the individually in vivo induced chromosome aberrations. Due to preliminary investigations dose elevations of about 60% compared to natural background result in increased aberration rates of blood chromosomes. These findings are in accordance with studies carried out in the city of Salzburg one year after the Chernobyl accident.

PRESENT STATE OF BIOLOGICAL DOSIMETRY IN CUBA.

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ABSTRACT

The most significant research in biological dosimetry by chromosome aberration analysis carried out in Cuba includes: the establishment of dose response relationship for chromosome aberration and micronuclei, the examination of 56 persons involved in minor radiological incidents, the study of 69 children from areas affected by the Chernobyl accident, and 10 persons irradiated in Goiania, as well as the co-ordination and performance of an intercomparison exercise in cytogenetic dosimetry with the participation of laboratories from Argentina, Brazil, Chile and Peru.

Biochemical indicators suggested for the evaluation of persons accidentally exposed to high radiation dose are available.

The possible use of nucleic acid concentration in blood leucocytes, as an early predictor of survival possibility of irradiated persons is now under study in whole body irradiated patients for bone marrow transplantation.

INTRODUCTION

Research in biological dosimetry started in Cuba in 1985 with the foundation of the Center for Hygiene and Radiation Protection. The main objective for biological dosimetry at that time was to introduce in the country the methods for cytogenetic dosimetry. Later, some researches with biochemical indicators of radiation injury were started. This paper presents the most significant results achieved in the past ten years.

CYTOGENETIC DOSIMETRY

Dose response relationship for chromosome aberration and micronuclei were obtained by "in vitro" irradiation of human blood with ^{60}Co in 0.5 - 4 Gy dose range at a dose rate of 1.26 Gy/min for chromosomal aberration and 0.43 Gy/min for micronuclei.

For obtaining metaphases, cultures of 48 hours of 1 ml of total blood in 9 ml RPMI 1640 culture medium supplemented with 10% of fetal calf serum with the presence of phytohemaglutinin were used. For obtaining binucleated lynfocytes equal rates of culture medium and blood, both volumes four times smaller, culture time of 69 hours and cytochalasin B in 6 μg/ml concentration were used.

The dose response curves have been fitted to the linear quadratic model by the Papworth method (1).

The relationship between the frequency of aberrations/ 100 cells (Y) and dose (Gy) is:

for dicentric $Y = 0.0+(2.84\pm0.97)D+(4.68\pm0.38)D^2$

for micronuclei $Y = 1.19\pm0.28+(1.48\pm0.55)D+(3.33\pm0.22)D^2$

Up to the moment, 56 persons involved in minor radiological incidents have been analyzed. Hospital workers are more frequently involved in such incidents. In five cases dicentric frequencies were above normal.

A computer program which includes U-test and dose estimation in total and partial irradiations was developed and is used in Cuba and other Latin American countries for cytogenetic dosimetry.

CYTOGENETIC STUDY OF CHILDREN FROM AREAS AFFECTED BY THE CHERNOBYL ACCIDENT

Children from areas affected by the Chernobyl accident have been receiving medical care in Cuba since 1990. As part of the radiological assessment, chromosome aberrations and micronuclei frequency were established in 28 (19 males) children evacuated from Pripyat, 21 (8 males) living in Kiev and 20 (10 males) from Ovruch. The number of scored metaphases and binucleated lymphocytes was: 11425 and 13500 in Pripyat, 5406 and 13959 in Kiev, and 2779 and 9891 in Ovruch.

Dicentric and micronuclei frequencies per 100 cells were: 0.02±0.01 and 0.56±0.06 in Pripyat, 0.04±0.03 and 0.60±0.06 in Kiev, and 0.03±0.03 and 0.87±0.09 in Ovruch. All individual frequencies were normal.

CYTOGENETIC STUDY IN PERSONS IRRADIATED IN GOIANIA

Translocations (G-banded cells), dicentrics and micronuclei frequencies were established in 1992 in 10 persons (4 males), 8 with 4.40 - 0.15 Gy dose and 2 "intra-utero" exposed during the Goiania accident.

Dicentric frequencies above normal were found in 2 persons (4.4 and 1.1 Gy). For the first person, the dicentric frequency has decreased 100 times, for the second one, 10 times since the accident occurred. All

micronuclei frequencies were normal. Translocations and dicentric frequencies were similar in a 4.4 Gy-exposed individual.

INTERCOMPARISON IN CYTOGENETIC DOSIMETRY

As a part of a regional IAEA collaborative project, laboratories from Argentina, Brazil, Chile, Cuba and Peru participated in an intercomparison in cytogenetic dosimetry. Human lymphocytes were irradiated with ^{60}Co (0, 0.75, 1.5 and 3.0 Gy). Code slides for chromosomal aberration and micronucleus analysis were prepared by the organizing laboratory and sent to the other participants. Dicentric and micronuclei frequency were used for dose estimates.

Eleven of the 15 estimates of dose based on dicentrics and 9 of the 12 based on micronuclei fell within ± 30% of the true dose. When considering the uncertainties of the dose estimates, the true doses fell within the 95% confidence limits of the estimates on 8 of the 15 occasions for dicentrics and 4 of the 12 for micronuclei. In summary, the scoring for all 5 laboratories was similar and it was concluded that they can contribute to the scoring of blood samples for dose estimations in a large scale radiation accident (2).

BIOCHEMICAL INDICATORS

Radiation produces a drop in concentration of nucleic acid in peripheral blood leukocytes (3).

The nucleic acid dynamics in 14 whole body irradiated patients for bone marrow transplantation and 16 breast irradiated women was studied. A peak on the nucleic acid concentration curve obtained was observed 16 - 18 days after whole body irradiation in successful cases. Fig 1 show the nucleic acid dynamics and leukocytes count in one patient irradiated up to dose of 10 Gy (4 Gy + 6 Gy in days -2 and -1 before bone marrow transplantation at day 0). The prospects of using this indicator in the follow-up of the bone marrow recovery is under study.

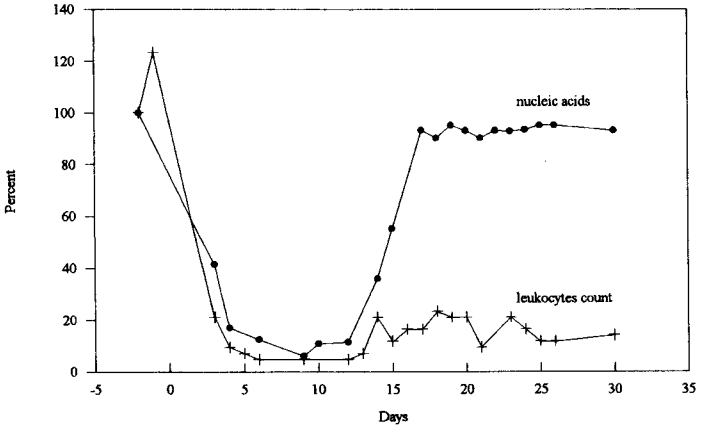

Fig 1. Nucleic acid dynamics and leukocytes count after whole body irradiation

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CELLS WITH MULTIPLE MICRONUCLEI IN CAT SCRATCH DISEASE

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ABSTRACT

In several cytogenetic studies carried out in inhabitants of zones affected by Chernobyl accident cells with multiple aberrations have been detected. The presence of hot particles with beta and alpha activity in polluted zones has contributed to considering them as possible inductor agent. Even though the etiology of these cells has not been clearly established, viral infections are considered its most probable cause.

As a part of evaluation of the radiological impact of the Chernobyl accident, chromosome aberrations and micronuclei in lymphocytes with blockade of the cytokinesis frequencies were established in 14 children of affected areas, with the preliminary diagnosis of oncohematological affection.

The most outstanding cytogenetic finding was the presence of 4 cells that contained 7-11 micronuclei in a case, where 504 lymphocytes were analyzed and whose final diagnosis was cat scratch disease. The viral origin of this illness has been reported. This micronuclei distribution was not found in 6074 cells analyzed in the rest of the members of the group.

INTRODUCTION

The impact on the health of the Chernobyl accident has been evaluated by means of numerous studies. Among them, cytogenetic studies stand out as they use indicators of the highest sensitivity to the action of ionizing radiations. One of the most interesting findings of cytogenetic studies has been the presence of cells with multiple aberrations ("rogue cells") in some of the people studied. The presence of hot particles with beta and alpha activity in contaminated zones has contributed to considering them as a possible inductor agent.

The origin of these cells has not been clearly established, but their presence has been reported in numerous population groups that have not been exposed to chemical agents or ionizing radiations (1).

As a part of the evaluation that is developed in Cuba to children from areas affected by the Chernobyl accident, cytogenetic studies in several groups were carried out. The presence of cells with multiple micronuclei was detected during the study of a group of patient with presumed hematologic diseases, who arrived in the Island in 1990.

MATERIALS AND METHODS

The frequencies of stable chromosome aberrations and micronuclei in 14 patients with preliminary diagnosis of leukemia were established. They arrived in Cuba in 1990 from Ukrainian regions affected by the Chernobyl accident. The characteristics of these patients are shown in Table I.

TABLE I. CHARACTERISTICS OF THE STUDIED PATIENTS

NO.	AGE	SEX	DIAGNOSIS
1	4	M	ALL in remission
2	7	M	ALL in remission
3	5	M	ALL with infiltration
4	6	F	ALL in remission
5	15	F	Bone marrow aplasy
6	8	F	ALL in remission
7	14	M	Cat scratch disease
8	12	M	ALNL
9	3	M	ALL in remision
10	6	M	ALL with infiltration
11	7	F	ALL in remision
12	6	M	-
13	10	F	ALL with infiltration
14	14	M	ALL with infiltration

*ALL-Acute Lymphoblastic Leukemia.
ANLL-Acute Non-Lymphoblastic Leukemia.

For the blood cultures and for obtainig metaphases and binucleated lymphocytes methods standardized in our laboratory were used (2), which in sum are: for obtaining metaphases, cultures of 48 hours of 1mL of total blood in 10 mL of RPMI 1640 culture medium supplemented with 10% of fetal calf serum and with the presence of phytohemaglutinin; and for obtaining binucleated lymphocytes, equal rates of culture medium and blood, but volumes 4 times smaller, culture time of 69 hours and the use of cytochalasin B in concentration of 6 µg/mL. Cases in which were possible to analyze at least 200 metaphases, and 300 binucleated lymphocytes were considered.

RESULTS AND DISCUSSION

Table II and III shown the presence of observed chromosome aberrations and micronuclei respectively.

TABLE II. FREQUENCY AND TYPE OF CHROMOSOME ABERRATION (C.A.)

NO.	NO. OF METAPHASES ANALYSED	FREQUENCY OF C.A./ 100 CELLS		
		ACENTRICS	DICENTRICS	TOTAL
1	500	1.00±0.44	0	1.00±0.44
2	500	1.00±0.44	0	1.00±0.44
3	425	0.23±0.23	0	0.23±0.23
4	409	0.73±0.42	0	0.73±0.42
5	-	-	-	-
6	500	2,60±0.71	0	2.60±0.71
7	500	4.40±0.94	0.20±0.20	4.60±0.94
8	-	-	-	-
9	330	0.30±0.30	0	0.30±0.30
10	-	-	-	-
11	252	0	0	0
12	-	-	-	-
13	500	0	0	0
14	-	-	-	-

TABLE III. FRECUENCY AND DISTRIBUTION OF MICRONUCLEI(MN) PER CELL

NO.	NO. OF LYMPHOCYTES BINUCLEATED ANALYZED	FREQUENCY MN/100 CELLS	DISTRIBUTION OF MN / CEL				
			0	1	2	3	+4
1	356	2.20±0.78	348	8			
2	500	1.40±0.52	493	7			
3	345	1.40±0.63	340	5			
4	500	1.60±0.56	492	8			
5	500	2.80±0.74	489	8	3		
6	373	3.40±0.94	360	13			
7	504	11.10±1.40	484	13	2	1	4*
8	500	1.80±0.59	491	9			
9	500	0.80±0.40	496	4			
10	500	0.60±0.33	497	3			
11	500	1.20±0.49	496	2	2		
12	500	0.20±0.20	499	1			
13	500	1.20±0.49	494	6			
14	500	0.20±0.20	499	1			

* 4 cells with 7-11 micronuclei

The most outstanding cytogenetic finding was the presence of 4 cells that contained 7-11 micronuclei in case No.7. No cogent explanations were found to this phenomenon or to the high frequency of acentrics, until the diagnosis of cat scratch disease for this case was known finally.

The viral origin of this illness has been suggested based in epidemiological test (3), which lead us to believe that the most probable cause of this finding is the viral infection.

The relationship between cells with multiple aberrations "rogue cells" and virus has been suggested by several authors (4-6), and the relationship between chromosomal aberrations and micronuclei is known (7). These are the reasons why,in our case, we did not discard the possibility that cells with multiple aberrations have give origin to cells with multiple micronuclei. No typical "rogue cells" were found in analyzed metaphases.

High frequencies of acentrics in cases 1, 2 and 6; and of micronuclei in cases 1, 5 and 6 were also detected. The possibility of previous treatments with cytostatic to be the cause of these increments could not be discarded since practically all the cases had been treated before arriving in our country and it was impossible to determine the kind and duration of treatment.

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CHROMOSOME ABERRATION OF ACCIDENTALLY EXPOSED PERSONS IN FRANCE : REVIEW OF CASES 1992-95.

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INTRODUCTION

Since more than 20 years, biological expertise concerning dose estimation of suspected overexposure of personnel to ionising radiation has been undertaken by a small number of laboratories in each continent (1). Dose estimation by change of some biological parameters appears a useful compliment to clinical and physical dosimetries together with the knowledge of the health status of the potentially exposured people. It becomes particularly important when physical dosimetry is not present and clinical dosimetry is not consistent (2). Biological estimates of accident overexposure is usually obtained by the scoring of radio-induced unstable structural chromosome aberrations (dicentrics, centric rings and fragments) in peripheral blood lymphocytes (3). A correlation between the dicentrics yield and the physical dose can be obtained by reference with an *in vitro* dose-response relationship, which is related to radiation quality and dose rate (3, 4). Although some new biological indicators such as micronuclei or translocation yields may constitute interesting alternatives, dicentric scoring is considered as the most specific and sensitive biological dosimeter.

During the 1992 - 1995 years period, 33 persons were investigated by the Laboratory of Multiparametric Biological Dosimetry of the Institute of Protection and Nuclear Safety (I.P.S.N.) for a potentially overexposure accident. Although limited, this number of cases was considered large enough for a first statistical overview. This report summarises the related observations and conclusions.

METHODS USED

A blood sample is collected during medical examination of the patient by antecubital venepuncture in tubes containing lithium heparin as anticoagulant (Becton Dickinson).

After arrival in the laboratory, the lymphocytes are cultured using a protocol similar to that described in the IAEA technical report (4). Briefly, several replicates of each blood sample were added to a culture medium (Life Technologies) supplemented by phytohaemaglutinin (PHA, Life Technologies), a mitogenic factor of lymphocytes, heat-inactivated fetal calf serum and antibiotics. Bromodeoxyuridine (BrdU, Sigma), a thymidine analogue, is also included in the blood culture, in order to score unstable chromosome aberrations in the first metaphases.

After 46 hours of culture at 37 °C, a mitotic inhibitor (Demecolcine, Life Technologies) is added and culture delayed for 2 hours. Cells are harvested and metaphase spreads prepared according to standard techniques. The slides were stained with Fluorescence Plus Giemsa (FPG) so that cells in their first and second divisions can be distinguished.

Unstable chromosome aberrations (dicentrics, centric rings and fragments) are scored microscopically only in the first complete metaphases (with 46 centromeres). The estimation of potentially received dose is made by reference to dose response curve produced by exposing blood *in vitro* to ^{60}Co gamma rays at 0.5 Gy.min^{-1} :

$$Y = 0.042 \times D + 0.054 \times D^2$$

(Y = yield of dicentrics and D = dose in Gy)

RESULTS and DISCUSSION

Several reasons are behind an expertise of biological dosimetry by cytogenetics, in case of suspicion of exposure to ionizing radiation. The most common one is that potentially exposured people, professional or public, do not carry a physical dosimeter when an overexposure is suspected (24 out of 33 cases). Even when there is possibility of *a posteriori* dosimetric reconstruction, uncertainties linked to the former justifies the examination of a biological parameter linked to the subject himself. A second reason is that, although a physical dosimeter is

effectively carried in the frame of their professional activities, the registered dose was abnormally high or took into account only localised irradiation (9 out of 33 cases). Five cases have concerned people submitted to chronic or fractionated irradiations, so that the biological estimate appeared as a useful adjunct in the knowledge of the current status of the patient.

Suspicion of overexposure can be divided in two major categories (Table 1): people whose professional occupation consists of working with radio-activity, radio-active sources or close to them (so called « professional » in table 1), and persons whose activities (professional or private) do not usually include usage of ionizing radiation. These former ones are called in this report with the general term « public ». It can therefore be surprisingly observed in the period 1992-95 a global imbalance between these two categories, an imbalance that however is not found for all years. In addition, whatever the category of persons is, the suspicion of irradiation always concerns the external exposition, never the internal contamination.

Table 1. Distribution of the number of personnel suspected of overexposure during the period 1992-95, according to the « professional » or « public » status of exposured persons.

	Number of cases		
Years	« professional »	« public »	Total
1992	5	10	15
1993	3	2	5
1994	2	1	3
1995	7	3	10
Total	17	16	33

According to their main professional activities, three classes of workers can be distinguished under the vocable « professional » (table 2). The distribution of suspected overexposure accidents shows a relative homogeneity of the accidental origin for these three classes : gammagraphy sources for testing purposes that constitutes the majority of typical industrial irradiations generally involves problems linked to dysfunction of the devices. In the health institutions, whether for radiographic use or patient treatment, it most often concerns the inattention of operators to protection and safety details. The greatest variability of the origin of accidents is linked to nuclear industry. This was probably due to the introduction in this category of three foreign persons who came to France for medical examination. On the one hand, it concerns a maintenance worker of the Chernobyl nuclear power plant during several years following the fallout and a physicist having periodically studied the reactor core inside the sarcophagus for 4 years. On the other hand, it includes a foreign physicist with badly irradiated hands by an accelerator of electrons. No dicentrics were scored in the first case what was coherent with clinical status of the patient. In the second case however, the biological dose (0.42 Gy) was clearly under-estimated in contrast with the diagnosis of total sterility. For the last one, the physical dosimetry reconstruction has suggested a dose within the range 10-40 Gy to both hands. This is quite different from 6 dicentrics scored in 500 lymphocyte metaphases which gives an integrated dose of 0.3 Gy. This example underlines well the limitations of cytogenetic dosimetry in case of very localized irradiation.

Table 2. Distribution of professional workers in three categories, and presence or absence of physical dosimeter (generally chest dosimeter) during the suspected overexposure accidents.

Physical dosimeter	with	without	Total
industrial radiography	4	2	6
health institutions	1	4	5
nuclear organisations	4	1	5
Total	9	7	16

The Table 2 also shows the proportion of potentially classified workers who carried a physical dosimeter (finger, chest) during their professional activities and/or their overexposure period. As far as the limited number of

overexposed cases observed allows, it can be stressed, therefore, a radioprotection problem, especially for the health workers category.

Within the « public » category, many overexposure suspects imply persons who was working beside a source of ionizing radiation and who believed have been exposed accidentally; for example, unknowingly entering in the irradiation room when the source is functional. The result of all this is generally short, acute exposure. However, a case of particularly important chronic irradiation concerns a family (the father and two children) having recovered and stored a small source of 4 Ci (148 GBq) of ^{137}Cs. The father who recovered the source, has kept it only for a short time in his office before giving it to his son. The son has kept it for sixteen months in his bedroom, which was beside his sister's room. The doses estimated by biological dosimetry correctly reflected the duration of the exposition as well as the personal dose distribution (in equivalent whole body, 0.5, 0.9 and 0.64 Gy respectively). In contrast, it does not allow to reconstitute correctly the dose absorbed by the son notably because a strong lymphopenia.

More generally, the estimated dose has been « zero dose » or doses below 0.5Gy in the great majority of cases of these three years as summarised in Table 3. A « zero dose » means that no dicentrics have been found in the observed metaphases, often 500. However, there exists a possible minimum number of dicentrics to be observed in normal individuals, which is approximately 1 for 2000 metaphases. This suggests that the estimated dose is in fact included in the background considering that the accuracy of the method is proportional to the number of metaphases observed. On the curve of reference of the laboratory (^{60}Co, 0.5 $Gy.min^{-1}$), this corresponds to a 95% confidence interval of 0.2 Gy.

Table 3. The dose ranges estimated by biological dosimetry related to the repartition of the overexposured cases according to the different categories of implied people.

Dose range (Gy)	« zero dose »	0 - 0.19	0.2 - 0.49	0.5 - 0.9	Total Number
industrial radiography	2	1	3	0	6
health institutions	5	0	0	0	5
nuclear organisations	3	0	2	0	5
public	8	0	3	5	17
Total	18	1	8	5	33

It is necessary to note that the main limitation of the biological dosimetry is to give a dose integrated to the body, which poses problems in case of localized irradiations. It is indisputable that the majority of overexposures to ionizing radiations concern localized irradiations. In the three cases where hands have been mainly exposed to doses above 10 Gy, the dose given by dicentric scoring did not exceed 0.3 Gy. Furthermore, the selection of the reference curve has incontestably an importance on the estimation of the dose, independently of the presence or not of dicentrics, since it would have to take into account the quality of the radiation and the dose rate when they are known. In our experience for the years 1992-95, it appears that implied radiation sources are essentially from ^{192}Ir,^{60}Co, X -rays with a low dose rate. This justifies therefore the use of a reference curve established from gamma radiation of ^{60}Co with 0.5 $Gy.min^{-1}$. The comparison with the physical dosimetry in the cases where the former is known with a good accuracy shows elsewhere a good agreement with the cytogenetic estimate. Nevertheless, other reference curves for other qualities of radiation and dose rate are prerequisite.

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THE DICENTRIC ELIMINATION IN LYMPHOCYTES OF PATIENTS EXPOSED IN THE TIME OF CHERNOBYL ACCIDENT

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INTRODUCTION

The chromosome studies of atomic bomb survivors in Hiroshima and Nagasaki in some decades after the irradiation demonstrate a primary elimination of cells carrying unstable-type chromosome aberrations (dicentrics, rings, fragments). Its are met in less than 10% of all aberrant cells. A frequency of stable-type chromosome aberrations is remained almost unchanged in time in blood lymphocyte cultures of exposed persons (1). Therefore the stable-type chromosome aberrations are considered the most effective indicator of radiation dose in this time. However in first some years after exposure a level of dicentrics (a principal kind of chromosome aberrations for biological dosimetry in a near time after an irradiation) may be sufficiently high although over the first 4 years the decrease of the dicentric frequency at a rate of about 43% per year was observed (2). The number is a result of a cytogenetical investigation of a group of patients treated with X-rays for ankylosing spondylitis. Until recently the similar works with a use of therapeutic partial prolonged fractionated irradiation were a principal source of a data for study of the elimination chromosome aberration regularities. The initial (immediatly after an irradiation) and several repeated cytogenetic analysises after acute external total uniform or non-uniform exposures were carried out more or less in detail mainly for a small number of persons irradiated accidently (3-8). Unfortunately large numbers of persons irradiated different doses appeared after two radiation catastrophes of 1986-1987 years (Chernobyl and Goiania accidents). In this paper the results of reported counts of radiation-induced dicentrics for Chernobyl patients are presented. Its are important for an investigation of possibilites of biological retrospectival dose estimations.

MATERIALS AND METHODS

The initial chromosome aberration analyses of lymphocyte cultures from peripheral blood and bone marrow were made for a large number of exposed people during the nearest 1.5 days - 7 weeks after Chernobyl accident. The cytogenetical dose estimations were produced for 192 patients in the range from 0 Gy (dicentris were not found in lymphocyte cultures from 48 patients) to 13.7 Gy. Relatively uniform affection of a body (hemopoietic tissue) with the exception of a skin could be anticipated on the basis of a conformation of dicentric distributions to a Poisson distribution in lymphocyte cultures from the most of these patients who had been exposed at doses that were critical for the develepment of the bone marrow syndrome. Repeated cytogenetical examinations were made for the study of elimination chromosome aberration regularities. The peripheral blood lymphocyte cultures were incubated during 50 or 67 hours. The chromosome aberration counts were made in the first in vitro metaphase cells determined by fluorescense plus Giemsa staining.

108 lymphocyte cultures from 61 patients were investigated in the general time range from 3 to 30 months after accident. The number of repeatedly analysed lymphocyte cultures (in different times) for individual patients fluctuated from 1 to 4. The analysed cell numbers in several cultures varied from 15-41 cells (4 cultures) and 50-98 (9 cultures) to 100-200 cells (95 cultures).

RESULTS AND DISCUSSION

All patients were divided in development from initially estimated doses at 3 groups: exposed in doses 0.4-1.9 Gy (21 patiants, 34 repeated cultures), 2.1-3.7 Gy (26 patients, 48 repeated cultures) and 3.9-8.7 Gy (14 patients, 26 repeated cultures). After the termination of cytogenetical counts the percents of cells with dicentrics, average frequences of dicentrics and dicentric frequences in cells with dicentrics were selected for a sequent examination. The relationships (exponentional model) between percents from initial levels for repeated observed values of these cytogenetical indicies and the time after irradiation (in mounths) were received. The corresponding coefficients of the equations are presented in Table.

In general, our data evidently demonstrate the faster rates of decrease of average frequences of dicentrics and percents of cells with dicentrics in time after greater ininial doses than smaller initial doses. Also, dicentric frequencies in cells with dicentrics decreased in time slower than average frequencies of dicentrics. This fact and the reproduction of non-aberrant cells have to result in an apearance of the overdispertion of dicentric distributions by cells with respect to Poisson distribution. Apparently, the cells carrying the greater quantity of dicentrics relatively more rarely divide in vivo than cells with smaller quantity of dicentrics, although in general all cells divide some time or other. The dicentric frequency in cells carrying dicentrics made 1.00-1.25 in group of 0.9-1.4 Gy (that was greater 1.20 in 1 culture), 1,08-1,59 in group of 2.1-3.7 Gy (that was smaller 1.20 in 7 cultures and greater or equal 1.50 in 2 cultures) and 1.43-3.52 in group of 3.9-8.7 Gy (that was smaller 1.50 in 2 cultures).

The large individual variability was observed for rates and characteristics of decline of dicentric frequencies independently from initially valueted doses. Its could decline by degrees or by leaps. The long time periods of persistent yields of dicentrics could find out. Those were up to 19 mounths in the group 0.4-1.9 Gy, up to 11-12 mounths in the group of 2.1-3.7 Gy and up to 10 mounths in the group of 3.9-8.7 Gy. Sometimes the undulating changes of dicentric frequencies were observed 1 year after the accident (in general at the decreased levels in comparison with the initial yields).

Thus, essential difficulties and indefinities will appear in the time of the cytogenetical biological retrospectival dose estimations by dicentrics at the blood sampling delay of a few years.

CONCLUSIONS

1. The average rate of dicentric elimination was more after higher doses.
2. Dicentric frequencies in cells with dicentrics decreased in a development from time slower than average dicentric frequencies.
3. The essential individual variation was observed.

Table

The relationships between several cytogenetical indices (Y, % from initial values) and the time (t, months) after the irradiation at Chernobyl accident

Cytogenetical indices	Initial valuations of doses, Gy	Y = b•exp(a•t)		Coefficient of correlation	p
		a	b		
Avarage frequency of dicentrics per 100 cells	0.4–1.9	–0.049	93.7	–0.71	< 0.001
	2.1–3.7	–0.071	91.8	–0.85	< 0.001
	3.9–8.7	–0.112	97.5	–0.87	< 0.001
Percent of cells with dicentrics	0.4–1.9	–0.047	93.7	–0.72	< 0.001
	2.1–3.7	–0.069	94.6	–0.85	< 0.001
	3.9–8.7	–0.102	99.5	–0.88	< 0.001
Dicentric frequency in cells with dicentrics	0.4–1.9	–0.002	99.5	–0.27	< 0.1
	2.1–3.7	–0.004	97.5	–0.33	< 0.01
	3.9–8.7	–0.011	97.5	–0.50	< 0.01

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THE FREQUENCY OF DICENTRIC CHROMOSOMES IN RELATION TO ^{131}I CONTAMINATION

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INTRODUCTION

In vitro investigations conducted by now have indicated a correlation between the applied doses of ionising radiation and frequency of dicentric chromosomes (1). In vivo investigations do not always confirm that relationship, especially not in case of subjects occupationally exposed to ionising radiation. The aim of this paper was to elucidate that possible disagreement between registered doses and the number of chromosomal aberrations may be ostensible. Risk estimation should therefore involve precisely defined groups of examinees and not individual subjects.

SUBJECTS

Out of the total number of 107 subjects included in the study, 41 subject were taken as a control group and 66 workers employed at a department of nuclear medicine represented exposed group. The subjects had not been exposed to radiation for therapeutic or diagnostic purposes over the preceding year and had not taken any chemo-therapeutic drugs.

The activity of the department of nuclear medicine involves the use of radioactive material for therapeutical treatment and diagnostics. The variety and wide range of specific jobs at the department of nuclear medicine requires the staff including different professions. All these employees; such as cleaners, nurses, technicians, physicians, biologists, physicists and chemists, have in common equal possibility to be irradiated in their everyday work. Irradiation may occur during the handling of radioactive material, in the process of application of radioactive material to patients, from the patient subjected to application of radioactive material and by contamination. It should be emphasized that work at the department of nuclear medicine involves a risk for the employees to be continuously exposed to ionising radiation over the whole working hours. Since the risk from external irradiation, from contaminatin and intake of radionuclides in the body depends mainly on the nature of specific tasks of the employees, the exposed group was divided into four subgroups by corresponding jobs and tasks performed daily at the department of nuclear medicine.

METHODS

Analysis of chromosomal aberrations were carried out on standard 24-hour lymphocyte cultures. Two hundred metaphases were analysed per each sample, and only first in vitro metaphases were analysed. Chromatid and chromosomal damages detected per each analysed metaphase were classified into chromatid breaks, chromosomal breaks and bicentric chromosomes (2).

Samples of urine to be gamma-spectrometrically analysed were collected in glass dishes over 24 hours. Daily aliquot in the volume of 110 ml per subject was taken for analysis. Gamma-spectrometrical analysis of 24-hour urine samples was carried out using Ge (Li) detector, resolution 1.78% (Co), efficiency 16.8% with 4K channel analyser, Canberra series 10. The detector was protected with a lead layer, thickness 10 cm, 1 mm of cadmium and 2 mm of copper. The samples were measured in a cillindric dish, volume 110 ml. Specific activities of ^{131}I were calculated on basis of known efficiency of the counter, percentage of gamma rays, quantity of the sample and intensity of photo peak.

RESULTS AND DUSCUSSION

The frequency of total chromosomal aberrations found in the control group was 0.75%, and in the group of employees of the department of nuclear medicine 1.84%. The frequency of individual chromosomal and cromatid aberrations per cell was given in Table 1.

Group	Cromatid break	Chromosomal break	Acentric	Dicentric
Control	1.34 ± 1.33	0.78 ± 1.0	0.63 ± 0.86	0
Exposed	1.97±1.22	1.56 ± 1.52	1.67 ± 1.62	0.48± 0.85

Table 1. Total frequency of cromatid and chromosomal aberrations per cell

By means of gammspectrometrical analysis ^{131}I contamination was stated in 11 subjects of the exposed group. Activity of I in urine ranged in some workers from 0 to 78.66 Bq/kg. In the control subjects ^{131}I contamination was not detected.

Comparing aberration frequency in the control and exposed group the difference between the groups was found at the level of significance $P < 0.05$, except for chromosomal breaks. The results obtained correspond with the data reported by Pohl-Ruling (3,4) pertaining to frequency of chromosomal aberrations in the lymphocytes of subjects exposed to external gamma radiation and internal radiation of radon and its decay products.

Comparing the frequency of dicentric chromosomes with ^{131}I burden in certain examinees no regularity was found between the two investigated parameters.

According to the obtained data and equal possibility of external exposure to radiation and contamination workplaces and tasks at the department of nuclear medicine, the staff was divided into four subgroups regardless of their profession.

The average values for the frequency of dicentrics per cell and examinees and data on average contamination by individual subgroups were given in Figure 1.

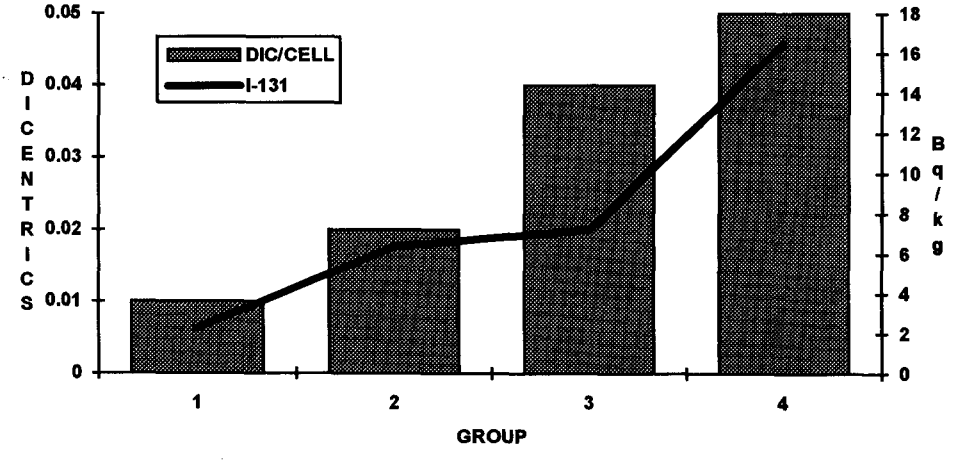

Figure 1. Average frequency of dicentrics, average contamination with ^{131}I for subgroups of examinees in the department of nuclear medicine

It can be seen that increase in the average ^{131}I contamination is followed by the average frequency of dicentric chromosomes in all four subgroups. The fact that dicentric chromosomes are found also in the subjects in whom ^{131}I contamination was not observed can be explained by the biological half-lives of ^{131}I and the time of its decay. As biological half-life of ^{131}I is 12 days and of the decay 8.04 days, there is low probability that it would be detected in urine during periodical sampling, taking into account that contamination is not

continuous. This fact is opposed by the fact that persistence of chromosomal aberrations takes several months to several years.

Taking this into account it seems appropriate to believe that for this reason the authors in previous investigations have not found any correlation between the registered doses and incidence of dicentric chromosomes.

The obtained data indicated the existence of regularity of the dose-response relationship in case of dicentric chromosomes even in cases of irradiation with small fractionate doses over the prolonged period of exposure.

Analysing the data obtained in this kind of investigations it should be mentioned that conclusions should not rely on physical measurements only. The subjects should be observed in subgroups due to their tasks which ensures great probability that doses received would be equal.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE HOW TO ESTABLISH BIODOSIMETRIC LAB. (THE SCIENTIFIC BASES AND REQUIRED EQUIPMENTS)

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MAJOR SCIENTIFIC TOPIC NUMBER 3.4.... (see page 7)

ABSTRACT (See instructions overleaf)

Chromosomal aberration in human T-Lymphocyte of peripheral blood are sensitive indicators of radiation exposures even at low levels of dose. At present, chromosome analysis is the method of choice for a biological estimation of an equivalent whole-body dose of actual or suspected over-exposures to ionising radiation.

Cytogenetic dosimetry is a valuable tool in supplementing physical dosimetry in practical radiation protection since, individual ; body doses down to 100 mSv. gamma-rays can be reliably detected. There are some limitations for a quantification of individual acute exposures below 50 mGy; for partial-body, chronic exposures and in particular for incidents with incorparation of radionuclides with selective depositions.

Four biodosimetric methods are used for the evaluation of the radiation dose, viz: glycophorin-A (GPA)-somatic mutations; chromosome translocations; micronulei; and dicentrics. Two of these biodosimeters; (GPA) and translocations are stable with time post-exposure and are therefore expected to integrate radiation damage under chronic exposure conditions. The other two (micronuclei and dicentrics); are unstable, and can only detect recent exposures.

Two different approaches were used to detect chromosome translocations in peripheral blood lymphocytes -viz: fluorescence in situ hybridization (FISH) and G-banding.

CYTOGENETIC DOSIMETRY OF RADIATION DAMAGES IN DOMESTIC ANIMALS

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ABSTRACT

In case of mass radiation of people and animals the estimation of radiation dose is the basic prerequisite for undertaking protection measures and the triage of radiated individuals. In the early phase of the accident, estimation of radiation dose in animals only on the base of the physical dosimetry is often deficient and unreliable.

In this paper a possibility of using biodosimetric methods for estimation of absorbed radiation doses in domestic animals in case of an accident is discussed. For this purpose radiation of full animal blood with 6 different radiation doses was carried out. After the blood radiation, preparation and cultivation of lymphocytes as well as a certain number of specific chromosomal aberrations for each radiation dose were performed. The obtained results indicate that the number of specific chromosomal aberrations increases with the increase of radiation dose.

INTRODUCTION

It is necessary to know radiation doses in the first phase of an accident to be able to carry out protection measures effectively in case of radiation emergency (1). Estimation and measuring of exposure doses can successfully be performed by using physical methods of ionizing radiation dosimetry. However, from the aspect of the biological effect on live organisms, it is necessary to know the absorbed radiation dose in the tissue of each individual (2-4). Physical dosimetry methods can not be applied for measuring of the absorbed dose of ionizing radiation in case of an accident. This particularly refers to the estimation and establishing of the absorbed radiation dose in domestic animals and people, on the basis of which it is possible to estimate the level of radiation damage and to undertake further protection measures. Because of this, the aim of this investigation was to determine the level of radiation damage in pigs on the basis of cytogenetic methods of ionizing radiation dosimetry.

MATERIAL AND METHODS

The blood of healthy pigs, in which the number of spontaneous chromosomal aberrations was previously determined, was irradiated with 6 different doses of high-energy-X-rays (0.1; 0.2; 0.5; 1.0; 2.0 and 3.0 Gy). After irradiation, lymphocyte preparation and cultivation according to the modified method of Morhed were performed and the number of specific chromosomal aberrations dicentric and ringlike chromosomes) determined. The analysis of the number of chromosomal aberrations was carried out on at least 1000 good visible metaphase figures for each radiation dose.

RESULTS AND DISCUSSION

In the blood of healthy unexposed pigs aberrations of chromatid type as gas and chromatid breakage were mostly represented. The total number of spontaneous chromosomal aberrations in healthy pigs was 0.1%. After exposure of blood to high-energy-X-rays, already during the first 24 hours after irradiation a significant increase of the number of specific chromosomal aberrations took place (table 1).

Table 1. Frequency of chromosomal aberrations in lymphocytes of pigs after in vitro irradiation of blood with X-rays.

Dose in Gy	No. of anal. cells	Structural aberrations of chromosomes: Chromatid Gap	Chromatid Break.	Chromosomal Break.	Chromosomal Ac.fragm.	D+R	Change of ploidity	% of aberrations
0.0	1105	-	1	-	-	1	4	0.54
0.1	1000	4	1	-	3	3	4	1.50
0.2	1000	9	3	-	3	5	3	2.30
0.5	1000	2	-	-	7	6	3	2.80
1.0	1000	4	-	3	16	18	7	4.80
2.0	1000	15	-	1	91	76	8	19.10
3.0	1000	-	-	-	241	255	16	51.20

D - dicentric, R - ring

It can be noticed from the obtained results that with the increase of radiation dose increases also the number of specific chromosomal aberrations. For the lower radiation doses (up to 0.5 Gy), the number of aberrations was represented with a linear parameter, and for higher dose levels with a square parameter. On the basis of obtained results, it is possible to work out the calibration dose-response curve, which is the mathematical, statistical and graphic presentation of an experimentally determined relation between the number of specific chromosomal aberrations and the absorbed radiation doses, what will be the subject of our further investigations.

CONCLUSION

Cytogenetic dosimetry of ionizing radiation is one of the methods for early detecting of radiation damages in domestic animals. By applying this method, already during the first 24 hours after irradiation of pigs (much before the appearance of clinical symptoms of the disease) it is possible to estimate the level of radiation damage on the basis of specific chromosomal aberrations and to undertake protection measures and/or measures of economical exploitation of irradiated animals. For more reliable estimation of the absorbed radiation dose it is necessary to work out also calibration dose-response curves, which are a mathematical, statistical and graphic presentation of experimentally determined relationship between the number of chromosomal aberrations and the absorbed radiation doses. On the basis of these curves it is possible to estimate the absorbed radiation doses in people and animals in accidental and emergency situations.

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SERUM PROTEINS IN ATOMIC INDUSTRY WORKERS

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INTRODUCTION

At present time the clinical and experimental materials have been accumulated, which evidence about the influence of irradiation on different sides of the protein metabolism (1). The most of these studies have been conducted at the early time period after acute and chronic irradiation exposures. Because data on more late consequences of irradiation influence on the protein metabolism in humans are not numerous and have some differences, it is necessary to have future studies of this problem. The results, obtained for the last years in epidemiological studies of humans, indicate on the relationship the protein metabolism changes with ageing, increased morbidity and mortality and confirm the importance to continue the studies in this area (2).

MATERIALS and METHODS

989 workers of the "Mayak" atomic industry plant (683 male, 306 female) at ages of 37 - 78, exposed to chronic external and/or internal irradiations 17-40 years ago, were examined. The summary doses of external γ-irradiation were in the range of 0.01 - 7.6 Gy. The incorporation of plutonium-239 in organism in the range of 0.148 - 25.27 kBq was determined for 723 persons examined, including 28.2% cases, in which incorporation exceeded 1.48 kBq. In serum the following parameters were determined: total protein level by the biuret reaction method, protein fractions by electrophoresis on acetate-cellulose membranes and further by densitometric estimation, the level of immunoglobulins (IgA, IgG, IgM) by radial immunodiffusion of Mancini. The absolute and relative values of protein fractions were calculated. The serum proteins entropy was used as integral index of relative content of electrophoretic fractions. Relationships between the studying parameters and 11 factors: 2 radiation and 9 non-radiation (age, sex, body mass index (BMI) in kg/m^2, smoking, alcohol abuse, chronic non-specific lung diseases (CNLD), chronic gastritis (CG), chronic cholecystitis (CC), and rheumatic arthritis (RA)) were estimated. The protein synthesis function (PSF) of liver was studied in separate investigation. Prealbumin (PA) by method of radial immunodiffusion Kit, albumin by reaction with BCP, and activity of cholinesterase (CHE) by method with achetylcholinechloride in blood serum of 190 workers were determined. Relationships of the serum proteins with radiation and non-radiation factors by stepwise regression analysis were studied (3).

RESULTS and DISCUSSION

The Table 1 presents the results of comparing the protein parameters in 3 groups of atomic industry workers with different summary doses of external γ-irradiation. The first group (the comparison group) consists of the persons who have the annual accumulated doses of external γ-irradiation not exceeded 0.05 Gy. The second and the third groups comprise the persons with more significant dose of irradiation. This Table also shows that there are changes of some serum proteins in examined persons with increasing of summary doses, e.i., the increases of total protein level, absolute value of alpha-1-globulin, absolute and relative values of alpha-2-globulin and beta-globulin, entropy values of serum protein, IgG level and the decrease of relative value of

Table 1

Serum proteins (X±S.E.) in the persons exposed to external γ-irradiation

Proteins	Summary doses of γ-irradiation, Gy		
	(1) 0.01-0.50	(2) 0.51-4.00	(3) 4.01-7.60
n	315	562	112
Total protein, g/L	74.5±0.36	74.9±0.26	76.7±0.61**
Albumin, %	59.6±0.29	58.6±0.20*	57.2±0.50**
g/L	44.4±0.30	43.8±0.19	43.9±0.51
Alpha-1 globulin, %	3.3±0.08	3.4±0.06	3.6±0.14
g/L	2.46±0.06	2.56±0.04	2.78±0.11*
Alpha-2 globulin, %	8.7±0.14	9.1±0.10*	9.9±0.25**
g/L	6.5±0.11	6.84±0.08*	7.62±0.21**
Beta globulin, %	12.7±0.16	13.1±0.11*	13.6±0.30*
g/L	9.45±0.13	9.77±0.09*	10.44±0.24**
Gamma globulin, %	15.6±0.20	15.8±0.16	15.6±0.38
g/L	11.66±0.17	12.04±0.17	2.01±0.31
Proteins entropy, a.u.	1.2439±0.0081	1.2722±0.0055*	1.3117± 0.0134**
Ig A, g/L	2.24±0.06	2.29±0.05	2.28±0.11
Ig G, g/L	11.39±0.18	11.50±0.15	12.27±0.35**
Ig M, g/L	1.23±0.05	1.26±0.03	1.33±0.09

* - $p < 0.05$ against to 1 group; ** - $p < 0.05$ against to 1 and 2 groups

albumin. On the whole, the changes were more significant in the group with more high doses of exposure. Levels of IgA and IgM in the groups compared did not significantly differ (see Table 1). The changes of protein parameters in people with different levels of plutonium-239 incorporation were not found. The obtained results indicate that there are relationships between the protein parameter changes in atomic industry workers and external γ-irradiation exposure. The multifactory analysis has been made for obtaining more precise dependence of the studuing parameters on the external irradiation, and for confirming the absence of relations of those with internal exposure, and for determining of contribution the irradiation and non-radiation factors in serum protein changes.

It was found by the stepwise regression that 8 of 11 factors, which were taken for analysis, have the significant relationships ($p < 0.05$) with the protein parameters. The Table 2 contains the partial determination coefficients, the values and the signs which characterize the influence of each factor, allowing to compare the factors. This table also includes the multiple determination coefficients.

The linear equations of multiple regression were obtained as the results of this analysis. These equations reflect the relible dependence of the protein parameters on the summary dose of external γ-irradiation and non-radiation factors:

Total protein, g/L =69.88+0.55*X3+0.14*X5 ($F=13.67$; $p < 0.001$)

Alpha-1 globulin, g/L =2.46+0.05*X3 ($F=4.49$; $p < 0.05$)

Alpha-2 globulin, g/L =5.73+0.21*X3+0.51*X4 ($F=20.94$; $p < 0.001$)

Beta globulin, g/L =7.44+0.16*X3+0.07*X5 ($F=14.62$; $p < 0.001$)

Proteins entropy, a.u.=1.1713+0.0016*X1+0.0217*X2+0.0091*X3 ($F=9.97$;$p<0.001$)

where: X1 - age, year; X2 - sex (1 - male, 2 - female); X3 - summary dose of γ-irradiation, Gy; X4 - smoking (1 - no, 2 - yes); X5 - BMI; F - Fisher criterion.

Table 2

Partial (d) and multiple (D) determination coefficients (%) of factors influenced serum proteins level

Proteins	Partial determination coefficients of factors*								D
	Age	Sex	γ-irradiation	Smoking	Alcohol	BMI	CNLD	CG	
Total protein, g/L	-	-	2.3	-	-	0.9	-	-	3.2
Albumin, %	-0.5	-1.3	-1.6	-	-	-0.5	-	-	3.9
g/L	-0.6	-	-	-	-	-	-	-	0.6
Alpha-1 globulin, %	-	-	-	0.8	-	-	-	-	0.8
g/L	-	-	0.6	-	-	-	-	-	0.6
Alpha-2 globulin,%	-	-	2.0	2.1	-	-	-	-	4.1
g/L	-	-	3.4	1.5	-	-	-	-	4.9
Beta globulin, %	-	-	0.6	-	-	1.3	-	-	1.9
g/L	-	-	1.6	-	-	1.9	-	-	3.5
Gamma globulin,%	-	-	-	-0.8	-	-	-	-	0.8
g/L	-	-	-	-0.9	-	-	-	-	0.9
Proteins entropy, a.u.	0.7	0.9	1.6	-	-	-	-	-	3.2
Ig A, g/L	0.6	-	-	-	0.6	0.9	2.6	-	4.7
Ig G, g/L	0.5	-	-	-1.9	-	-	-	1.0	3.4
Ig M, g/L	-	1.1	-	-	-	-	-	-	1.1

* - d=0.5-0.7: $p < 0.05$; d=0.8-1.2: $p < 0.01$; $d > 1.2$: $p < 0.001$

The values of regression coefficients for summary dose of γ-irradiation evidence that increasing the dose exposure by 1.0 Gy causes the increase of total protein level by 0.55 g/l, including alpha-1-globulin by 0.05 g/l, alpha-2-globulin by 0.21 g/l, and beta-globulin by 0.16 g/l. Proteins entropy at the same dose of exposure increases by 0.0091 a.u. Theses results were well coincided with the data received in atomic-bomb survivors in Japan (4). The level of PA in the workers with plutonium-239 incorporation more than 1.48 kBq was less (259+6.0 mg/l, $p < 0.05$) than in the control group (276+3.9 mg/l). Frequency of hypo-PA-emia in these workers was not increased, but hyper-PA-emia was absent. The level of albumin and activity of CHE in those was not changed. Level of PA and other indicators of PSF of liver in the workers with different summury doses of external γ-irradiation was not changed.

Thus, chronical external and internal irradiation influence on different sides of protein metabolism of the atomic industry plant workers exposed 17-40 years ago.

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RADIATION INDUCED CHROMOSOME ABERRATIONS IN CULTURED LYMPHOCYTES FROM 4 PATIENTS WITH SKIN RADIATION INJURIES

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INTRODUCTION

Therapeutic irradiation results sometimes (in about 20 per cent cases) in formation of long-term skin radiation injury. Its treatment is difficult and more often ineffective. As the majority of patients exposed under similar conditions has no long-term skin reactions, it seems plausible that such reactions are conditioned by an enhanced individual skin radiosensitivity. Determination of such individuals before the onset of radiation therapy should become an important prophylactic measure. Taking in mind this objective and assuming that the individual proneness to the formation of long-term skin ulcers is (at least partly) of genetic origin, we intend to elucidate whether the chromosome radiation response of patients with late skin injuries differs from that of healthy persons. Peripheral lymphocytes are chosen as the test system because radiosensitivity of blood is one of essential constituents of integral organism radiosensitivity.

Aliquots of venous blood were taken from 4 healthy volunteers (below referred to as donors 1-4) and 4 patients of the MRRC clinic with skin radiation injuries (patients 5-8) appeared after courses of therapeutic irradiation. Preparation of lymphocyte cultures followed standard technique (1), with slight modifications. Lymphocytes were exposed to ^{60}Co γ-rays at about 20 rad/min; irradiation schedules and doses are given below. Cells were harvested at 51 h after PHA-stimulation. On the obtained metaphase preparations, all the aberrations which needed no kariotype identification for their detection were scored. The following cytogenetic end-points were assessed: (1) aberration levels in unirradiated cultures, (2) radiosensitivity at the G_0 stage, (3) radiosensitivity at the G_2 stage, (4) capability to the induction of an adaptive response.

Abbreviations: PHA, phytohaemagglutinin; D., donor; D_m, mean value for healthy donors; AM, aberrant metaphases; CA, total chromosome aberrations; CsA, chromosome type aberrations; CsF, chromosome type fragments; CsE, chromosome type exchanges; Dic, dicentrics; CtA, chromatid type aberrations; CtF, chromatid type fragments; CtE, chromatid type exchanges; AR, adaptive response.

ABERRATION LEVELS IN UNIRRADIATED CULTURES

The data on aberration frequency in unirradiated cultures are shown in Table 1. As χ^2-test showed the homogeneity of the frequencies for healthy donors, the table gives the pooled values. Aberration yields in patients show a tendency to exceed those in healthy donors, but only in patients 7 and 8 the difference is significant for most aberration types, being the most expressed for chromosome exchanges. It is natural to suppose that the increased aberration yield in patients is associated mainly with therapeutic irradiation they had received earlier. Time elapsed after the irradiation was more than 20 months for patients 5, 6 and less than 16 months for patients 7, 8; thus, our results agree with existing data on decrease of frequency of *in vivo* induced unstable aberrations in lymphocytes with time after exposure (2). However, some features of the obtained data are in a contradiction with what should be expected as a result of *in vivo* local irradiation. These are the significantly increased frequency of chromatid fragments in patient 8, and Poisson-type cellular distribution of aberrations in patient 7. It is possible that these features are associated with individual peculiarities of spontaneous aberrations patterns in these two patients.

Table 1. Aberration frequencies (per 100 cells) in unirradiated lymphocyte cultures. In brackets, standard errors. *, significant difference ($P<0.05$) from D_m.

D.	AM	CA	CsA	CsF	CsE	Dic	CtA	CtF	CtE
D_m	1.38 (0.52)	1.38 (0.52)	0.13 (0.16)	0.13 (0.16)	0	0	1.25 (0.50)	1.00 (0.45)	0.25 (0.45)
5	4 (2)	5 (2)	4 (2)	4 (2)	0	0	1 (1)	1 (1)	0
6	3 (2)	3 (2)	0	0	0	0	3 (2)	3 (2)	0
7	10* (3)	12* (3)	10* (3)	4 (2)	6* (2)	3 (2)	2 (1)	1 (1)	1 (1)
8	23* (4)	35* (6)	21* (6)	10* (3)	11* (3)	7* (3)	14* (4)	10* (3)	4 (2)

RADIOSENSITIVITY AT THE G_0 STAGE

In this set of experiments, the blood samples were exposed to 2 Gy gamma-rays; lymphocyte cultures were prepared and stimulated immediately after the exposure. The frequencies of induced aberrations are given in Table 2. χ^2-test having showed the homogeneity of values for healthy donors, except those of chromosome type exchanges, these values were averaged. Yields of total aberrations and of chromosome type aberrations exceed controls significantly in three of four patients; in addition, patients 5 and 7 reveal an increased frequency of chromosome type fragments, and patient 8 - that of chromatid type aberrations (fragments). Analysis of cellular distribution of various aberration types showed that for healthy donors all the distributions were poissonic, while for patients some were poissonic, others were not. However, two kinds of the distributions, namely that of total chromosome type aberrations and that of chromosome type fragments, differed from Poisson distribution significantly ($P<0.01$) for each patient.

The above results allow to suggest the frequency of chromosome type aberrations induced at the G_0 stage as an end-point which is essentially different in individuals with enhanced skin radiosensitivity as compared to healthy persons. The difference resides in the facts that in patients (1) this frequency exceeds the control value and (2) its distribution is non-poissonic.

Table2. Aberration frequencies (per 100 cells) in lymphocyte cultures irradiated by 2 Gy at the G_0 stage. In brackets, standard errors. *, significant difference ($P<0.05$) from D_m.

D.	AM	CA	CsA	CsF	CtA	CtF	CtE
D_m	40.5 (2.8)	53.0 (3.3)	50.8 (3.2)	16.0 (1.8)	2.5 (0.7)	2.0 (0.6)	0.5 (0.3)
5	45 (5)	68 (8)	63 (8)	28* (5)	5 (2)	2 (1)	3 (2)
6	49 (5)	73* (9)	71* (8)	21 (5)	2 (1)	2 (1)	0
7	46 (5)	79* (9)	71* (8)	29* (5)	8 (3)	4 (2)	4 (2)
8	57 (8)	126* (19)	94* (16)	26 (9)	31* (9)	26* (9)	6 (4)

RADIOSENSITIVITY AT THE G_2 STAGE

The below data were obtained only for donors 2-7. The lymphocyte cultures were exposed to 0.5 Gy 3 h before harvest. The comparison of aberration frequencies showed that there was no difference from mean donor values for patients 5 and 6, while for patient 7 essentially all the frequencies exceeded the controls significantly ($P<0.05$). The analysis of cellular distribution was made for frequencies of total aberrations, chromatid aberrations and chromatid fragments. For healthy donors, all the three end-points gave a non-poissonic pattern; the same was found for patients 6 and 7; in patient 5, on the contrary, all the distributions were poissonic. These results are insufficient to reveal an end-point permitting to distinguish between chromosome radiation responses of the patients and healthy persons. However, the data obtained for patient 7 suggest that the further investigation might be more fruitful.

ADAPTIVE RESPONSE

To study the ability of lymphocytes to produce the AR, the blood samples were exposed to the adaptive dose 0.05 Gy, then the cultures were set, stimulated and after 48 h incubation exposed to the challenge dose 0.5 Gy. This schedule of irradiation had been tested in our laboratory and proved to give a good reproducibility of the results of AR studies (3). The obtained data are shown in Table 3. If an aberration frequency induced by both the doses appeared significantly less than that induced by the challenge dose only, it was regarded as a positive AR. It can be seen from the table that the AR was observed for most of the investigated end-points in healthy donor 3 and in patient 7. In donor 2, on the contrary, a sensitizing effect of the adaptive dose was detected. For donors 4, 5 and 6 there were no differences between experimental alternatives.

Table 3. Data on the AR induction. Comparison of aberration frequencies induced by challenge dose only (0C) and by adaptive and challenge doses (AC). In brackets, standard errors. #, AC exceeds 0C significantly; *, 0C exceeds AC significantly ($P<0.05$).

D.		AM	CA	CsA	CsF	CsE	Dic	CtA	CtF	CtE.
2	0C	37(6)	96(10)	3(2)	2(1)	1(1)	1(1)	93(10)	90(9)	3(2)
	AC	47(7)	133(12)#	7(3)	5(2)	2(1)	0	126(11)#	121(11)#	5(2)
3	0C	48(6)	88(11)	3(2)	0	3(2)	3(2)	85(11)	75(11)	10(4)
	AC	28(5)*	40(7)*	0	0	0	0	40(7)*	36(8)*	4(2)
4	0C	48(7)	79(12)	2(2)	2(2)	0	0	78(12)	59(10)	19(6)
	AC	52(5)	81(10)	5(2)	2(2)	2(2)	2(2)	76(10)	69(9)	7(3)
5	0C	47(7)	76(9)	2(1)	2(1)	0	0	74(8)	67(8)	7(3)
	AC	48(7)	100(10)	7(3)	5(2)	2(1)#	1(1)	93(10)	89(9)	11(3)
6	0C	42(6)	99(10)	4(2)	1(1)	3(2)	3(2)	95(10)	86(9)	9(3)
	AC	50(4)	92(7)	3(1)	2(1)	2(1)	1(1)	89(7)	87(7)	3(1)
7	0C	72(4)	197(14)	19(4)	12(3)	7(3)	5(2)	178(13)	154(12)	24(5)
	AC	52(8)*	98(15)*	4(3)*	2(2)*	2(2)	0*	92(15)*	83(14)*	9(5)*

The analysis of cellular distribution of aberrations showed that the AR observed in donors 3 and 7 was associated mainly with a decreased number of aberrant cells. Also, it seems noteworthy that for these two donors, and only for them, non-poissonic distributions obtained for aberrations induced by the challenge dose "give way" to poissonic distributions in the case of exposure to both adaptive and challenge doses. Conceivably this observation might be useful for understanding of the AR mechanisms. As to the question whether the examined patients can be distinguished from healthy persons by their ability to the AR induction, of course it cannot yet be answered basing on the above preliminary results, and needs further work.

CONCLUSION

A preliminary evidence has been gained that, in donors with enhanced skin radiosensitivity, lymphocyte radiosensitivity is also increased. This increase reveals itself in some features of "spontaneous" aberration level, and in elevated yield and shifted distribution of chromosome aberrations induced *in vitro* at the G_0 (and probably also G_2) stage.

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INFLUENCE OF AGE, SEX AND LIFE STYLE FACTORS ON THE SPONTANEOUS AND RADIATION INDUCED MICRONUCLEI FREQUENCIES

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SUMMARY

Several endpoints have been used for monitoring human populations for environmental or occupational exposure to genotoxic agents, particularly ionizing radiation. The cytokinesis-block micronucleus (MN) assay in peripheral lymphocytes is a reliable method for assessing radiation induced chromosomal damage (DNA breaks and mitotic spindle disturbances) and thus, a suitable dosimeter for estimating in vivo whole body exposures.
To further define the use of this assay in Biological Dosimetry, a study to determine the influence of age, sex and life style factors (smoking habit) on the spontaneous and radiation induced MN frequencies was performed.
The estimation of MN frequencies was analyzed in lymphocytes cultures from 50 healthy donors aged between 4 and 62 years. On the basis of their smoking habit they were divided into 2 groups. A fraction of the sample was irradiated in vitro with γ rays in the range of 0.35 Gy to 4 Gy.
A statistically significant influence on the spontaneous MN frequency was observed ($R^2 = 0.59$) when the variables age and smoking habit were analyzed and also a statistically significant influence on the radiation induced MN frequency was obtained ($R^2 = 0.96$) when dose, age and smoking habit were studied. Sex did not influence MN variability significantly but there was a greater dispersion in the results for females when compared to males, possibly due to the loss of X chromosomes. The comparison of the data from smoking to non smoking donors supports the convenience to take into account the smoking habit for estimating in vivo whole body exposures to γ rays for doses below 2 Gy.

INTRODUCTION

Since Countryman and Heddle (1) first reported the induction of MN in human peripheral lymphocytes, the MN test has been broadly used to detect clastogenic effects (chromosome breaking) and aneugenic effects (effects on mitotic spindle resulting in abnormal segregation of chromosomes after nuclear division) due to chemical and physical (ionizing radiation) agents and thus, a suitable dosimeter for the evaluation and monitoring of genotoxic exposures in vivo and in vitro (2).
Scoring of MN in cytokinesis blocked binucleate peripheral blood lymphocytes is a relatively easy and fast procedure used as an alternative biological dosimeter to the conventional technique of analyzing unstable chromosomal aberrations (dicentrics + rings) for assessing radiation induced chromosomal damage, being particularly useful in the case of large scale radiation accidents.
MN are derived from acentric fragments or whole chromosomes that lag behind at anaphase and failed to become incorporated into either of the two daughter nuclei during cell division. Studies employing inmunofluorescence staining of kinetochores in MN with antikinetochore antibodies (3) have provided evidence that radiation induced MN are predominantly derived from acentric fragments.
After a genotoxic insult, only those cells that divided can express MN. Fenech and Morley (4) have developed a cytokinesis blocked method using cytochalasin B, which enables scoring of MN only in binucleated cells that have completed the first nuclear division (second interphase). Cytochalasin B inhibits cytoplasm division after nuclear division, resulting in binucleated cells.
For the purpose of Biological Dosimetry, the recognition of radiation induced changes in the cytogenetic status requires an adequate knowledge of the spontaneous MN frequency.
The mean spontaneous frequency for a pool of healthy donors previously obtained in our laboratory was 0,013±0,008 (5). This high spontaneous MN frequency shows a large interindividual variability. Some authors suggest that the variability is due to factors such as age, sex and life style conditions (smoking habit).
The objective of our work has been to evaluate the behaviour of MN dosimeter in the low dose range, analyzing the influence of the so called "confounding factors": age, sex and smoking habit on the spontaneous and radiation induced MN frequencies in order to characterize the extent of the application of this dosimeter, considering the confounding factors in the accidental overexposure dose assessment.

MATERIALS AND METHODS

The MN frequency was estimated on 81 peripheral blood lymphocytes cultures from 50 healthy donors ranging between 4 and 62 years, divided into two groups according to their smoking habit; age and sex. A fraction of the blood sample was irradiated in vitro with gamma rays from a Co-60 source (PICKER C4M60) ranging from 0.35Gy to 4Gy, with a mean dose rate of 0.70Gy/min.

Culture method: 0.2 -0.4ml of whole blood was incubated with 8ml of RPMI 1640 medium adding 20% (v/v) foetal calf serum for 72h. at 37°C. Cellular division was stimulated with phytohemagglutinin P (PHA P, 1μg/ml, Difco). After 44h. cytochalasin-B was added to a final concentration of 4.5μg/ml, in order to block cytokinesis. After 72h. of incubation, cells were centrifuged and treated with hipotonic solution (Iskandar method) in order to preserve the cytoplasm. The cells were fixed with methanol/acetic acid (3:1) and stained with 5% Giemsa (pH 6.8).

MN frequency was established evaluating 500 to 2500 binucleated cells per sample and per dose point. The criteria of Fenech (6), were applied for the identification of binucleated cytokinesis blocked cells and MN.

Statistical methods: In order to assess the age, sex and smoking habit influence on the spontaneous and radiation induced MN frequencies, two different evaluation methodologies were applied to the data:

I) Multiple linear regression model on transformed data:

MN frequencies were transformed by the natural logarithm to reduce non linear relationships among the frequencies and to reduce non normality in the data, both assumptions of this analysis.

The linear regression model can be expressed as: $y = \beta_0 + \beta_1 X_1 + \beta_2 X_2 + \beta_3 X_3 + \beta_4 X_4$

II) Poisson regression model. This is an alternative model (7) that seems to apply well to rare events such as MN induction, which follows a Poisson distribution. The Poisson regression model can be expressed as:

$y = e^{\beta 0 + \beta 1 X1 + \beta 2 X2 + \beta 3 X3 + \beta 4 X4}$

being for model I and II:

y = MN Frequency

β_0 = Spontaneous Frequency

β_1 , β_2 , β_3 and β_4 = Estimated Parameters

X_1 = Dose (0Gy - 4Gy)

X_2 = Age (4 - 62 years)

X_3 = Sex (0: Male - 1: Female)

X_4 = Smoking Habit (0: Non Smoker - 1: Smoker)

In order to assess the age influence on the spontaneous MN frequencies, a linear regression was applied.

Calibration curves for smoking and non smoking donors were fitted by applying an iteratively reweighted least square method, where the weight is the variance. The analysis of MN distribution in binucleated cells was carried out using the Papworth "u" test and the variance/mean relationship (σ^2/y). The heterogeneity factor and the variance/mean relationship were applied to correct the coefficient errors of the calibration curves and the Poisson variance associated with the experimental data respectively.

RESULTS AND CONCLUSIONS

Using a linear regression model, the age influence on the spontaneous MN frequency was analyzed, taking the data as a pool and differentiating smokers from non smokers (Fig.I). The results obtained, clearly suggest a significant positive correlation ($p < 0.0001$) of the spontaneous MN frequency with age. Data analysis considering two arbitrarily chosen groups: donors from 25 to 41 years and donors from 42 to 62 years, suggests that the differences observed due to the age on MN frequencies, between both groups, are smaller than the differences observed due to the smoking habit condition and that the smoking habit condition increases three times the frequency of MN with respect to non smoking condition, excluding donors under 25 years in this analysis.

The above mentioned facts justified the study of the dose response relationships for smoking and non smoking donors. The fitting of both curves resulted in a linear quadratic model, according to the following equations:

$y = c + \alpha D + \beta D^2$, where

y = MN frequency for the evaluated dose

D = dose in Gy

Coefficient errors were conveniently corrected taking into account overdispersion.

a) non smokers:

$c = 1{,}14 \cdot 10^{-2} \pm 0{,}14\ 10^{-2}$

$\alpha = (2{,}29 \cdot 10^{-2} \pm 0{,}73 \cdot 10^{-2})$ 1/Gy

$\beta = (3{,}08 \cdot 10^{-2} \pm 0{,}28 \cdot 10^{-2})$ 1/Gy^2

$\chi^2 = 53{,}3$ DF = 24

b) smokers:

$c = 2{,}32 \cdot 10^{-2} \pm 0{,}19 \cdot 10^{-2}$

$\alpha = (3{,}33 \cdot 10^{-2} \pm 0{,}79 \cdot 10^{-2})$ 1/Gy

$\beta = (2{,}28 \cdot 10^{-2} \pm 0{,}45 \cdot 10^{-2})$ 1/Gy^2

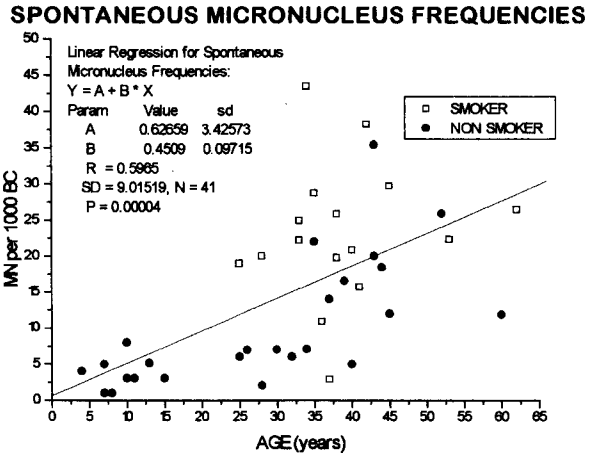
$\chi^2 = 57{,}22$ DF = 27

The analysis of the parameters obtained shows that the calibration curves intersect at the 2Gy dose point and then joint. From a preliminary comparison of the data from smoking and non smoking donors, it seems appropriate to increase the sample for donors ranging from 0Gy to 2Gy in order to evaluate the significance of the difference observed between both curves.

The analysis of MN distribution in cytokinesis blocked cells, both for spontaneous and radiation induced MN was carried out, using the Papworth "u" test and the variance/mean ratio in order to compare with the Poisson

distribution. The observed trend is: while the spontaneous MN frequencies do not differ significantly from the expected Poisson distribution, the radiation induced frequencies are overdistributed. Both, multiple linear regression and Poisson regression models were applied to the data base in order to asses the age, sex and smoking habit influence on the spontaneous and radiation induced MN frequencies. It was observed a significant positive correlation (R^2 =0.59) of the spontaneous MN frequencies (dependent variable) with the age and smoking habit (independent variables). Furthermore, a positive correlation (R^2 = 0.96) of radiation induced MN frequencies with the dose, age and smoking habit was observed too. Multiple linear regression performed better than Poisson regression in the case of spontaneous MN frequencies analysis while Poisson regression resulted more appropriate in the case of radiation induced MN frequencies.
Sex did not influence MN frequencies significantly, but it was observed a wider dispersion of the results for females compared to the one observed on males possibly due to the loss of X chromosomes in elderly females.
These results suggest that the confounding factors: age and smoking habit have a statistically significant influence on the spontaneous and radiation induced MN frequencies, being the smoking habit the strongest confounding factor. Additionally, the difference observed between smoking and non smoking calibration curves supports the convenience of taking into account the smoking habit for estimating in vivo whole body exposures to γ rays for doses below 2Gy.

Fig.I



ACKNOWLEDGEMENTS

This work was supported in part by the International Atomic Energy Agency - Contract N° 302-J1-ARG 8369.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE ESTIMATION OF SOMATIC CELL MUTATION IN YUGOSLAV WORKERS INTERNALLY CONTAMINATED BY RADIONUCLIDES AS A CONSEQUENCE OF CHERNOBYL ACCIDENT

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 (see page 7)

ABSTRACT (See instructions overleaf)

The paper deals with absorbed doses determined on the basis of chromosomal aberrations in lymphocytes of 139 examinees, being residents of Zhlobyn area at the time of Chernobyl accident. They remained continuously there for another at least three months. Chromosomal aberrations in 25 230 cells been scored and by comparison with control subjects, the dicentric plus ring and excess acentric fragment frequencies are higher for these persons who worked in Zhlobyn zone.

Obtained results were compared with results of surface and human contamination monitoring program. Results of urine bioassay and whole body counter measurements are presented as well.

The results obtained proved necessity of defining criteria for further bio-medical surveillance.

ESTIMATION OF CYTOGENETIC RISK IN THE PROCESS OF NON-DESTRUCTIVE TESTING OF WELDS

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INTRODUCTION

In the process of non-destructive testing of welds the workers are very often exposed to both X-ray or gamma radiation and ultrasound , in dependence with the requirements of the test method applied.Despite training courses and awareness of workers that protection is necessary in the USA has been developed a special audit program for assuring radiation safety during radiographic examination operations (1). Such program can be actually implanted in every company which is licensed for weld testing activities. The advantage of this program is also that the number of radiographic inspections is controlled in order to prevent possibility of accidental exposures as the results of fatigue (2).

The estimation of dose based on chromosomal aberration analysis is a relible and generally accepted method, and it indicates genome damages earlier than any other method used in medicine. However, according to available literature data it could be seen that in the cases of overexposure of radiographers detected by film dosimeter only skin changes are quite often diagnosed even without heamatologocal analysis (3). Since no biodosimetrical study so far provides data on genome damages of radiographers caused by combined exposure to gamma irradiation and ultasound the aim of this study was to compare the effects of the exposure to ionizing radiation alone and combined with application of ultrasound during the process of weld testing.

SUBJECTS AND METHODS

Twenty three industrial radiographers working on testing of welds for an average of 10 years were chosen for cytogenetic examination.During radiography examinees used the ^{192} Ir source of ionizing radiation which had activity of 1,85 TBq.In the case of application of ultrasound the frequency ranged from 0,1MHz to 35 MHz.A control group included 20 male subjects from the general population, aged 35-45 years old. Among radiographers and control group those with recent X-ray exposure for diagnostic purposes and drug treatment were excluded from the study. Examined subjects did not exceed permited annual dose of 50 mSv measured by film dosimeter.

The chromosome aberration assay was carried out on the cultures of phytohemagglutinin-stimulated blood lymphocytes. Fixation of cultures and preparation of slides were carried out according to conventional methods (4).Two hundered well-spread and complete metaphases were analyzed for every person and results are presented as percentages.

RESULTS AND DISCUSSION

A group of 23 subjects occupationally exposed to ionizing radiation and ultrasound during the process of non-destructive control of welds was analysed using the cytogenetic method of chromosome aberration assay. The number of structural chromosome aberrations such as dicentric, ring, triradial and tetraradial chromosomes was significantly increased in exposed subjects compared to the control group. Table 1 shows that the frequency of unstable chromosome aberrations such as dicentric, triradial and tetraradial chromosomes are more frequent in cases of combined exposure to ionizing radiation and ultrasound than ionizing radiation alone.This finding corresponding with the data pertaining to occupational exposure to ultrasound emphasizes the importance of acceptaning and considering the ultrasound as a source of radiation with possible health consequences (5).

It can be concluded that in cases of combined occupational exposure estimation of dose received by radiographers using film dosimetry should be accompanied by cytogenetic monitoring because personal dosimeter for ultrasound has not been constructed yet.In order to minimize health risk biomonitoring can detect possible synergistic action of both ultrasound and ionizing radiation which is not measurable by any physical method.

IONIZING RADIATION

Subject	Chromatid break %	chromosome break %	acentric fragment %	dicentric %	tetraradial %	triradial %
1	1.5	0.5		0.5		
2	1.5	0.5	1			
3	0.5	0.5				
4	0.5					
5		0.5				
6	1.5	1.5				
7	1.5	1				
8	0.5	1				
9	0.5	1.5				
10	1	0.5				

IONIZING RADIATION + ULTRASOUND

1	0.5	1.5				
2	1	1.5	0.5	0.5		
3	0.5	1	1	0.5		
4	0.5	1.5	1		0.5	0.5
5	0.5	1	1			
6	0.5	1.5	0.5			
7	0.5	1	0.5			
8		1.5				
9	1	0.5				
10	1	0.5				0.5
11		1				
12	0.5	0.5	0.5	0.5		
13	0.5	2	2	2		
control values	0.8	0.6	0.7			

Table 1. Comparison between chromosome aberrations of radiographers exposed on their working places to ionizing radiation and chromosome aberrations of radiographers exposed to combined action of ionizing radiation and ultrasound

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PLUTONIUM IN HAIR OF RESIDENTS OF BELARUS

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INTRODUCTION

Nearly 3.5% of the reactor core have been released from the destroyed Unit-IV of the Chernobyl NPP into environment, including isotopes of uranium and transuranium elements. Plutonium radionuclides are among them. This part of radioactivity in the form of fuel particles of various dispersity has deposited mainly within the 30-km zone. However, as the investigations have shown, alpha-emitting nuclides of the Chernobyl origin have been found in Poland, Sweden, Bulgaria ands other countries [1]. In the composition of fuel particles plutonium displays low migration mobility. Partially they are in the soluble form and by the metabolism chains find the way into the organizm. The determination of plutonium in the organizms of the residents of the Gomel Region has shown, that its content exceeds the level attributed to the radioactive depositions as a result of the nuclear weapons test in the atmosphere. The internal doses, produced by plutonium account for fraction of per cent of the total irradiation of humans by natural radioactive sources. Inclusion of the possible contribution of transuranium elements into the formation of the distant after-effects should be taken into account, because in the process of the biogenic transformation the release of plutonium from the fuel particles into the mobile forms is possible. It increases the conversion coefficients in the chain soil-organizm and respectively the contribution into the radiation dose [2].

The existing methods for in-vivo determination of plutonium in organizm on the basis of the analysis of urine or blood don't enable the evaluation of this isotope in organizm within the acceptable accuracy, as the obtained values reflect to a considerable extent the transit function of plutonium, the value of which varies considerably depending on its additional ingress into organizm by inhalation or with food.

In recent years hair are used as the test-tissues in solving the sanitary problems. It has been shown, that hair can be both the tracer test-tissues in evaluation of the ecological situation and quantitative indicator of its content in organs and tissues of some microelements and radionuclides [3].

MATERIALS AND METHODS

The collection of hair in hair-dresser's shops of the populated areas of the Gomel Region and Minsk has been carried out. The part of the hair samples has been taken in children from the hospital of the Institute of Radiation Medicine for a check-up. The collected material has been studied at the individual and population levels. The chemical preparation of hair for the analysis has been conducted in agreement with the recommendations of IAEA [4].

The determination of plutonium after radiochemical extraction has been carried out on a alpha-spectrometer Aladin, using plutonium-236,242 isotopes as an internal standard. The method of the solid-state track detector by fission fragments has been also used. The irradiation has been conducted with the neutron flux $2 \cdot 10^{15}$ n/cm^2 on IRT-M reactor.

RESULTS

The first studies on the evaluation of the total content of the fissile radionuclides in hair of residents of populated areas in the Gomel Region (Khojniki, Bragin, Narovlya, Chechersk) have been carried out in 1986 and have shown, that the density of the tracks of the fission fragments is by 2 - 3 orders of magnitude greater, than in hair of the residents of Minsk. However, the impossibility to differentiate the tracks of natural uranium-235 and plutonium-239 has required the improvement of the method of the radiochemical extraction of plutonium-239 from hair [5].

After radiochemical extraction, the analysis with the use of the track method and alpha-spectrometry has shown, that the specific activity of plutonium-239 in hair of the residents of the territories suffering after the accident at the ChNPP is by two and more orders of magnitude greater than the content of these radionuclides in organs and tissues of the residents of the Earth, dependent on the global fall-outs as a result of the nuclear weapon tests, including the plutonium in hair of the workers of plutonium in the plutonium industry equal to 0. mBq/kg [6,7].

Thus, for instance, its content in hair of some individuals has reached 20.0 Bq/kg (the resident of Bragin, M.Z.M., born in 1963). High values of plutonium-239 content in hair of children have been found out (Table 1). The high level of the accumulation of plutonium in the hair of children is explained by the age peculiarities of metabolism of children organizm, which is supported by the experimental studies. In examination of the

distribution of plutonium-239 in animals the primary effect of age on the level of the ingress of plutonium into the fur has been shown: this process proceeds more intensively in young animals [8].

As investigations show, in recent years the reduction of concentration of plutonium in hair has took place. However. it should be noted, that individual high concentrations of plutonium-239 in residents of Bragin is seen systematically ($17.7 \cdot 10^3$ mBq/kg) alongside with the reduction of the average values, including the high concentration of plutonium in hair with relatively low densities of contamination of soils with plutonium (the Vetka town). It needs additional studies.

In order to determine the possibility of using hair as test-tissues for the in-vivo evaluation of the content of plutonium in human organs and tissue, in 1994 the content of this radionuclide in hair, liver, lung and ribs of the residents of the Gomel Region killed in the fatal accidents has been analysed (Table 2).

It follows from the given data, that the average values of the concentration of plutonium in organs and tissues of the residents of the Gomel Region on the average are by an order of magnitude higher in comparison with the residents of the European countries and caused by the global fall-outs. The range of variations between the maximum and minimum values of concentration are the most important. If the ratio between the maximum and minimum values in liver, lung and ribs of the residents of the European countries have been within the limits of 2 - 2.5 times, for the residents of the Gomel region these variations have been higher: for liver, for instance, they have reached 22 and for lung 110. It results from the uneven deposition of plutonium in the areas after the accident at the ChNPP in comparison with its deposition after nuclear weapons tests.

In analysis of the correlation between the content of plutonium in hair and inner organs, the statistically reliable dependence has been determined only between the content of plutonium in hair Ab) and ribs (Ap), which is approximated by the expression $Ap = 0.6 \cdot Ab + 4$. The correlation coefficient is 0.8 with the significance level 0.1 (Fig.).

Proceeding from the established dependence, the concentration of plutonium in skeleton of adult humans with the content in hair $19.4 \cdot 10^3$ mBq /kg) in 1987, has reached $13.6 \cdot 10^3$ mBq/kg. It is 30% of the permissible level of the content of plutonium in the skeleton of professional personnel (P_{CA}). The evaluation of the dependence between these values in children needs revision.

CONCLUSION

On the basis of the obtained results it is evident, that hair can be both the test-tissue in evaluation of the dynamics of the ingress of plutonium into organizm and the quatitative indicator of its content in skeleton. The later allows to organize the in-vivo observation over the process of migration of the radionuclides in the bioshere and its contribution into the internal dose of irradiation upon its ingress into organizm.

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Table 1. Plutonium-239 content (mBq/kg) in hair of children of the Minsk and Gomel Regions (1987).

Minsk					Gomel				
N	Name	Residence	Birth	Pu-239	N	Name	Residence	Birth	Pu-239
1	B.I.	Minsk	1982	$0.30 \cdot 10^3$	1	K.O.	Khojniki	schoolboy	$15.2 \cdot 10^3$
2	N.I.V.	Minsk	1975	$0.42 \cdot 10^3$	2	E.D.*	Khojniki	1980	$12.6 \cdot 10^3$
3	K.N.P.	Minsk	1976	$0.14 \cdot 10^3$	3	Sh.G.P.*	Bragin	1970	$22.9 \cdot 10^3$
4	D.T.V	Minsk	1975	$0.21 \cdot 10^3$	4	Sh.E.I.	Bragin	1970	$16.35 \cdot 10^3$
5	Kh.L.	Minsk	1974	$0.18 \cdot 10^3$	5	K.S.I.	Bragin	1976	$25.6 \cdot 10^3$
6	I.S.	Minsk	1975	$0.5 \cdot 10^3$	6	S.T.M.*	Skuraty- Bragin	1975	$28.0 \cdot 10^3$
7	M.N.	Minsk	1971	$0.5 \cdot 10^3$	7	L.N.	Khojniki	1980	$8.51 \cdot 10^3$
					8	K.L.*	Zhlobin	1982	$1.5 \cdot 10^3$
					9	F.G.*	Borisov.	1975	$4.57 \cdot 10^3$
					10	K.E.S*	Zhlobin	1976	$0.82 \cdot 10^3$
					11	L.S.M.	Gomel	1978	$0.71 \cdot 10^3$

Hair have been collected in the hospital of IRM by E.V. Davydovoj

Table 2. Content of plutonium in organs and hair in various regions of the Europe (mBq/kg / raw material

Country	Liver		Lung		Ribs		Hair	
	aver.	min.-max.	aver.	min.-max.	aver.	min.-max	aver.	min.-max
Finland Great Britain Germany Italy (1959-1984)	18.0	11.0-26.0	1.7	1.0-2.3	3.4	1.5-4.0	-	-
IREP RB (1994)	18.8	6.7-54.0	27.0	1.0-110.0	34.1	11.0-70.0	63.0	15.0-150.0
BSU (1994)	27.4	3.0-51.0	21.6	1.3-42.0	-	-	-	-
RI RM (1992)	45.0	9.0-200	-	-	53.0	17.0-97.0	-	-

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PAPER TITLE PRESENT CHALLENGES IN RETROSPECTIVE DOSIMETRY

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MAJOR SCIENTIFIC TOPIC NUMBER 3.5 (see page 7)

The last decade was noted with significant development of retrospective dosimetry methods. This qualitative and quantitative growth was prompted by attention to problems of dose reconstruction in context of Chernobyl accident. It is expected that an acute need in tools for assessment of individual doses would be met with variety of biological, instrumental, and analytical approaches. Unfortunately, enthusiasm connected with use of methods of retrospective dosimetry is not quite justified. There are still lot of challenges on the way from acceptance of principal applicability of methods to widespread practical reconstruction of individual doses.

Both instrumental and biological methods are endowed with similar difficulties. One of shortcomings is caused by an existence of sensitivity threshold. Another challenge is connected with need of development of adequate methods for conversion of dose determined *in vitro* to dose received by individual. Finally, recent investigation have revealed a great importance of artifacts for correct determination of doses.

Although methods of analytical dose reconstruction do not have a threshold, an accuracy of results depends a lot on different factors. Another typical difficulty in analytical dose reconstruction is connected with lack of primary information. These two problems are inevitably connected and should be treated together.

Another important challenge in field of retrospective dosimetry is problem of combination and assessment of results gained by different methods. Both cross-validation of various techniques and common use of dose assessments achieved by different means will assist in production of reliable values of individual doses applicable for practical use in risk assessment.

RETROSPECTIVE RADIATION DOSIMETRY USING OPTICALLY STIMULATED LUMINESCENCE ON NATURAL AND SYNTHETIC MATERIALS.

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INTRODUCTION.

Optically stimulated luminescence (OSL) techniques especially aimed at using natural materials for retrospective reconstruction of accidental radiation doses in populated areas were developed and studied at Risø as part of an EU research project. Quartz and feldspars separated from building materials, such as bricks and tiles, in addition to porcelain items had their OSL signals measured using different light sources for stimulation to assess radiation doses received by the material. Radiation doses were also evaluated from OSL measured directly on unseparated samples i.e. directly from the surface of brick and tile materials.

APPARATUS AND TECHNIQUES.

The apparatus used for the experimental work were mainly OSL units developed as attachments to the existing automated Risø TL reader and include monochromators for obtaining wavelength resolved luminescence measurements. An automatic OSL scanning instrument was also developed with the aim of being able to perform continuous OSL scanning measurements of brick cross-sections, allowing radiation depth dose profiles to be measured directly.

The basic OSL unit developed contains light sources for both green light and infrared stimulation, enabling measurements of OSL signals from both quartz and feldspar samples. (Bøtter-Jensen and Duller , 1992).

A compact module was developed that allows for the monochromatic illumination of samples in the wavelength range 380 to 1020 nm, enabling the measurement of energy resolved OSL (Bøtter-Jensen et al. 1994 a, Bøtter-Jensen et al. 1994 b). A schematic diagram of the combined OSL attachment is shown in Fig. 1A.

A continuous OSL core scanner system was developed that allows the optical sensors to be moved across either sediment or brick cores. The core is scanned using an excitation slit beam of 1.0 mm x 1 mm which determines the resolution of the system. OSL dose normalization is made either by using short wave UV light or exposing the brick cores to a Cs-137 gamma field and afterwards scanning the OSL sensitivity across the brick profile. A schematic diagram of the OSL scanner system is shown in Fig. 1B.

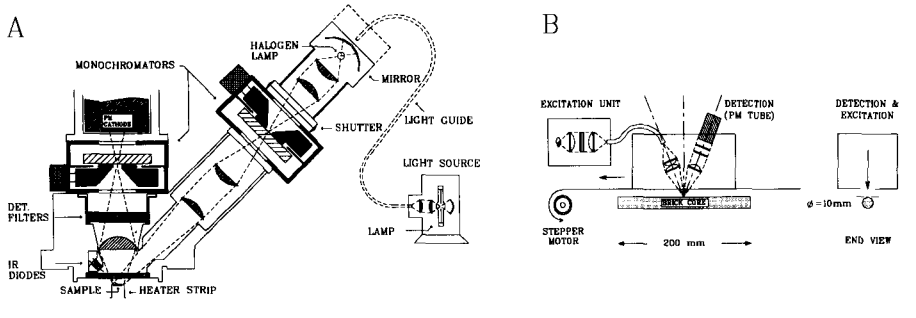

Fig.1.
(A) Schematic diagram of the OSL attachment showing the excitation lamp system with monochromators mounted on both the excitation side and the detection side.
(B) Schematic diagram of the automatic OSL brick core scanning system: the excitation beam is 1 mm wide.

OSL DOSIMETRY CHARACTERISTICS OF QUARTZ.

An attempt to determine the lower detection limit for OSL stimulated with green light on fired quartz was made by obtaining the dose response curves for a variety of quartz samples extracted from different specimens such as bricks, burnt stones and clay. An example of OSL versus dose for a sensitive quartz extracted from burnt clay obtained using the multiple aliquot method is shown in Fig. 2A. As seen, the lowest detectable dose for this material is well below 1 mGy. (Bluszcz and Bøtter-Jensen, 1995).

Quartz grains were extracted from a 40 years old brick and the absorbed dose was determined by GLSL using the additive dose technique. As seen from the dose response curve in Fig. 2B the dose was estimated to be about 200 mGy which is in very good agreement with the expected value based on an annual dose rate of about 5 mGy/y from the environmental radiation and the natural radioactivity in a typical brick. For this particular brick the lower detection limit for an additional accidental dose would be of the order of 20 mGy (10% above the background).

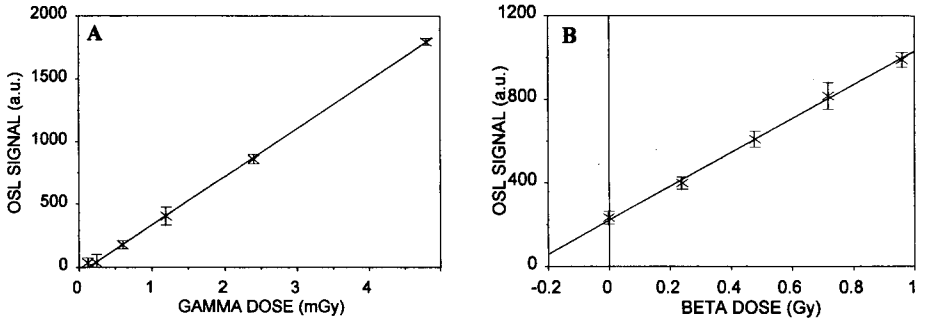

Fig.2.
(A) OSL versus ^{60}Co gamma dose (multiple sample technique) for quartz extracted from a burnt stone.
(B) OSL growth curve (multiple sample technique) for quartz extracted from a 40 years old brick.

OSL DOSIMETRY CHARACTERISTICS OF UNSEPARATED BRICK MATERIALS.

Depth-dose profiles in bricks were determined by measuring the OSL signals directly from the unseparated material across the brick using the automated OSL scanning system (Bøtter-Jensen et al., 1995). Examples of the normalized OSL as function of depth into brick material are shown for a ^{137}Cs irradiated bricks in Fig. 5A and 5 B. The fitted exponential curve also shown, correspond well with the expected attenuation in the brick material.

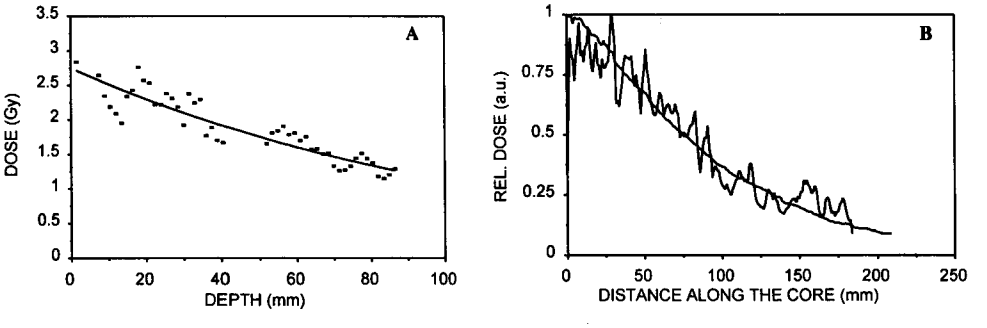

Fig.5.
(A) Dose versus depth into a brick irradiated with ^{137}Cs gamma radiation from one side. Measurements were made using green light OSL on 8 mm x 1 mm slices cut from a core through the brick.
(B) Relative depth dose profile into the same brick from ^{137}Cs gamma radiation exposed from one side and subsequently measured with the automatic OSL core scanner. The attenuation curve calculated by the Monte Carlo code MCNP is shown for comparison.

OSL DOSIMETRY CHARACTERISTICS OF PORCELAIN.

In view of the relevance of using OSL on porcelain in the field of retrospective dosimetry, we studied materials from a collection of mass produced ceramics that we consider would be representative of materials commonly found in many households. (Bøtter-Jensen et al. 1996).

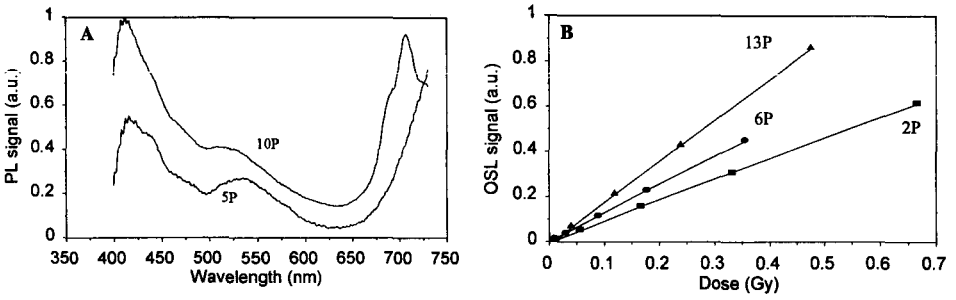

Fig.6.
(A) PL spectra (PL versus wavelength) for two bulk porcelain samples. The emission from Al_2O_3 is clearly demonstrated by the typical peaks at 410 and 700 nm.
(B) OSL versus ^{60}Co gamma dose for three different porcelain samples.

The time stable photoluminescence (PL) emission spectra were recorded with UV stimulation produced by a halogen lamp, filtered with U-340 filters (peak transmission at 340 nm). Fig. 6A shows PL spectra obtained from 2 different porcelain samples. The PL spectra compared with that obtained from Al_2O_3:C TL material, which has been shown to be an extremely sensitive OSL dosemeter material (Poolton et al.,1995).

Measured dose response curves, i.e. OSL versus ^{60}Co gamma dose, are shown for three porcelain samples in Fig 6B. For most porcelain samples the OSL signal increases linearly from 10 mGy up to 20 Gy and shows a further sublinear increase up to at least 200 Gy.

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AN OVERVIEW OF RETROSPECTIVE OCCUPATIONAL DOSIMETRY AT BNFL

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INTRODUCTION

During recent years a key area for occupational dosimetry in BNFL has been the retrospective review of recorded doses, particularly those generated in the 1950s and 1960s when the nuclear industry and many of the associated methods of work were in their infancy. The event which triggered this interest in retrospective dosimetry was the litigation (Reay and Hope versus British Nuclear Fuels plc) which alleged that pre-conception paternal occupational radiation exposure had been a material contributory cause of leukaemia and lymphoma in children. While in the event the cases failed, the detailed investigations carried out in support of the litigation have provided a wealth of knowledge to enable retrospective judgements to be made regarding both the efficiency of the monitoring systems in use in the early years of the nuclear industry and ultimately the validity of the recorded doses. While initially applied to the litigation cases, this knowledge base has subsequently been developed and used for other current applications including the industry-based Compensation Scheme for Radiation Linked Diseases and epidemiological studies into the effects of occupational exposure on the workforce. The paper provides an overview of the main findings in relation to retrospective occupational internal and external dosimetry and the key applications of litigation, Compensation Scheme and epidemiology.

DOSIMETRY REVIEW

A major review of occupational dosimetry systems used at Sellafield since the start of site operations in 1950 has been carried out in BNFL over recent years prompted by the litigation referred to above. The main conclusions from this review have been incorporated into scientific papers either published (1), (2) or in preparation (3), (4) and are summarised below. The implications of the dosimetry review on the applications of litigation, Compensation Scheme and epidemiology are shown in the attached table.

External Radiation Dose

Photon Dose

Film badges of various designs over time have been used for routine personal monitoring and the measured doses recorded in individual's dose records since the start of operations. The primary reason for monitoring was that of control of radiation exposure by comparison with contemporary standards. The review of recorded doses took into account the performance of the various film badge designs used and the characteristics of the radiation fields in which they were worn, along with the dose recording conventions used. The performance of the film badges used is considered to be satisfactory,with the main uncertainties in recorded dose associated with: (a) the limitations of the early film badge designs in use prior to 1963 which, combined with the varied photon energy spectra on the site, would result in positive or negative bias depending on the working area; (b) the general contribution of 'storage dose' from raised background levels on the site while the films are stored between shifts; (c) the proximity of general levels of occupational dose to the detection threshold of the badges, particularly in the 1980s and 1990s; and (d) the procedures used in the event of lost or damaged badges which have varied from not entering dose estimates in the early years to using dose estimates based on the pro rata to the annual dose limit in later years. Overall, the cumulative recorded doses are judged to be within a factor of two of the true dose and are more likely to be an overestimate than an underestimate of dose.

Neutron Dose

The main sources of neutrons on the site were the reactors and the plants processing plutonium extracted from irradiated fuel. The areas of neutron exposure were generally limited and localised by shielding, particularly on the reactors, and while portable neutron detectors were used for operational control, personal neutron dosimetry has been a very difficult technical area (5) because of the complex and varied nature of the neutron fields on the site. It is now recognised that the dose records of some workers on the BNFL sites, particularly at Sellafield, do not fully reflect the likely neutron doses received, with the potential for unrecorded neutron dose arising from:

(a) the lack of availability of a suitable personal neutron dosemeter in the period 1950-59 when plants like the Windscale Piles and the associated plutonium processing plants were in operation and would have been expected to make a significant contribution to the overall external radiation dose of a small part of the workforce,

(b) the poor performance and relatively limited issue of early neutron dosemeters in use from 1959-78, resulting in doubts regarding the validity of the neutron doses recorded for workers on the reactors (the Magnox stations at Calder Hall and the prototype AGR) and the plutonium processing plants operating during this period.

Internal Radiation Dose

Doses from intakes of radioactive materials have generally been much smaller than external doses. However, for some workers internal doses have been significant, mainly from intakes of plutonium and uranium. Unlike external doses, early records for individuals do not generally contain assessed doses, but rather the "raw" results of monitoring, for example activity content of biological samples . Control of exposure was exercised by comparison of the results of monitoring with investigation levels and action levels set directly for these parameters. Conversion to dose was rarely necessary. Retrospective dose assessments are therefore carried out for litigation, compensation, or epidemiological purposes using modern methods and models in conjunction with the recorded results of monitoring and sampling. The principal difficulties experienced that are particular to the early historical data are the poor quality of the measurements, and the lack of detailed and reliable work histories for individuals. The poor quality of the measurements arises both from the comparative lack of refinement of the analysis methods in the 1950s and 1960s, and of the collection methods(6). These in turn led to high limits of detection, and frequent contamination of samples by non metabolised material. The lack of good work histories for individuals leads to difficulty in interpretation of samples since conversion of excretion history to dose relies on assumptions concerning mode and timing of intake. The uncertainties in assessed internal doses are therefore generally much greater than those for external doses.

SUMMARY

Reconstruction of occupational radiation doses in BNFL has been shown to be feasible but has required a significant level of technical resources to cater for the complexity and range of the plants involved and is generally limited in scale of use because of the key requirement for detailed work history information in the dose reconstruction process. The main dose reconstruction work required in BNFL has involved neutron doses for litigation and Compensation Scheme purposes, reflecting the historical difficulties associated with personal neutron dosimetry. Early indications are that the level of reconstructed neutron dose is, in general, small compared to the recorded photon dose.

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TABLE - IMPLICATIONS OF DOSIMETRY REVIEW ON KEY APPLICATIONS

Area of Dosimetry	Litigation	Compensation Scheme	Epidemiology
External radiation:			
Photon dose:	Retrospective adjustments were made to the early film badge results (pre-1963) for the Gardner case-control subjects resulting in generally small changes compared to the recorded doses. The need for individual work histories limits the extent to which this is practicable.	Recorded doses are accepted as generally conservative and as a result a reasonable estimate for Compensation Scheme purposes.	Dose reconstruction is not practicable for cohort studies and is unlikely to be warranted. Further consideration of the uncertainties in recorded photon dose to epidemiological studies is warranted.
Neutron dose:	Reconstruction of neutron dose was carried out for Gardner case-control subjects with the assessed doses generally small compared to the recorded photon dose. Reconstruction represented a major technical programme requiring detailed knowledge of historical plants including the assessment of work place neutron doses through the interpretation of historical neutron measurements and the identification of worker occupancy patterns.	Reconstruction of neutron doses is being carried out for individual claims for Sellafield workers . The reconstruction process uses a set of neutron dose protocols prepared for each plant and incorporating the main features identified in the litigation section. Twelve protocols have been prepared and are now being applied; they require specific work history information for the individual and are time consuming to apply. The assessments are generally conservative and early indications are that the level of neutron dose is small compared to the recorded photon dose.	Further consideration of the implications of unrecorded neutron dose is needed but in general the level of unrecorded dose is relatively small and is likely to be of limited consequence to cohort studies involving BNFL dose records. Reconstruction of neutron dose for the Sellafield cohort is not practicable and unlikely to be warranted. Reconstruction of neutron dose for the limited numbers involved in case-control studies is feasible with development of the Compensation Scheme neutron dose protocols to remove the main areas of conservatism.
Internal radiation:			
Internal dose:	Reconstruction of the doses for the Gardner subjects was made using conservative assumptions but the doses were small compared with external dose.	Internal doses are estimated from available monitoring data using conservative assumptions to favour the claimant under a jointly agreed protocol.	Owing to the complexity and cost of internal dose assessment, and the limit to which assessments can be automated, doses are not readily amenable to cohort studies. Overall, some progress has been made recently to use internal dose data by providing confidence levels to the assessments made.

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PAPER TITLE NEUTRON MEASUREMENTS AT LARGE DISTANCES FROM THE U. S. ARMY PULSE RADIATION FACILITY BARE CRITICAL ASSEMBLY

AUTHOR(S) NAME(S) P. Goldhagen, F. Hajnal, M. Reginatto, P. Shebell, G. Klemic, N. Azziz (U. S. Department of Energy); C. Heimbach, M. A. Oliver, M. B. Stanka (U. S. Army Pulse Radiation Facility)

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MAJOR SCIENTIFIC TOPIC NUMBER 3.5... (see page 7)

ABSTRACT (See instructions overleaf)

Most of our knowledge about the long-term effects of radiation on people comes from studies of the Japanese atomic bomb survivors. At distances over 1 km from the Hiroshima detonation, transport calculations predict a far lower fluence of thermal neutrons than that determined from activation measurements. To help resolve this discrepancy, we have performed neutron measurements at various distances from the bare critical assembly at the U. S. Army Pulse Radiation Facility for comparison with long-range transport calculations.

Neutron fluences and energy spectra were obtained using a multisphere neutron spectrometer with 12 detectors, each of which contains a BF_3-filled pulse ionization chamber. Measurements were made at distances of 715, 1084, and 1588 m from the critical assembly when the ground was very wet and at distances of 300, 1588, and 1986 m when the ground was relatively dry. Gamma-ray energy spectrum and dose-rate measurements were also made.

The shape of the neutron spectrum did not change much with distance. The ratio of thermals to fast neutrons was larger when the ground was wet. The falloff of the thermal neutron fluence with distance matches that for the thermal neutrons at Hiroshima determined from activation measurements. This suggests that some higher-energy fission-like neutrons may have escaped from the weapon. If the Hiroshima survivors were exposed to significant numbers of fast neutrons, and not just to gamma rays, estimates of the health risks from exposure to all types of radiation would be affected.

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PAPER TITLE Thyroid dose reconstruction for the evacuees from Pripjat

AUTHOR(S) NAME(S)
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ABSTRACT (See instructions overleaf)

Thyroids of many Ukrainian people were highly exposured due to the Chernobyl accident. Because of this the problem of dose reconstruction is very important for different groups of population and especially for evacuated people.

49,360 inhabitants of Pripjat were evacuated shortly after the Chernobyl accident (most of them within 35-37 hours after the accident). During the evacuation they often passed through highly contaminated territories. According to the results of wide scale public survey (more than 10,000 questionnaire data about behaviour after the accident), only about 50% of evacuees left contaminated areas within 5 days and 30% of them stayed there for more than 30 days. Thyroid doses were estimated for the group of evacuees who were measured on the ^{131}I content in thyroid. Their individual intake functions were derived from the information about route of evacuation. The collective doses and radiation risks for thyroid cancer were assessed for different age groups on the base of these astimations.

The method for the individual thyroid dose assessment was also developed for the group of evacuees from Pripjat who were not measured. For this purpose the empirical equation was used to describe the correlation between calculated doses and behaviour parameters (place of residence in Pripjat, consumption of stable iodine and the age at the time of the accident).

STOCHASTIC MODELS FOR INDIVIDUAL EXTERNAL DOSE RECONSTRUCTION OF POPULATION EVACUATED AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

As a result of the Chernobyl accident, the residents of Pripjat and other settlements of the 30-km zone were affected by substantial amounts of radiation with the potential of delivering them significant radiation exposures before their evacuation. A reconstruction of individual doses of evacuees, including the determination of the uncertainty of the results, is of interest for epidemiological investigations. Dose rate measurements conducted in the time span between accident and evacuation and individual behavior data obtained by a questionnaire survey involving more than 30,000 evacuees allow for a reconstruction of individual external doses of a large fraction of the evacuated population. A first assessment has been reported in (1). The results revealed a large variation range for doses of different individuals, showing the importance of performing individual dose reconstructions.

This paper reports on developments and results of a second stage of individual external dose reconstruction. In particular, stochastic models based on Monte Carlo methods (2) were developed in order to determine uncertainty distributions of calculated doses from an assessment of the uncertainty of the data.

INDIVIDUAL DOSE RECONSTRUCTION FOR EVACUEES FROM PRIPJAT

In the city of Pripjat, dose rate measurements started two hours after the accident and lasted until after the completion of the evacuation two days later. They were conducted by two independent teams in intervals of 3.5 hours on the average. Figure 1 shows the position of the measurement points and dose rates measured 30 hours after the accident. Information on individual pre-evacuation histories of about 16,000 evacuees from Pripjat was obtained by a questionnaire survey (1). As indicated in Figure 1, the city was divided into eight sectors according to the measured open air doses and the evacuees were asked to reconstruct their location (indoor or outdoor, and in which sector) hour by hour until their evacuation.

The stochastic models developed for calculation of individual external doses are designed with a modular structure, separating the calculation of parameters which do not depend on individual behavior from the final computation of the individual dose. The uncertainty of the parameters are represented by frequency distributions of the parameter values. They are obtained from an assessment of data uncertainty by Monte Carlo sampling methods coded in the program PRISM (3). Taking the parameter values, the external dose for each individual is repeatedly (for example, 500 times) calculated, obtaining a frequency distribution representing its uncertainty. The individual external effective dose distribution E is calculated according to

$$E = C \cdot \sum_{h=1}^{h_{evac}} D(h,s) \cdot \Delta t \cdot L(h) \qquad \text{with } \Delta t = 1 \text{ hour.}$$

The summation extends over each hour h to the time h_{evac} the individual was evacuated; C is the conversion factor from absorbed dose in air to effective dose. The D(h,s) are uncertainty

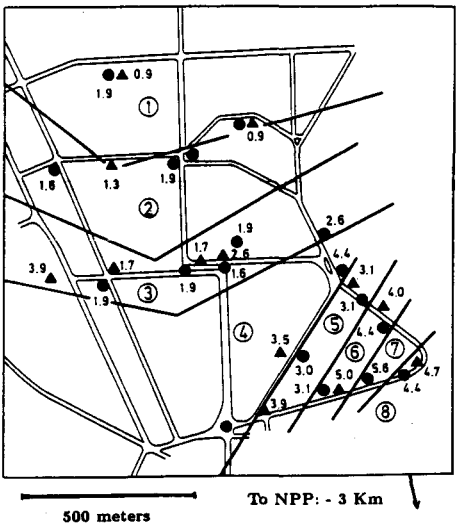

Figure 1. Plan of the city of Pripjat showing the points where dose rate measurements were performed by two independent teams (triangles respectively circles). Also indicated are dose rates in mGy/h measured 30 hours after the accident and the subdivision of the city into sectors for the questionnaire survey.

distributions of dose rate relative to reference area for the hour h and the sector s in which the individual was staying at this hour; the L(h) are location (shielding) factors appropriate for the indoor or outdoor location at which the individual had been at the hour h.

The distributions D(h,s) of dose rate relative to reference ara describe the uncertainty of the average dose rate of the sector and the variability of the dose rate within the sector. They are determined from the dose rate measurement data combined with information on the environment of each measurement point and an assessment of the uncertainties due to measurement errors and modifications by the environment. Incorporating time interpolations and Kriging methods (4) for interpolations in space in the PRISM code, the set of dose rate distributions D(h,s) is obtained. Figure 2 shows the distribution resulting 30 hours after the accident for one of the sectors of Pripjat.

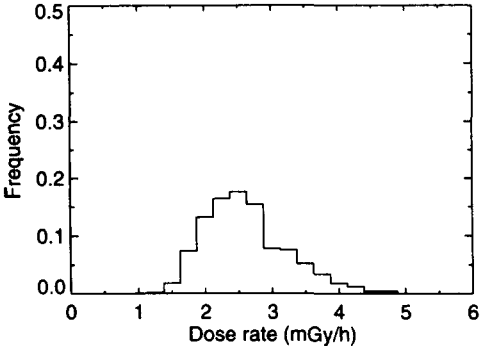

Figure 2. Frequency distribution of dose rate relative to reference area in sector 3 of Pripjat (compare Figure 1), 30 hours after the accident. The distribution reflects the uncertainty of the average dose rate of the sector and the variability of dose rate within the sector.

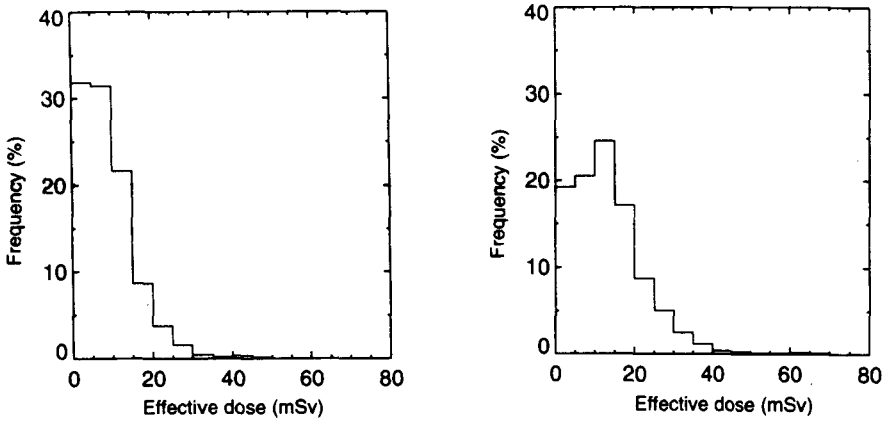

Figure 3. Relative frequency distribution of individual external effective doses of children (age group 3-7 years, left side) and adults (age group 25-55 years, right side) evacuated from Pripjat.

Location factors, defined as the ratio of dose rate at a particular indoor or outdoor location to the dose rate at a reference area (5), were calculated by Monte Carlo photon transport simulations for a specific five storey housing block typical for Pripjat. The results of separate calculations for different source configurations and energies provide a data base for estimating location factors and their uncertainty ranges. From a first analysis, for Pripjat a location factor of 0.05 with a range of 0.01 to 0.2 was estimated for indoors and of 0.8 for outdoor locations.

Based on the distributions of dose rates relative to reference area and the estimated location factors, distributions of individual external effective doses of 12,653 evacuees from Pripjat are being calculated. In a first step, deterministic calculations were performed with this data, obtaining a mean external effective dose for this population of 11 mSv, which is only slightly lower than the value of 11.5 mSv reported in (1). Frequency distributions of individual external doses for children and adults are compared in Figure 3. Children had been staying a larger fraction of time indoors, so that their external doses are generally lower.

CONCLUSIONS

The development of stochastic models which allow to determine uncertainty distributions of external effective dose individually has been discussed for population evacuated from Pripjat. Similar models are being developed for evacuees from the other settlements of the 30-km zone. The results will allow to obtain a clearer vision of values and uncertainty ranges of individual doses received by population from the near zone of the Chernobyl nuclear power plant before its evacuation.

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Vienna, Austria

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PAPER TITLE Radiation Doses to Population in the Western Districts of the Bryansk Region Following the Chernobyl Accident

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MAJOR SCIENTIFIC TOPIC NUMBER 3.5 ..... (see page 7)

ABSTRACT (See instructions overleaf)

In the Russian Federation the areas most affected by the Chernobyl accident are the western districts of the Bryansk Region. At the time of the accident 112 thousands residents happened to live in the 'strict control zone' (SCZ with caesium-137 contamination in excess of 0.55 MBq/sq.m). Over 250 thousands of whole body measurements on caesium radionuclides contents were carried out in the SCZ and at some settlements in adjacent territory, including 150 thousands made in the first two years after the accident. 10 thousands individual measurements of exposure to external radiation were made with TLD in samples of people from settlements inhabited by 90% of the total population of the SCZ. Due to results obtained from a limited set of reliable measurements of iodine-131 in the thyroid gland made in May 1986, a method of thyroid dose reconstruction was developed. The method was employed to assess the distribution of individual thyroid doses at residents of the SCZ. With the aim of effective use of available information to reconstruct individual doses, a data bank is under development in the Institute of Radiation Hygiene. This data bank includes primary records on radiometric examinations of people and on questionnaire surveys, environmental and social information relevant to exposure conditions. Some results of individual dose reconstruction for population in the SCZ as well as examples of distributions of personal doses (individual doses ascribed to identified persons) are presented in the paper. Discussion is given on methodology of personal dose reconstruction.

RETROSPECTIVE EVALUATION OF EXTERNAL COMPONENT OF INDIVIDUAL DOSES FOR TECHA RIVERSIDE RESIDENTS

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INTRODUCTION

The discharges of liquid wastes into the Techa river (1949-1956) by Mayak, first Soviet industrial nuclear facility should be listed among the most significant waste accidents. The Techa river belongs to Iset-Tobol-Irtysh-Ob river system. During only period 1949-1951, 76×10^6 m^3 of liquid radioactive wastes were discharged, with total activity of 100 PBq (1). Internal exposure of the Techa riverside inhabitants was caused mainly by the incorporation of Sr-90. The values of the dose due to Sr-90 can be estimated on the basis of the measurements of body burdens using the whole body counter. There is no possibility to determine experimentally the individual contribution of external exposure by the Techa river. The average value of this contribution for each community was estimated on the basis of the measurements of gamma dose rate along the banks of the river, on the shore within a few hundred meters of the water in specific areas of villages, and inside certain houses. Validation of these estimations and their comparison with individual data are also of considerable importance.

Recently, a method was found for individual dose assessment using tooth enamel and *Electron Paramagnetic Resonance* (EPR) spectroscopy for dosimetry. The method is based on the measurement of the concentration of radiation induced radicals CO_2^- in hydroxyapatite which is the mineral component of tooth tissues and bones. This paper presents the first preliminary results of the use of EPR method for retrospective evaluation of external component of individual doses for Techa riverside residents.

SAMPLES AND METHODS

Totally 28,000 people resided on the Techa banks during the period of massive releases. About 4,500 of them were living in the upper Techa, up to 65 km downstream of the site of radioactive release. These people received mainly external dose of irradiation from the Techa river as it was confirmed by measurements with whole-body counter (2). The residents of the middle and lower Techa (78 - 237 km) received the doses of internal irradiation mainly due to Sr-90. Twenty-nine teeth from 27 Techa riverside inhabitants were measured using EPR method. Five of them were from upper Techa. Collected teeth were extracted in accordance with medical indications. All Techa riverside inhabitants studied were also investigated using whole-body counter for Sr-90 developed at Urals Research Center for Radiation Medicine (URCRM) (3).

An important component of the method of EPR dose reconstruction is the procedure of sample preparation. The details of this procedure and experimental parameters of EPR measurements were already given in a few previous publications (4,5).

The dose reconstruction was done by the additive dose method with unweighted linear least-square-fit employing the Levenberg-Marquart algorithm. Six-five measurements, one original and four after additional irradiation in steps of 300 mGy each were used for the linear fit.

Since 1974 the large scale investigations of Techa riverside residents with body counting system SICH-9.1 (the Russian acronym) were started. The content of Sr-90 in the skeleton is determined with this equipment. The detailed description of SICH-9.1 and procedure of measurements was given in (3).

RESULTS AND DISCUSSION

The results of EPR reconstruction (doses absorbed in tooth enamel) for the Techa riverside inhabitants versus the distance between the place of permanent residence during 1949-1956 and the site of release are shown in Fig. 1. The dote line in this figure is the fit of experimental data by Gaussian, whereas the solid line is site-specific average dose from gamma emitters at the same distance from the point of release. These average external doses were taken from (2).

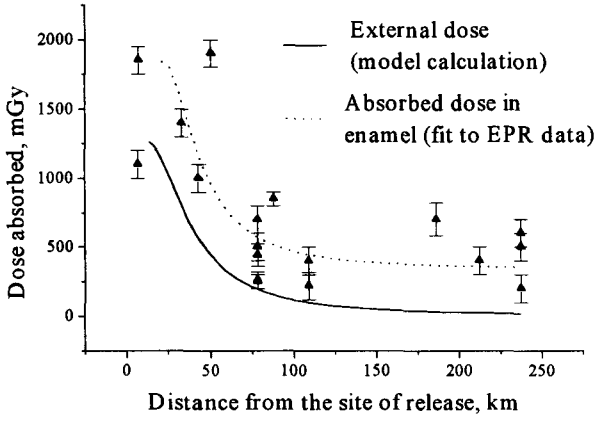

Figure 1. Correlation of absorbed dose in tooth enamel measured by EPR and distance between the place of residence and the site of release.

As it is seen from Fig. 1, the shape of experimental and calculated dose-distance curves is similar, however, results of EPR reconstruction are always higher calculated doses from gamma emitters on 300-500 mGy. The latter can be explained by the necessity to take into account the contributions to dose absorbed from the sources other than the Techa river. Another important peculiarity of Fig. 1: there is no dependence of dose on distances more than 78 km from site of release. The village of Muslyumovo is located at the distance of 78 km and according to our previous evaluations the internal component of exposure begins to play a main role downstream of this point.

In the present rough analyses of doses received by the residents of the Techa communities we have selected three groups of sources contributed to the total dose: 1) external exposure from the Techa river and it's banks; 2) internal exposure due to incorporation of radionuclides ingested with river water and local food-stuffs; and 3) other sources of exposure (natural background, medical procedures, radon, global and local fallouts etc.). The population of the upper Techa received the most essential dose contribution from external exposure, because the exposure dose rates near the river during the period of massive releases were extremely high (2). For the residents of the middle and lower Techa internal exposure mainly due to Sr-90 was predominant. The annual background dose could be taken from our results of EPR reconstruction for the 86 residents of the town of Kamensk-Uralskii (5). The linear

approximation of dose-age dependence gives 4.0 mGy as annual dose rate for the Urals region. The value obtained allows also to explain the excess of results of EPR reconstruction for the South Ural nuclear workers in comparison with data of individual monitoring cited in (4). The excess dose for workers (about 200 mGy) is close to the average EPR dose for Ozyorsk residents (4) and may be interpreted as background level.

As noted above, the internal exposure of the Urals population occurred due to the incorporation of mainly Sr-90. The tooth enamel is irradiated by Sr-90 incorporated in tooth dentine and enamel itself, since path length of beta particles from Sr-90 - Y-90 in bone tissue is about 1 mm. Hence the dose absorbed by tooth enamel depends on strontium metabolism in dentine and enamel. Tooth enamel is the most mineralized part of the skeleton and the metabolism in the enamel for adults is extremely slow. Therefore whole-body counter data have to reflect Sr-90 content in dentine which determine the dose reconstructed by EPR.

The suggestion that EPR doses for the residents of the middle and lower Techa indicate mostly internal exposure is confirmed by correlation between the results of EPR reconstruction and data of Sr-90 body burden. The linear regression (Y=a+bX) of this correlation gives a=260 mGy and b=20 mGy/kBq. The value of 260 mGy is very close to the dose registered for the Kamensk-Uralskii residents of the same age (5), which shows the reasonableness of the choice of the last group of people for the assessment of background level. The coefficient of linear regression between the value of Sr-90 body burden and dose absorbed by tooth enamel (20 mGy/kBq) allows to estimate the Sr-90 contribution for the residents of the upper Techa. The results of calculations of Sr-90 and background (annual dose rate 4 mGy) contributions are summarised in Table 1. The difference between sum of Sr-90 and background contributions and result of EPR reconstruction is due to external dose of the Techa river. As seen from Table 1 the contribution of background and internal doses to the total value varies from 15% to 75%, so it is significant part of the total dose in tooth enamel for the upper Techa. This table also shows the comparison between individual external doses and average external doses for the settlements where these persons lived.

Table 1.

Per. n°	Age, year	Sr-90 body burden, kBq	Total dose absorbed (EPR), mGy	Background component, mGy	Internal component, mGy	External component, mGy	Average external dose, mGy
1	51	5.0	1100	200	100	800	1300
2	79	5.1	1900	320	100	1480	1300
3	80	1.7	1430	320	35	1075	900
4	65	22.5	1000	260	500	240	600
5	65	0.9	1850	260	20	1570	430

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PAPER TITLE
The technique of EPR-dosimetry with tooth enamel: achievements and problems.

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MAJOR SCIENTIFIC TOPIC NUMBER 3.5. (see page 7)

ABSTRACT (See instructions overleaf)

Usually in the EPR-dosimetry technique ones proceed from the assumption of the linear relation between intensity of radiation-induced EPR signal and the value of the absorbed dose. Such the assumption is true only in some cases since in the general case both the value of the absorbed dose and the intensity of the EPR signal are a sum of few components. The dose accumulated by a tooth D_{accum} is a sum of such components: the accidental component; component due to environmental radiation and the contribution of roentgen procedures. The measured intensity of the EPR signal from radiation-induced centers consists of the following items: I_{accum}, depending on the accumulated dose; caused by the exposure to ultraviolet light and caused by mechanic deformations. Only the relation between I_{accum} and D_{accum} has the linear form with a proportionality coefficient depending on geometry and energy of the incident radiation.

Apart from errors in the determination of items described above, some additional error may occurs at the secondary exposure of enamel while the additive dose method is used. This inexactitude is effected through next factors. Firstly, the radiation sensitivity of the fine-grained enamel sample may differ from the sensitivity of the whole tooth, and secondary, the geometries of irradiation of tooth *in vivo* and tooth enamel particles under laboratory conditions are also different.

The present paper addresses a thorough investigation of the above-mentioned aspects of the EPR-dosimetry technique with tooth enamel.

ALKALINE-EARTH METAL DITHIONATES-BASED SPIN-RESONANCE β -RADIATION DOSIMETERS

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Our previous studies have revealed that stable paramagnetic centres (PMC) classified as ion-radicals SO_3^- and characterized by the EPR isotropic signal (ΔH~5 G, g=2,0036) are formed on γ-irradiating the alkaline-earth metal dithionates of a general formula $MeS_2O_6 \cdot xH_2O$, where M is the alkaline-earth metal (1). Further studies gave us the reason to suggest barium dithionate ($BaS_2O_6 \cdot 2H_2O$) as a γ-dosimeter in the range of 10^{-1} to 10^4 Gy (2).

The study of ionizing radiation effect on the alkaline-earth metal dithionates has revealed also that this class of substances is not only γ-sensitive but β-sensitive as well. The literature on the subject cite only a few chemical compounds and materials used for the purposes of electron radiation dosimetry (3-6), e.g. L-alanine with the operating measurement range of 10^2 to $2 \cdot 10^5$ Gy (6). This paper describes a problem of dithionates application for the purposes of spin-resonance electron radiation dosimetry.

Barium and strontium dithionates were chosen as samples. The purity was tested by means of X-ray phase analysis, EPR technique and IR-spectroscopy. The preground salt powders were subjected to external and incorporated β-irradiation. External β-irradiation was carried out in the air at 20°C by the electron flow (10^{11} to 10^{14} cm^{-2}, E=4 MeV) and $^{90}Sr+^{90}Y$ sources of 10^4 and 10^5 Bq activity. Internal β-irradiation was performed by ^{90}Sr radionuclides of 10^3 and 10^4 Bq activity introduced into the substances structure during their recrystallization with $^{90}Sr(NO_3)_2$.

Ion-radicals SO_3^- were exclusively stabilized in the result of external β-irradiation of alkaline-earth metal dithionates by monoenergetic flows (E=4 MeV) and $^{90}Sr+^{90}Y$ radionuclides as well as in the case of γ-irradiation. The primary stage of homolytic decay of the excited $S_2O_6^{2-}$-anion S-S bond was believed to be the reason of radiation-induced SO_3^- formation. And this very, the so-

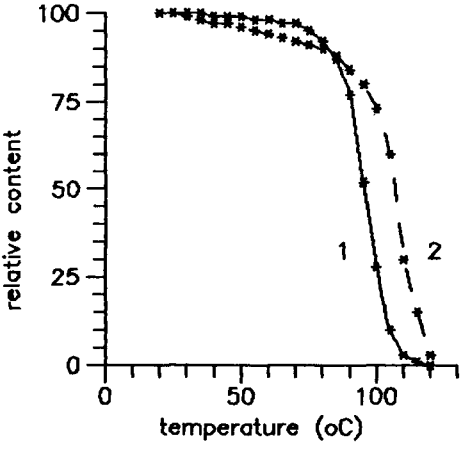

Figure 1. Crystallization water (1) and ion-radicals SO_3^- (2) content (%) vs temperature for $BaS_2O_6 \cdot 2H_2O$

called predissociative decay which essence is the preceding excitation energy migration and concentration on a particular bond, was suggested (7) for some (O-O, C-C, C-H) bonds, S-S bonds including.

Our studies have revealed that the process of stabilization of radiation-induced SO_3^- in the structure of dithionates was affected by the crystallization water. These PMCs were found to be very stable at 20°C. Nevertheless, even insignificant sample heating caused gradual recombination. Figure 1 (curve 2) shows the plot of SO_3^- relative concentration in $BaS_2O_6 \cdot 2H_2O$ structure versus temperature. This Figure shows insignificant annealing of radicals at about 90°C and their intensive recombination in the crystallization water removal region (curve 1). Noteworthy is that both processes terminate practically at a time. Besides, impossibility of dithionate original structure recovery by annealing SO_3^- was noticable. These observations had led to the conclusion that the processes of radicals recombination and crystallization water removal are interrelated. Such interrelation was also found out for radical particles stabilized in another non-organic systems, e.g. mixed crystals of $CaCO_3$ and $CaHPO_4 \cdot 2H_2O$ (8).

The comparison of external γ- and β-radiation effects reveals that the values of EPR signal intensity when exposing barium dithionate to equal doses of both kinds of radiation SO_3^- PMCs agree within the measurement accuracy (20%, 1σ) (Fig. 2). The intensity of EPR signal of

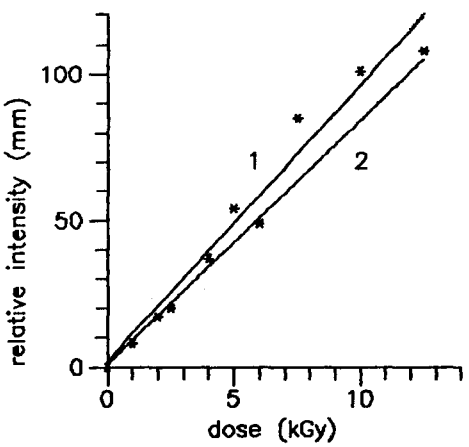

Figure 2. EPR signal relative intensity vs barium dithionate external gamma- (1) and beta-radiation (2)

SO_3^- was always a linear function both of β-radiation dose and exposure time for various activities of $^{90}Sr+^{90}Y$ sources used. Naturally, the angle of a straight line with the abscissa was changing. For example, the change of the source activity by one order (from 10^4 to 10^5 Bq) caused the increase of radiation-induced SO_3^- accumulation rate by about 7 or 8 time (Fig. 3).

The experimental data available made it possible now to preliminary estimate the external β-radiation absorbed dose by comparing the EPR signal intensities for γ- and β-radiation cases. Hence, the alkaline-earth metal dithionates may become promising spin-resonance dosimeters of β-radiation.

At present, large difficulties are experienced in β-radiation dose measurements for the case of incorporated radionuclides. We have attempted to compare the doses due to external electron and

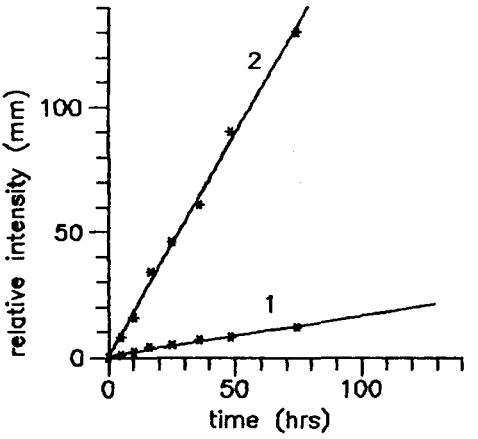

Figure 3. EPR signal relative intensity vs time. External β-irradiation of $BaS_2O_6 \cdot 2H_2O$ by $^{90}Sr+^{90}Y$ radionuclides of 10^4 Bq (1) and 10^5 Bq (2)

incorporated radiations. The intensity of a corresponding EPR signal was a working hypothesis when comparing the doses. When investigating crystal $BaS_2O_6 \cdot 2H_2O$ doped by $^{90}Sr+^{90}Y$ in the air at 20°C, the EPR signal was detected; this was a signal of radicals SO_3^- and its intensity was increasing linearly in time. Consequently, as it was the case of external β-radiation, it became possible to estimate the value of the absorbed dose of internal radiation by the very same method of compariny the number of radiation-induced PMCs respective to the applied dose of external γ- and internal β-radiation.

Thus the submitted material points out the below described advantages of alkaline-earth metal dithionates used as spin-resonance β-radiation dosimeters. Such advantages are high sensitivity to external and incorporated β-radiation, the narrow isotropic EPR signal for radiation-induced SO_3^-, large interval of linear accumulation, the PMC stability and discovered unsusceptibility to solar light. The instability at high temperature may be considered as a disadvantage of dosimeters described. Nevertheless, as it has been already pointed out, the ion-radicals SO_3^- are very stable at room temperature. The study of EPR signal intensity upon time has revealed that the intensity value is practically stable at temperatures below 40°C for at least 2 years (with ±10% accuracy (2σ)). This feature may provide for the lasting storage of information by sealed dithionate dosimeters.

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SYNTHETIC HYDROXYAPATITES AS A POSSIBLE EPR DOSIMETER

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INTRODUCTION

Considerable interest to investigations of irradiation effects in synthetic hydroxyapatite, $Ca_{10}(PO_4)_2(OH)_6$, by electron paramagnetic resonance (EPR) (see, for example, (1-4)) is provided mainly by two reasons. Firstly, natural hydroxyapatite as a mineral component of tooth enamel, dentine and bones could be used for retrospective dosimetry. Therefore the knowledge of origin of different components of its EPR spectrum is important for development of the correct procedure of dose reconstruction. Secondly, increasing application of EPR dosimetry with alanin asks for the search of possible alternatives. From this point of view, such parameters of tooth enamel as threshold dose (100-200 mGy), time (10^7 years) and thermal (80° C) stability, sensitivity to irradiation by alpha-, beta-, gamma-, x-rays, UV and heavy ions allow to consider hydroxyapatite as a possible EPR dosimeter. EPR dose reconstruction and EPR dosimetry are based on the measurement of the concentration of radiation induced radicals in hydroxyapatite. Above radicals in hydroxyapatite originate from the carbonate impurity. Therefore the concentration of carbonate plays an important role in EPR dosimetry with teeth and bones and determines radiation sensitivity of hydroxyapatites. In the present contribution we are reporting some preliminary results of EPR study of irradiated hydroxiapatites synthesised at various pH and with different carbonate content. The parameters of this compound as EPR dosimeter are discussed. On the base of results obtained a hypothesis about origin of background signal in EPR tooth enamel spectrum and variation of its radiation sensitivity are proposed.

MATERIALS AND METHODS

Synthetic apatites with different carbonate content were prepared by hydrolyzing monetite, $CaHPO_4$, in $NaCO_3$ solutions at 95° C, as it was described in (1). The carbonate concentration in apatite was regulated by variation of $NaCO_3$ content in solution. Three different concentrations of $NaCO_3$ were used in this investigation - 0.01, 0.1 and 0.25 mol/l. The conditions of chemical synthesis were also modified by variation of pH. Three different pH every is higher than 9 were tested. Totally nine different samples were investigated in this study. All samples have been investigated with X-ray diffraction, which showed sharp and well resolved peaks characteristic of well-crystallized apatite.

Three different types of irradiation sources were used for irradiation in this investigation, namely: medical ^{60}Co source (Eldorado), X-ray tube (60 kV, 40 mA) and UV lamp (NK 6/12 Heraeus). This UV lamp is low-pressure mercury lamp with power of 6 W and major spectral line 254 nm.

EPR measurements were made at room temperature with an X-band Bruker ESP 300 spectrometer with 100 kHz modulation and with microwave power 12 mW.

RESULTS AND DISCUSSION

There are two types of components in EPR spectrum of both biological and synthetic carbonate hydroxyapatite. The first type signals correspond to spectrum of non-irradiated hydroxyapatite. Second type signals are induced by irradiation.

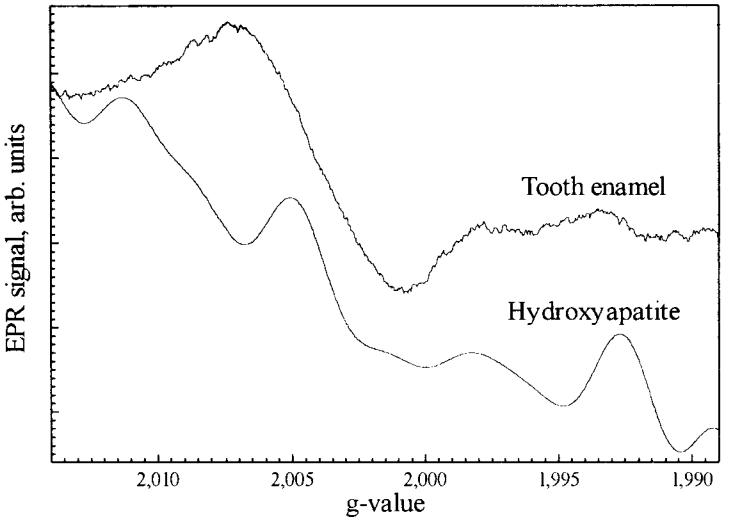

Fig. 1. EPR spectra of tooth enamel and synthetic hydroxyapatite obtained with 0.01 M of $NaCO_3$.

Among first type EPR signals in tooth enamel so-call background component (ΔH=0.9-1.0 mT, g=2.005) can be selected. The origin of this signal is not exactly established. Existence of this signal essentially limits a threshold of retrospective dosimetry with tooth enamel. Our comparative analysis of EPR spectra of the synthesized samples and irradiated tooth enamel has shown that amplitude and structure of background signal depends on carbonate concentration and pH during synthesis. Fig. 1 demonstrates EPR spectra of non-irradiated synthetic hydroxyapatite (0.1 M of Na_2CO_3) and tooth enamel. In spite of some difference in details the structure and parameters of background signals in these samples are very close. Our investigations of synthetic apatites with different concentrations of carbonate allow to suggest connection of these background signals with existence of other impurity carbonate phases in compound obtained. Most typically is $CaCO_3$. Such conclusion was also confirmed by X-ray diffraction. Possibility of synthesis of impurity carbonate phases are connected with pH in solution. Therefore, in considerable degree the individual variation of the radiation sensitivity and amplitudes of background signal in tooth enamel are caused by individual value of pH in the saliva. The latter allows to suggest, firstly, not so wide variation of radiation sensitivity inside a group of the people with similar diet and, secondly, strong dependence of this parameter on distortion of pH at some problems with metabolism.

As known (1,2), four different signals (caused by four different radicals) were found after irradiation: i) isotropic signal with g_x=2.0115, ΔB=0.33 mT (CO_3^- at phosphate lattice site); ii) anisotropic signal with g_x=2.00170 g_y=2.00084, g_z=2.0060, ΔB=0.34 mT (surface CO_3^-); iii) isotropic signal with g=2.0006, ΔB=0.30 mT (CO_2^- at phosphate lattice site); iv) anisotropic signal with g_x=2.0030 g_y=2.0015, g_z=1.9970, ΔB=0.18 mT (surface CO_2^-) of so far minor

significance. Our investigations were shown, X-ray and UV irradiations mainly produce the isotropic signals (i) and (iii), and Co-60 irradiation produces the anisotropic signal (iv). The observation of different EPR spectra in synthetic apatites after irradiation by Co-60, X-rays and UV allows, in principal, to distinguish between the these radiation types. In case of Co-60 irradiation a lower detection threshold of about 1 Gy was found so far; it could, probably be improved by changing the conditions of chemical synthesis and increasing the carbonate content. For X-rays and UV radiation the sensitivity is considerably higher. The existence of a non-stable line in the spectrum of synthetic hydroxyapatite after X-ray and UV irradiation can be used to estimate the time elapsed between irradiation and measurement.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
(Instructions for preparation on reverse)

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Abstract No.	
Receipt	Eingang-TAK
Author	23. Aug. 1995
Acceptance	
Mini-Presentation	

PAPER TITLE Development of antiradiation preparation for radiotherapy and regions of radiation accidents.

AUTHOR(S) NAME(S) O.N.Chupakhin, L.N.Rasina

SUBMITTING AUTHOR

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3... (see page 7) 6.3

ABSTRACT (See instructions overleaf)

Search for biological active substances for preventive treatment and therapy of radiation influences is still actual. The results of development of preparation for protection of normal tissues on condition of irradiation cancroid's therapy and human organism in the regions of radiation accidents and radionuclides polluted areas are presented. For this purpose effectiveness of synthetized antiradiation compounds- derivates of phenilacetophenon, dithiocarbomates and ethylene diamine, antitoxix preparation with adaptogenic properties and antibacterial preparation was studied. These compounds are synthesized on the base of the structure analogues of nature compounds. The famous radioprotector- cystamin- effective on sharp irradiation conditions was used for comparing. BALB-line mouses were subjected to chronic gamma-irradiation with doze power 20 Gr during 60 days. Preparation were infused intra peritoneum or per orum every day during the first 30 days. Among them we found the effective preparations which protect BALB-line mouses from acute irradiation and had prophylaxis and therapy action on condition of chronic external gamma-irradiation. This preparation increase average life duration on condition of acute irradiation; on condition of chronic irradiation they are slow down the development of leycopenia, reduced lipid peroxide oxidation(LPO),r-protein level and distant citogenetic effects, bringing it close to norm. Animals external appearence,activity and mass normalization seven mouths later after the experiment.

NON-IONIZING AND IONIZING DOSIMETRY IN A SPACE RADIATION ENVIRONMENT WITH GaAs and SiC LEDs

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ABSTRACT

This paper describes a dosimetry experiment that will be carried onboard the Russian MIR space station. The experiment will compare the ionizing and Non-Ionizing Energy Loss (NIEL) in semiconductors of the radiation encountered in space. The ionizing dose will be detected using ThermoLuminescent Dosimeter (TLD) whereas SiC and GaAs LEDs will be used to measure the non-ionizing component. The tray will be mounted on the outside of the station for a minimum period of 4 months. The goal of the experiment is to determine the feasibility of using SiC and GaAs LEDs as NIEL dosimeters in space.

INTRODUCTION

Devices fabricated with GaAs and other compound semiconductors are expected to play an increasingly important role in space electronics. Satellites in space are exposed to the radiation belts, solar flares, and cosmic radiation. In contrast to silicon devices, the radiation damage to GaAs devices is caused by displacement of atoms from their crystal lattice sites by incident energetic particles. Burke et al. (1), Summers et al. (2), and Barry et al. (3) have shown that atomic displacement damage is proportional to the NIEL of the radiation in the semiconductor. In order to assess the sensitivity of compound semiconductor devices to space radiation, it is of great interest to develop a practical NIEL dosimeter. A second objective of the mission is to investigate a possible method of discriminating between radiation damage due to electrons and heavy particles.

EXPERIMENT DESCRIPTION

The monitor package will be installed on an aluminum tray measuring 300 mm × 75 mm. Three different experiments will be fitted on the tray. The NIEL experiment will occupy 68 mm × 75 mm and consists of i) 20 GaAs LEDs; ii) 10 SiC LEDs; iii) 13 locations for TLD 700s of varying thickness; iv) 4 pieces of 10^{15} n-doped; and v) 4 pieces of 10^{16} n-doped GaAs wafer. Figure 1 shows a schematic diagram showing the relative position of the various TLD 700s (B_1-B_4, D_1-D_3, F_1-F_3 and I_1-I_3), the SiC LEDs (A_1-A_5 and J_1-J_5), the GaAs LEDs (C_1-C_5, E_1-E_5, G_1-G_5, and H_1-H_5) while positions X_1, X_3, X_5, and X_7 and X_2, X_4, X_6, and X_8 are the 10^{15} and 10^{16} n-doped GaAs wafers, respectively.

In order to probe different energy regions of the radiation field (i.e. different points on the dose depth curve), the package is divided in 4 compartments: #1 in which all the devices, TLDs, SiC and GaAs LEDs and GaAs wafers are directly exposed to the radiation field; #2 in which all devices are covered by protected 13μm thick Kapton sheet; #3 is covered by a 51 μm steel sheet under a protected 51 μm thick Kapton sheet; and #4 in which all devices are covered by a 3 mm thick Tantalum absorber, itself covered by two different thicknesses (13 and 51 μm) of virgin Kapton sheet.

PRE-FLIGHT QUALIFICATION TESTS

The lens and caps of the GaAs and SiC LEDs (which were manufactured by Telefunken and CREE Research, respectively) were removed in order to directly exposed the chips to the radiation

environment and avoid any shielding due to these components. However, in order to stabilize the electrical characteristics and to protect them from Atomic Oxygen (AO), they were nitrided. Sample LEDs and TLDs were exposed to AO for 120 minutes (corresponding to approximately 3 months in low earth orbit) using a Large area Microwave Plasma (LMP) facility with an O_2-SF_6 mixture excited in the radio-frequency mode. The package will be exposed to temperature extremes in space; therefore, the sample LEDs and TLDs were subjected to repeated temperature cycling between liquid nitrogen and 115°C. The tests indicated that the samples suffered no apparent deterioration.

NIEL MONITORS

The use of LEDs as NIEL monitors was developed by Barry et al. (4) and their response over a wide energy range has been established (3). The NIEL is proportional to the change in the inverse of the minority carrier lifetime (5), which is deduced from the frequency response of the LED prior to and after exposure. However, due to the relatively short time the experiment will be in space, only a small dose is expected.

In order to maximize the precision for the small NIEL dose anticipated, a set of 25 GaAs LEDs (from 200) and 15 SiC LEDs (from 100) were selected for optimum repeatability and minimum standard deviation of their measured lifetime. The temperature sensitivity has been carefully measured and a correction factor is applied to the data by the software. A group of these LEDs are being stored as "standard" for checking any drift in the lifetime measuring equipment.

For low energy particles, a significant correction to the measured NIEL, due to the energy loss in the surface layer of the LED, is required. Thus the thicknesses of the epitaxial layers were determined from the threshold in the variation of the change in the minority carrier lifetime with proton energy (5).

The epitaxial top layer of the GaAs LEDs was measured to be 30 ± 2 microns. Using TRIM-92 code, the energy loss of a proton as it travels through the epitaxial layer, steel, Kapton, and Tantalum is estimated. With this information, the sensitivity of each compartment is determined: > 1.9 ± 0.1 MeV for Compartment #1; > 2.2 ± 0.1 MeV for Compartment #2; > 4.9 ± 0.2 MeV for Compartment #3; > 53 MeV for Compartment #4. Thus LEDs in Compartment #4 will be used as "background" monitors and compared to the series of LEDs kept in the laboratory.

GaAs WAFERS

The damage (the Gallium vacancy V_{ga} and Silicon at the Arsenic site Si_{As}) due to room temperature irradiation with 0.6 to 200 MeV protons anneals out to a large part at 550°C, while that due to 7 MeV electrons anneals to a much less extent (6). The purpose of the n-doped GaAs wafers is to investigate the use of this phenomenon to differentiate between electron and proton components in the space radiation.

CONCLUSION

Dosimeters for measuring the total and the NIEL of radiation in space has been shipped to the Ukrainian Space Agency and will be installed on the MIR space station for a minimum period of 4 months probably in early 1996. The post-flight analysis of the data will be carried out as soon as the tray is returned to the ground. A dose depth curve for the NIEL will be obtained from the analysis.

ACKNOWLEDGEMENT

The authors would like to thank Prof. R.C. Tennyson, Director of the University of Toronto Institute of Aerospace Studies for many useful discussions and for facilitating the flight of this flight. The work is supported by Space Systems Directorate, Canadian Space Agency, Contract # 9F028-5-5162/01-XSD.

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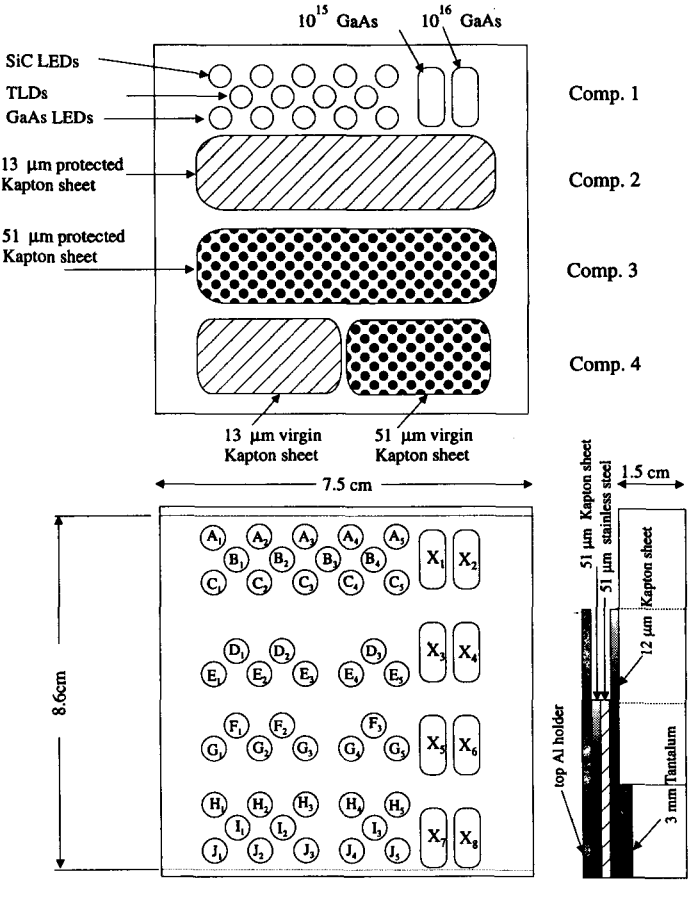

Figure 1 Schematic diagram of the NIEL experiment showing the four compartments of GaAs LEDs and wafers, SiC LEDs, and TLDs

DETERMINATION OF ^{210}Pb LOW ACTIVITY BY GAMMA SPECTROMETRY

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INTRODUCTION

The extensive studies aimed at finding out the dose-effect relationship in the area with increased in-door radon content and thus with supposed increased lung cancer incidence risk are carried out in the Czech Republic. The important part of such studies is to estimate the long term dose from radon and its short-lived progeny to lungs for each followed person. Usually, measurements of the activity of ^{222}Rn and its progeny in the in-door air of houses and the extrapolation to the past levels are the base for dose estimation There is also the possibility to assess the intake of short lived radon progeny deposited in the lungs by the in-vivo measurement of long lived radon progeny ^{210}Pb originating by their decay and then transported to other parts of the body and deposited mainly in bones. Annual average intake of ^{210}Pb is estimated to be 44 Bq which results in effective dose equivalent of 0.12 mSv (1). There is an effort to determine the amount of ^{210}Pb both in-vivo and in environmental samples by the detection of its 46.5 keV gamma rays. With regard to the efficiency of measurement the determination of ^{210}Pb deposited in bone surfaces of the skull seems to be the best solution for in-vivo measurement (2). The in-vivo determination is very difficult due to low yield of its gamma rays and also due to their low energy. In addition the activities which can be expected in the human body even with long term inhalation of high activities of radon and its short lived daughters are very low. At present time, the most serious problem is the sensitivity of instruments used. Even if this problem would be solved, the interpretation of activity of ^{210}Pb measured directly in-vivo in skull will be quite complicated because of direct intake of ^{210}Pb through inhalation and ingestion. The monitoring of amount of ^{210}Pb entering the body via direct ingestion and inhalation in areas with increased radon levels and comparison with areas considered as control, i.e. with average levels, is necessary for proper discrimination of individual components of ^{210}Pb retention. Therefore, in addition to the basic problem - calibration of special detectors for the measurement of ^{210}Pb in-vivo - an effort was undertaken to determine ^{210}Pb content in out-door air and particle size distribution of aerosols with attached ^{210}Pb, the content of ^{210}Pb in fallout and in drinking, mineral and table water. During the study the most suitable detector system was sought.

METHOD

The measurements of ^{210}Pb require special equipment with very low background. The equipment consists of special detectors (n type HPGe detectors, HPGe well detectors, HPGe planar detectors, LEGe detectors, special scintillation detectors - phoswichs). Very good shielding is substantial. In the NRPI there are 6 HPGe detectors available for the measurement of ^{210}Pb. Three of them are reverse electrodes detectors with thin Be window (n-type detectors). One well-type detector was used in the study. For the special case of measurement of low energy gamma emitters two LEGe (low energy Ge) detectors with beryllium and carbon-epoxy windows are used. The LEGe detector is a large area detector with a thin boron-implanted outer contact for high sensitivity at low energies. Because of device geometry, the capacitance is low, and this results in excellent resolution at low to moderate energies. The parameters of the detectors are summarized in Table 1. For comparison of capabilities of different detection systems also one p-type coaxial HPGe detector was included in the study. In addition to semiconductor detectors the possibility to use scintillation detectors was studied. Two scintillation NaI(Tl) detectors were used, one 5in x 4in with thin Al window (SC1) and one thin NaI(Tl) detector of 7 mm thickness with Be window (SC2).

RESULTS AND DISCUSSION

Sensitivity of different detection systems was studied. Because of low gamma energy of ^{210}Pb (46.5 keV) and low intensity of this energy line (4.05%), detector with high detection efficiency in this energy region and with low background is suitable. It was found, that all detectors with beryllium windows have approximately the same ^{210}Pb peak in their background, regardless of their detection efficiency (size), so that the source of ^{210}Pb is very near to the detector. As the lowest background in ^{210}Pb peak area was found with well detector (which endcap is from very thin aluminium with no beryllium), it was most probable that the source of ^{210}Pb is in beryllium window. The results of background measurements in the region of interest are summarized for semiconductor detectors in Table 2. There was the attempt to decrease the background by using additional shielding materials in close geometry e.g. copper, tungsten, steel, however with no significant effect. The

effort to decrease minimum detectable activity for ^{210}Pb resulted in complete reconstruction of endcap parts of LEGe detector No. 13, in which the beryllium window was replaced by the window from carbon-epoxy. The background count in the 46 keV peak area decreased approximately 10 times. This conclusion is very important for purchasing HPGe detectors when user has a.o. the aim to detect low activity of ^{210}Pb. The LEGe detector No. 20 was already purchased in special low-background arrangement.

Table 1 Parameters of semiconductor HPGe detectors used for ^{210}Pb measurement

Detector No.	Manufacturer	Type	Rel. efficiency 1333 keV [%]	FWHM 1333keV[keV]	FWHM 46.5keV [keV]	diameter [mm]	length [mm]
13	Canberra	LEGe	-	-	0.60	50.50	15.00
20	Canberra	LEGe	-	-	0.60	50.50	20.00
16	EG&G Ortec	n	38.30	2.00	1.00	59.80	61.60
53	EG&G Ortec	n	20.50	1.82	1.40	49.50	51.90
103	Canberra	n	15.50	1.68	1.10	46.50	39.50
93	Canberra	well	15.80	1.99	1.40	50.00	59.00
43	EG&G Ortec	p	25.00	1.85	1.30	52.90	60.60

Table 2 Summary of background in the region of 46.5 keV and minimum detectable activity for ^{210}Pb

Detector No.	Count rate in 46.5 keV peak [cps]	Integral count rate in the 46.5 keV region [cps]	Efficiency for planar source of ^{210}Pb	MDA [Bq]
13	0.00670	0.0104	0.151	0.20
13a[1]	0.00065	0.0026	0.151	0.10
20	0.00044	0.0031	0.137	0.12
16	0.00670	0.0273	0.159	0.30
53	0.00530	0.0220	0.106	0.40
103	0.00650	0.0099	0.113	0.33
93	0.00130	0.0146	0.381[2]	0.09
43	0.00130	0.0100	0.016	1.80
SC1	-	0.4000	0.169	1.20
SC2	-	0.1250	0.007	1.85

[1] after reconstruction of endcap

[2] efficiency for source inside well

Minimum detectable activity for different types of semiconductor and for scintillation detectors was calculated for typical geometry used for measurement of environmental samples, i.e. planar source with diameter of 5.2 cm. The results for 100000 s measurement time and for 5 % risk of error type I and II are included in Table 2. The determination concentration of ^{210}Pb in out-door air was the first attempt to estimate the amount of ^{210}Pb in various components of environment (3). High volume aerosol samplers (with air throughput from 60 $m^3.h^{-1}$ to 200 $m^3.h^{-1}$) were used for the aerosol collection altogether in four different places. Aerosol filters (PC-S, produced in SLZ, Hnúšťa, Slovakia), on which aerosol was collected one week, were measured without any treatment by semiconductor gamma spectrometry. Filters were measured in close geometry, using measuring time at least 100000 s. The average concentration of ^{210}Pb in the aerosol in out-door air in Prague for the whole sampling period 9 years (1986-1994) was found 540 $\mu Bq/m^3$. This value is in accordance with values given by UNSCEAR Report for the same geographical latitude. Similar values were found on other places in Czech republic, too. The ratio of activity of short lived progeny of ^{222}Rn and to the activity of radon itself to the activity of ^{210}Pb is approximately one order of magnitude higher in in-door air that in out-door air. It means that in-door aerosol with ^{210}Pb has much shorter mean time of life than the one in out-door air. There is higher

is higher probability of attachment of the particles on the walls and surfaces in-door than out-door. The ventilation of the room could influence ^{210}Pb content in in-door air, too.
Particle size distribution of aerosols with attached ^{210}Pb on their surfaces was studied with the use of 6-stage cascade impactor (Sierra Andersen, Model 236) attached to a high-volume air sampler Sierra Misco with a flow control system (Model Sierra Misco 5000). Flow-rate 0.565 $m^3.min^{-1}$ was used. Slotted fibreglass filter, sucked with solution of silicone grease to reduce re-entrainment and bounces of aerosol particles was placed on each stage. After the 6th stage the back-up filter for catching the rest of the particles was placed. Samplings with cascade impactor were performed both out-door and in-door. Main part of ^{210}Pb is attached both in out-door and in-door air to the small aerosol with aerodynamic diameter under 0.4 μm. Particle size distribution does not differ significantly from aerosol particle size distribution of short lived radon progeny. No difference in aerosol particle size distribution for the out-door and in-door air was found.
The yearly intake of ^{210}Pb by inhalation from out-door air is about 4 Bq (in accordance with UNSCEAR value). In the houses with elevated radon concentration, yearly intake by inhalation is from 70 to 700 Bq.
Table and mineral waters could be important contributors to the ingestion intake especially if their source is in places with high amount of uranium series radionuclides in the bedrocks. Water samples were measured in native form, after evaporation of 1 to 10 litres of water and after radiochemical separation of ^{210}Pb . Altogether 54 samples of potable, mineral and table water including one sample of pilsner beer were measured. The volume activity of ^{210}Pb ranged from detection limit (approx. 0.003 Bq/l) to 1.4 Bq/l. Majority of samples had the volume activity of ^{210}Pb below 0.1 Bq/l. Activity of ^{210}Pb in potable water was from less than 0.005 Bq/l to 1.4 Bq/l, activity in 16 samples of mineral and table water was lower with maximum of 0.2 Bq/l. Assuming the yearly consumption of potable water to be about 770 l for an average adult, the maximum intake of ^{210}Pb in exceptional cases could be about 1kBq which results in the dose of 1.3 mSv. The average yearly consumption of mineral and table water is for an adult about 14 l so the dose from ingestion of ^{210}Pb in this water is negligible.
A semiconductor LEGe detectors No.13 before and after reconstruction and No. 20 were used for in-vivo determination of ^{210}Pb in skull of people. Assumed distribution of ^{210}Pb on the skull bone surfaces was simulated by planar source moved on outer and inner surfaces of the skull. . The minimum detectable activity of ^{210}Pb in skull was determined for 5400 s measuring time to be about 20 Bq , i.e. about 130 Bq in total skeleton.

CONCLUSION

The study proved that the high resolution semiconductor gamma spectrometry is promising method for determination of ^{210}Pb in various kinds of environmental samples. Its sensitivity can be further improved using concentration and separation radiochemical methods for sample pretreatment. However, the most important field of application of semiconductor gamma spectrometry in measurement of ^{210}Pb is in-vivo determination of its amount in human body. For the detection of ^{210}Pb in-vivo in human skull further effort to decrease background in the 46.5 keV region is needed. The sensitivity can also be increased using geometry for simultaneous measurement with several LEGe detectors. Interpretation of measured values of ^{210}Pb in-vivo needs also the correction for enhanced direct intake of ^{210}Pb by inhalation in radon houses, which is enabled by the sampling and measurement of in-door air.

ACKNOWLEDGEMENT

This study was partly supported by research contract No. 7716/RB "The Particle Size Distribution of ^{210}Pb in the Air" as a part of IAEA Co-ordinated Programme: Radon in the Human Environment: Instrumentation, Modelling, Dosimetry and Survey.

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THE THEORY OF INDIVIDUAL VARIABILITY OF OSTEOTROPIC RADIONUCLIDES METABOLISM.

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The greatest radiation accidents - Kyshtym (in the Urals), on the Three Miles Island, in Chernobyl led to increased accumulation of artificial radionuclides in organisms of tens of thousands of people, and also in animals. Unfortunately, new similar and even more dangerous accidents are possible. Health condition examination and controlling of such great population contingent are unlikely possible. The only rational way is to reveal and to analyze groups of the increased risk. Individual dose overloading or prognosis of dose overloading from the internal radiation is one of criteria of these groups. Obviously, revealing of individuals with the increased defeat possibility has great medical , social, agricultural and economic importance.

Osteotropic radionuclides are of specific danger for human being long time after accident. This is connected first with such fact that most of them have a long half-life period and secondly, they can be selectively accumulated in the skeleton, being tightly bound with bones, Reaction to the inner irradiation by incorporated radionuclides depends on their kinetics that defines the dose load besides of individual radio sensitivity.

Despite the fact that thousands of works and dozens of mathematical models [5] are devoted to osteotropic radionuclides metabolism only isolated ones promise satisfactory reconstruction of individual absorbed doses in future and no one leads to their individual prediction.

The aim of this paper is to study mechanisms of individual peculiarities of osteotropic radionuclides skeletal metabolism, and to develop approach for dose value prediction. Some regulations explaining the importance of skeleton in radionuclides metabolism and showing the direction of search of its quantitative regularities are taken as a basis of these theory. The first, there are three evolution trends: universal significance of calcium in the cells in the row beginning from bacteria to highest organisms and increase of its regulator role in the multycellular on the level of organism; ability of calcium salts to increase mechanical strength of supporting tissues; growing connection of metabolic and supporting function. The second, bone is involved in metabolism of radionuclides as a structurally functional wholeness. The third, metabolic way of radionuclide from blood into bone consists of a number of stages. At each stage a life of radionuclide depends on its physical chemistry properties (alkaline earth, actinides and so on). The stages can be combined with respect to time and place. Radionuclides in blood are in ionized or colloid condition, they are partly fixed with proteins, blood cells, bioligands. At the same time they are mixed in circulation and are carried through capillary walls. Permeability is higher in the form of ions, high-dispersed colloids, complex compounds which are fixed neither with proteins nor with form elements of blood. "Deponing agent" of citrate type, breaking away radionuclid from proteins and cells and promoting permeability of capillary wall into extravascular space of a bone (20 $mm^3/gr.$) with its connective tissue and osteogene cells in which actinides are deposited appreciably, was found in bone capillaries. The next stage is bone surfaces having a number of local peculiarities. The processes of ion exchange, of adsorption and chemical combining, of diffusion are taking place here. Isotope is moving forward deep into bone along canaliculi-lacunar net. When in bone radionuclide is subjected to translocation, walling, resorption together with reorganized bone tissue. Their removal out of bone occur in the result of diffusion, desorption from the surface and cell resorption. Recirculation leads to new deponing. The fourth, radionuclides metabolism in skeleton is defined by 10 limiting morpho-physiological factors (LMPF). LMPF are physiological processes, physical chemistry reactions, biochemical substrata and hystological structures participating in the metabolism of radionuclides in skeleton tissues.

As a whole they are an integrated system which is necessary and sufficient for the complete description of radionuclides exchange in vertebrates skeletons. The influence of each of them is independent of the influence of others. They were distinguished from "innumerable" quantity of outward and inward factors influencing metabolism of radionuclides in skeleton. LMPF is: F_1 - blood circulating through skeleton; F_2 - ability to transcapillarity transfer; F_3 - deponing agent; F_4 -competitor role of intensity of radionuclide accumulation in soft tissues and its removal together with excreta; F_5 - desorbtional ability of the bone surface unit; F_6 - total area of skeleton; F7 - resorption; F_8 - growth leading to walling radionuclide being deposited on the surface and

accompanied by its local dislocation; F_9 - exchange inside of osteogenic cellular elements; F_{10} - surfacevolume correlation (effectivity of removing depends on a quantity of radionuclide deponed on the surface and in the volume of a bone). An individual variability can concern each of LMPF. Principle possibility of data using an individual kinetics of osteotropic radionuclides in dose loading retrospective evaluation and prediction has been studied by means of mathematical model (Fig. 1).

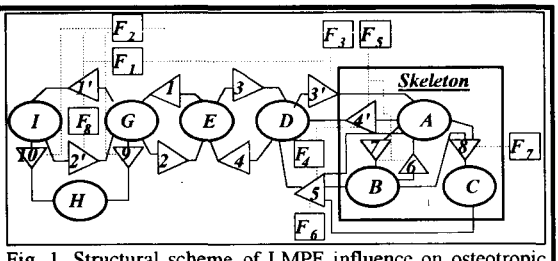

Fig. 1. Structural scheme of LMPF influence on osteotropic radionuclides metabolism.

Circles - chambers (A - bone surface; B - non-walled up; C - walled up volume; D - extravascular liquid of bone; E - blood; I - extravascular liquid of soft tissues; G - soft tissues; H - excreta. Triangles (arrows) - communications. Rectangles, F - LMPF.

Intercommunication of LMPF and their integral effect are represented in this model as an eight chamber model, in which the meaning of communication constants is defined by LMPF. A big number of chambers is justified by the fact that each of them reflects physiological reality. Thus chamber of extravascular space of bone and soft tissues introduced in the process of work turned out to be extremely significant. They allowed to describe processes of desorption taking place on the bone surface and in soft tissues more fully, that led to refusal from correction coefficients. The model is described by the system consisting of 8 linear differential equations having constant coefficients. In order to quantitatively characterize LMPF their parameters "a_1" were introduced. We illustrate specified approaches with some examples.

KO3 -is a velocity of transition out of extravascular space of a bone onto a bone surface which is defined by its sorptional ability and which is 88 times as high as permeability of capillary membrane KO3=58291.

K6,K7 - are the velocities of transition out of non-walled up volume onto the bone surface and from the bone surface into non-walled up volume, they are accordingly defined using the curves of washing off and accumulation of ^{90}Sr out of bone fragments.

Numerical values of parameters of LMPF of rats, mice, dogs, fish and men were defined. Qualitative agreement of radionuclides metabolism curves depending on their physical-chemical properties, species and physiological condition of organism (man, norm and Pedget illness) with real indexes was got with the help of this model.

An example of the results of study of individual variability of metabolism of osteotropic ^{91}Y and LMPF in the experiment and with the model is shown in the Fig. 2. The following parameters of LMPF were got: ability for permeability, part (0,16; 0,10; 0,22); skeleton surface area, cm^2 (569, 545, 731); desorptional ability, relative units(0,008; 0,017; 0,010); resorption, part/hour (0,015; 0,0009; 0,0009); surface-volume of a bone ratio, cm^{-1} (69;62;49). Average-species meanings of three other LMPFs are used in the model: blood-circulating, part (0,115), deponing agent, part (0,2), intensity of soft tissues metabolism and excretion, relative units (0,1). Satisfactory alignment of experimental points (vital registration) and calculated meanings of curves of removal is noticed.

In order to study the distribution of dose loading of internal β-radiationthe method of TLD-dosimetry was used (research are carried out in cooperation with dr.T.Betenekova, USTU-UPI). The distribution of ^{90}Sr in dog's and in rat's skeletons was measured TLD-dosimetric detectors of Al_2O_3 were used. TLD-indexes of absorbed dose for the unit of time (relative units) are shown in Fig. 3. You can see that ^{90}Sr is distributed along bone surfaces of a skull extremely irregularly (beginning from 6,4 relative-units in back part of a head till 22,6 relative units in the region of eye-sockets). Indications of TLD in the teeth (in enamel, dentine, cement) were also measured. They call for a separate discussion. Besides physiological aspects of metabolism of tooth, tissues the interest in tooth as to "an individual dosimeter" is very considerable. Interpretation of ESR-signal of enamel allows to evaluate adequately an absorbed dose of γ-radiation. However, this method is not adapted for β-radiation. β-source, including ^{90}Sr, in the result of peculiar structure, composition and physiology of different

reflected in the value of absorbed dose. Quantitative revealing of patterns of manifestation of these specific features for ^{90}Sr must become a bases for transition from absorbed dose in tooth tissues to the doses on the cells of skeleton surface, red bone marrow and inside the whole body.

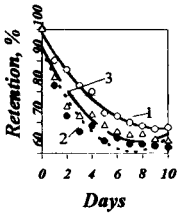

Fig. 2. Curves of retention of ^{91}Y with different individual parameters of LMPF in rats.

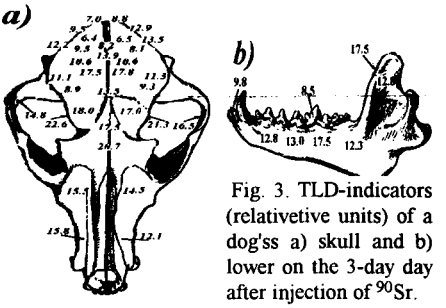

Fig. 3. TLD-indicators (relativetive units) of a dog'ss a) skull and b) lower on the 3-day day after injection of ^{90}Sr.

Specific features of radionuclide metabolism of teeth in our mathematical model are reflected in the following way: radionuclide comes from extravascular space of tooth pulp through the layer of odontoblasts to dentine surface and then to dentine canaliculi. In the framework of this model these stages are identical to extravascular space, bone surface, and bone volume. Radionuclide diffuses into enamel from dentine canaliculi which is described as a bone volume. Radionuclide also come into enamel from saliva (additional chamber and communicational constants: blood-saliva-enamel). Radionuclide moves along dentine canaliculi (analogy with bone volume) and from pericementary capillary net (in such a way as on the bone surfaces) into cement of tooth's root. An important peculiarity of tooth tissues is shown in removal of radionuclides deponed in teeth: normally they are subjected to remodeling very slightly (cement, dentine). Hence the most valuable in dosimetric relation quality of teeth: the most part of dentine and cement and practically all the enamel, being formed during the period of generation, are kept during the whole life, while any other area of bone is replaced repeatedly. Removal of radionuclides out of tooth tissue is made practically only by diffusion-desorption. Parameters of all LMPF of a tooth are quite accessed to quantitative definition and are identified now. Of course, not only kinetic data but steriometric relations between radiating tissues are taken into consideration when defining absorbed dose in tooth tissues with the help of ESR-method. So, dose loading is provided not so much by deponed isotope in it, as by the adjacent teeth and soft tissues, saliva and radionuclides of ration. Thus, the description of processes of radionuclides deponing in tooth tissues requires definite corrections in a structural scheme of a model and evaluations of parameters of appropriate chambers and communication constants.

Thus it is shown that individual peculiarities of kinetics of radionuclides in bones reflect quantitative differences of parameters of several limiting morphological and physiological factors (LMPF). This theory allows to hope that data of individual absorbed dose can be obtained from the results of living organism tests for LMPF parameters.

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RADIATION DOSE STRUCTURE FOR RESIDENTS EXPOSED TO DUE TO MAYAK PLANT OPERATION AND NATURAL FACTORS IN THE URALS REGION; METHODS FOR OPTIMIZATION OF RADIATION PROTECTION

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The studies interpreting the impacts of radiation accidents in the South Urals are mainly focused on the doses due to industrial sources of contamination of the territory. There is a lack of works dealing with assessment of total population exposure doses from all sources of ionizing radiation including natural, medical industrial and other types of uses of man-made radiation.

With the purpose of determining the structure of exposure doses for residents of 30 villages located in contaminated areas of Chelyabinsk, Sverdlovsk and Kurgan Regions the current doses from the following sources were analyzed:

- background (natural) external and internal radiation;
- medical radiation exposure of population due to roentgenologic diagnostic procedures;
- environmental exposure due to releases of Sr-90, Cs-137 and Pu isotopes in the environment.

The method described in [1] was applied to assess the gamma-background levels differentiated according to radiation sources.

Estimation of contents of radon, radon daughter products (RDP) and thoron daughter products (TDP) aerosols in the air inside dwellings and communal buildings was made on the basis of the routine schematic combining instant volumetric activity (VA) measurements of RDP and TDP aerosols with the use of aerosol radiometers [2], and measurements of radon VA in the air using integral radon track detectors (IRTD). This approach allowed a prompt yield of results, which is of special importance in screenings, on the one hand, and, on the other hand, it allowed a correct assessment of average annual values of the equivalent balanced volume activity (EEVA) of radon in the air inside dwellings in the surveyed area. The latter circumstance is very significant

for decision-making on the necessity of undertaking a comprehensive radiation investigation based on the data of screening studies.

The dose coefficients applied for calculating doses due to radon and thoron included those listed in the Report of UN SCEAR: the average annual equivalent balanced volume activity of radon equal to 1 Bq/cu m results in an effective doses (ED) equal to 0.061 mSv/yr inside dwellings, and 0.025 mSv/yr inside production buildings (at an 8-hour working day). Average annual EEVA for thoron equal to 1 Bq/cu m results in ED = 0.29 mSv/yr inside dwellings and ED = 0.11 mSv/yr inside production buildings. ED resulting from inhalation pathways of RDP was calculated on the basis of the radon volume activity measured using IRTD, and ED due to inhalation intakes of TDP was estimated on the basis of instant thoron EEVA values.

To estimate the external exposure due to environmental sources it was necessary to study the intakes of artificial radionuclides with foods. For the purpose of estimating the content of radionuclides in foodstuffs and vegetation the recommendations given in [3] were used. The estimation of Sr-90 was effected by selective extraction using monoisoactilic methylphosphonate (MIOMPK) of yttrium-90 balanced with Sr-90 from 0.3-0.4 of normal nitric-acid solutions of food ashes with a subsequent measurement of Y-90 activity on NRR-610 TESLA radiometer. The background counting rate of NRR-610 radiometer is 0.04 impulses/sec which ensures measuring activity of the order of 0.06 Bq/test, with an error restricted to a maximum of 30%.

Cs-137 estimation was conducted according to the same method [3]. The method is based on Cs-137 concentrations in the sediment of nickel ferrocyanide and its subsequent separation in the form of antimonous iodide or X-chlortelluric salts. Cs-137 measurements were performed using NRR-610 radiometer.

Assessment of external exposure was made using a differentiated approach with respect to natural and artificial radionuclides [1]. Current effective doses of man-made radiation exposure were estimated according to the method described in [4]. The values of average annual effective doses from medical exposure were calculated on the basis of the roentgenologic data collected on the surveyed territory, and averaged dose parameters of the most common roentgenologic procedures. The information obtained was processed using the software "Region-At1U.K" elaborated by V.V. Nikitin

(Saint-Petersburg Research Institute for Radiation Hygiene).

The analysis of the results obtained has shown that the contribution of natural radiation sources to the structure of current doses to population constitutes 94%, it particularly applies to products of radium and thorium. It follows from the data presented in the paper that this contribution is especially significant for rural residents. This fact can be explained by the peculiarities of the construction of rural dwellings. In contrast to armored concrete floors built in multi-storeyed town dwellings the wooden floors in rural houses do not prevent radon from penetrating from the rocks into the room air. Besides, a rather severe climate of the Urals region makes the residents try to keep warmth inside the houses which results in inadequate ventilation and exchange of air in dwellings. The latter feature is typical of private houses. On the average, exposure doses due to radon and thoron are 3-4 times higher for rural than for urban dwellers. A higher contribution of thoron should be noted (9-10% vs the commonly observed 1-2%). The doses due to radon and thoron correlate well with the rates of natural radioactive mineralization of the underlyinbg rock in the surveyed area; thus, particularly high doses of natural exposure were registered in the villages Allaki and Tatarsky Karabolka located within the bounds of South Konevsky zone of elevated radon potential.

Doses due to medical exposure proved to be near the average values for developed countries, and under the conditions of the Urals region contributed up to 18% of the total dose. The task of minimization of doses resulting from medical procedures for exposed population persists as very urgent in the Urals region. It particularly applies to Sverdlovsk region where the average effective dose from roentgenologic procedures amounts to over 1.0 mSv/man vs the respective all-Russia value of 0.9 mSv/man.

The contribution to the total dose made by current average individual effective doses resulting from environmental radiation sources is the lowest. In 18% of the villages studied this exposure source contributed up to 2.5% of the total dose. In the critical population group this contribution amounts to 25% of the total dose. The table presents information on the dose structure in villages which were studied in a most thorough way.

RISK ANALYSIS IN APPLICATION TO POST-ACCIDENT MANAGEMENT

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INTRODUCTION

As it follows from the experience in the assessment and analysis of the consequences of nuclear accidents or nuclear weapon tests as well as in the implementation of the protection and restoration measures, there are some reasons, on the one hand, to go beyond the scope of the radiation protection and to consider the non-radiation sources of risk as well. On the other hand, remaining in the framework of the radiation protection, it is not enough to base oneself only on the dose approach and using the current concept of the effective dose D_E, even though only the stochastic effects of exposure are under consideration (detailed discussion on these aspects see in (1)).

Values of risk determined by D_E do not involve the time and cannot make allowance for local and age features of population cohorts (or personnel) for which the aftereffects are estimated. The necessity of estimating non-radiation risks goes from the following:

- Some countermeasures being implemented can have negative side consequences of a non-radiological nature for a population; for example, the relocation, as follows from the experience available, may adversely affect the human health because of changing the social and other living conditions;
- Some possible trouble with the health of the population caused by local or national-wide social living conditions requires, in the context of the most efficient investitions in health protection, to assess by a unified way—through the risk analysis—the state of health as a whole and the background radiation and non-radiation risk causes;
- As it follows from the present-day methodology of estimating the radiological risk, the background values of carcinogenic risk must be known for the application of this methodology (models of relative risk).

All these points show that for assessment of consequences of nuclear accidents or tests and decision making on their mitigation health risk assessment from various radiation and non-radiation risk sources should be developed and used.

METHODOLOGY AND DATA BANK FOR RISK ASSESSMENT

To meet these needs a methodology and a bank of data for risk assessment and management are developed.

In the frame of the CIS state research programmes (Chernobyl and Altai case studies) and the international (EU–CIS) project JSP2 the research subproject "developing the methodology (MAR) and data bank (BARD) on risk analysis" started in 1994. The first version of MAR is published in (1). Main functions of BARD are:

- assessment of the radiological and non-radiological consequences of nuclear tests and accidents,
- assessment of the health of a population in terms of risk indices.
- analysis of effectiveness of radiation and social protection measures.

One version of BARD is developed as a module for the decision support computer system for a post-accident management.

BARD includes

1. Service and calculation codes realizing the methodology mentioned;
2. Health-demographic data (HDD: the age-cause-specific death rates and the age distribution density) which are neccessary for radiation and non-radiation risks assessment.

HDD have been prepared for population of many regions of Russia for different years, for some regions of CIS and some countries around the world.

Input data for BARD are: 1) values of absorbed or equivalent doses (short-term exposure) and dose rates (chronic exposure) of different human body organs due to radiation exposure from a source considered. These doses or (and) dose rates should be given in their dependence on age, time at exposure, countermeasures adopted etc.; 2) HDD from the internal BARD data base; 3) primary radiation or non-radiation risks models.

BARD to a certain extent analogous to the computer codes ASQRAD and SPIDER being developed by CEPN (France) and NRPB (UK). BARD differs from them by the large intrinsic HDD data base, the possibility of calculating non-radiological risks, areas of application etc.

BARD is constantly supported and developed in the two versions: local and distributed. The last one can be accessible through Internet (`http://144.206.130.230/`) at RRC "Kurchatov Institute" (Moscow, Russia).

RESULTS AND CONCLUSION

With these RAM and BARD estimations of health effects of the nuclear weapon tests on the Semipalatinsk test site and the Chernobyl accident respectively for population of the Altai region and Russia regions adjoined to the Chernobyl zone were made. The goal of the study is to obtain data on consequences of the tests and the Chernobyl accident for planning social protection measures. Two examples of these estimations for the rural population of the Bryansk region (for the most contaminated territory, the variant of exposure doses without countermeasures) are given (see Fig. 1 and 2).

Using risk assessment results can change notions about consequences of the nuclear tests or accidents and effectiveness of countermeasures.

A role of the approach based on the risk analysis in decision making on protection and restoration measures is discussed. One should note that in Russia the additional development of regulation documents for radioactively contaminated territories with using the risk assessment results has already began.

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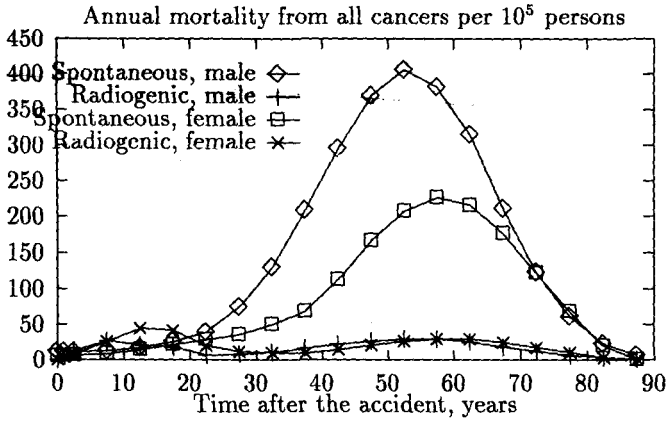

Figure 1: Annual mortality $\dot{M}(t)$ from spontaneous and radiogenic (due to the Chernobyl accident) cancers (per 100 000 persons with age 0 - 18 years at the accident), as a function of time t after the accident.

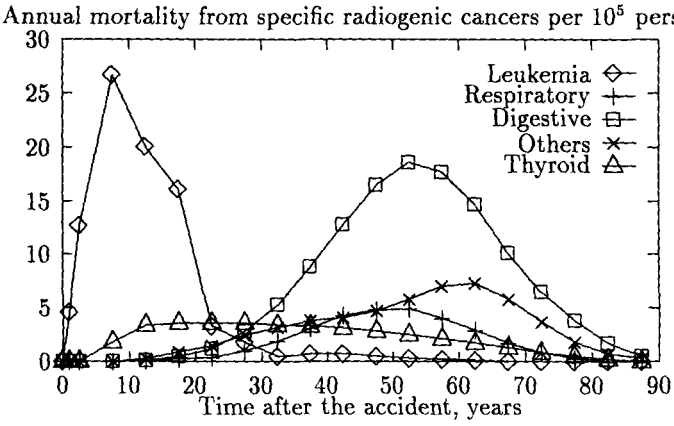

Figure 2: Annual excess mortality $\dot{M}(t)$ (morbidity for thyroid) from specific radiogenic cancers (per 100 000 persons with age 0 - 18 years at the accident, male), as a function of time t after the accident.

RESSAC *- A RESEARCH FACILITY FOR STUDYING RADIONUCLIDES BEHAVIOUR WITHIN ECOSYSTEMS

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* RESSAC: Radioecological research on Ecosystems Submitted to Accidental Situations in Controlled conditions

INTRODUCTION AND OBJECTIVES

In case of significant atmospheric contamination following a severe accident on a nuclear plant, it is crucial to be able to predict if, and how, the polluted environment may be detrimental to life, and in particular, human being. This prediction objective is currently supported by the development of controlled experimentations dedicated to elucidate : a) how, and to which extent, ecosystems and human-targetted food chains will be contaminated by radionuclides and, b) the appropriate actions to be undertaken, or not.

In this frame, IPSN developed a unique research facility were the study of radionuclides behaviour within ecosystems is undertaken. The facility, located into the Cadarache Nuclear Research Center, features in particular a 52 m long greenhouse specially designed for handling significant quantities of radioactive contaminants (Figure 1). The most immediate goal consists in clarifying, for a set of representative European environments, how radioactive contaminants will affect the food chains, due to : a) their specific interaction with water, soils and plants and, b) their fluxes of transfer through these compartments. A CEC supported European scientific collaboration has been assembled in the RESSAC Programme to address these questions.

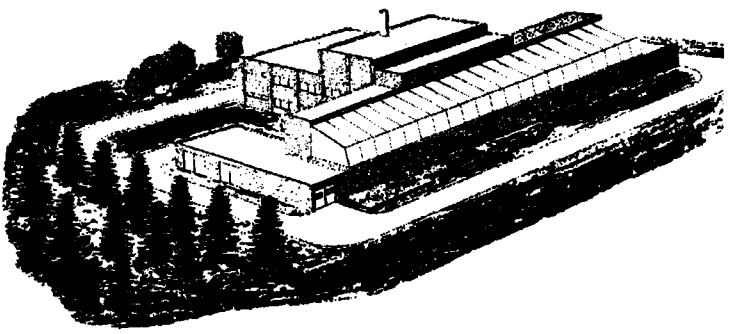

Figure 1: General view of the facility.

Figure 2 : Sample of soil (monolithe) extraction.

DESCRIPTION OF THE MAIN ASPECTS OF THE FACILITY

1) The large LYSIMETERS as realistic soil samples.

In order : a) to limit « border effects » which, at small scales, modify roots development, and b) to respect the vertical distribution of horizons and texture of the original soils, large undisturbed monolithes (2x2x1.4m, 20 tons mass each) have been extracted from selected sites in the surroundings of european

nuclear plants. Seven such « samples of soils » have been transported from, Belgium, Spain, Germany, the United Kingdom and France, for installation into the facility at Cadarache, (see Figure 2). Specific efforts have been devoted to reproduce, within each lysimeter, an hydric potential similar to that prevailing at the original sampling site. For this purpose, each lysimeter floor has been equipped with a special porous layer which contacts a water reservoir where the hydraulic pressure is controlled.

2) Simulation of the contaminated aerosols released from a PWR in accidental situation

A specific device has been developed (POLYR: Pollution of the LYsimeters Ressac) to carry out this simulation (Figure 3). POLYR consists in an inductive furnace which drives a fraction of 10^{-7} of a PWR core inventory to reach a temperature of 3 000°C, in a steam-saturated atmosphere at 80°C. The simulated corium in the furnace contains fifteen elements including fuel, clad, structure and control rods materials. ^{90}Sr and ^{137}Cs are also introduced as the two major radionuclide isotopes of radioecological relevance. A thirty minutes duration heating phase is required to reach the temperature plateau of 3000°C. Growth and maturation of the simulated aerosols require a fifteen minutes phase at this temperature. The resulting contaminated aerosols are next transferred above the lysimeters to be polluted. The pollution takes place by the natural fall down of the aerosols, a process which requires a duration of 24 h. Each lysimeter has been contaminated such as to reach an overall deposit of about 50 $MBq.m^{-2}$ for each of the two radio-elements. It must be noticed that this source-term can be modified on purpose to contaminate a variety of experimental objects in a variety of conditions.

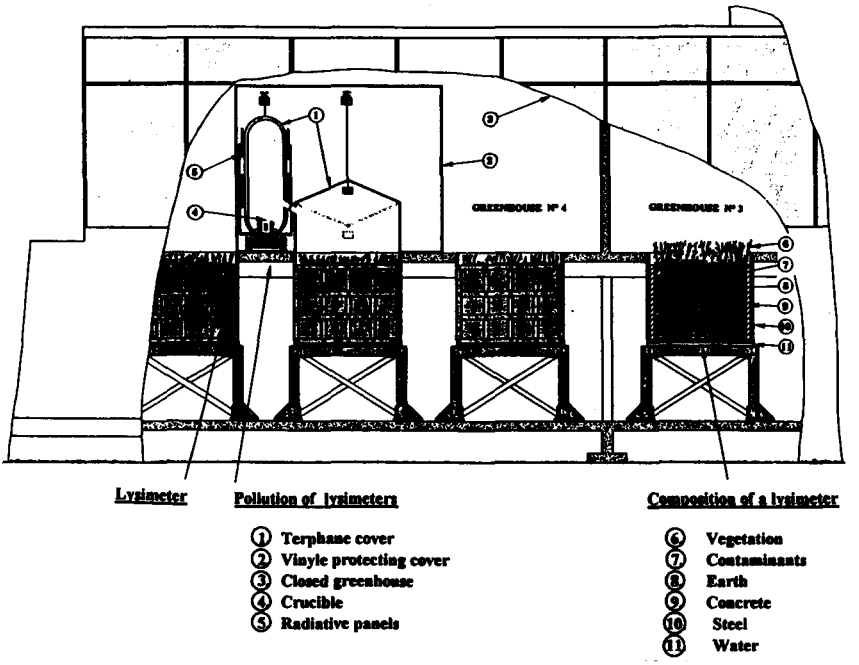

Figure 3 : The « POLYR » contamination device and installation of the lysimeters into the facility.

3) Climatic control within greenhouses for simulation of representative environments.

The facility includes four independent greenhouses (50m^2 on floor and 4m high) in a depressurised nuclear containment, associated with a radio-chemical laboratory. Physical and chemical preparation of the samples, as well as the gamma and beta countings, are carried out in this laboratory. In its lower part, the facility can host twelve large lysimeters, the surface of which is lining up the floor of the greenhouses in the upper part of the building (Figure 3). The lower part of the facility, where the lysimeters are installed along with their respective hydric potential control systems, is temperature-regulated, independently from the above-lying climatic conditions in the greenhouses. The controlled parameters on the ecosystems are : atmosphere temperature (0 to 35 °C) and relative humidity (10 to 90 %), rains, ligth irradiation (same spectrum than sun), underground temperature (5 to 20 °C) and soil hydric potential (0 to - 900mbar below atmospheric pressure).

CONCLUSION : FIRST EXPERIMENTAL RESULTS

The facility was totally operational at the beginning of 1994 and has first been used to study the interaction of radionuclide-containing aerosols with plant canopies (Figure 4), (1,2). As foreseen, the contamination has reached 30 to 80 Mbq.m^{-2} for ^{90}Sr as well as for ^{137}Cs. The wheat development stage influenced markedly the interception rates of ^{137}Cs and ^{90}Sr : small plants intercepted a reduced fraction of the aerosols deposit whereas large mature plants intercepted both radionuclides with a factor up to 90%. At each plant development stage, both radionuclides behaved similarly. Translocation of both radionuclides to the grains harvested from the wheat plants after they had reached maturity, showed that the grains were mostly contaminated by ^{137}Cs. The translocation to the grains was increasing when contamination of the plants occurred late in the plant development cycle, i. e. at a date close to maturity.

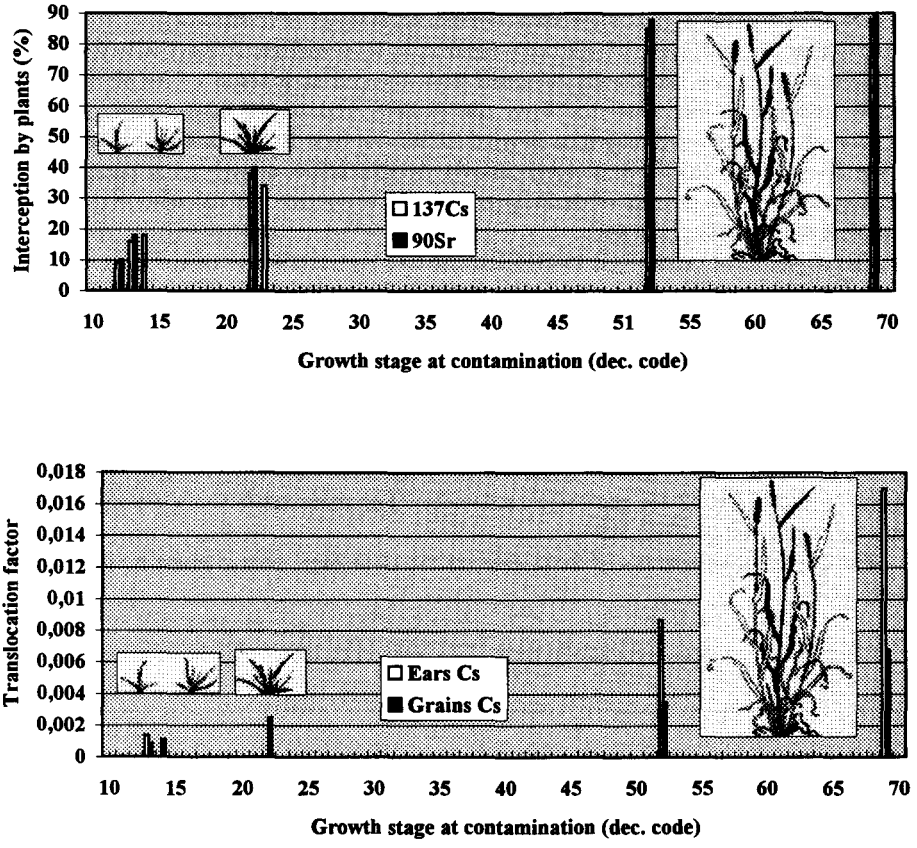

Figure 4 : Interception and subsequent translocation to grains of ^{137}Cs and ^{90}Sr versus wheat development

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COEFFICIENT OF SOIL DECONTAMINATION FROM STRONTIUM-90 AT THE TERRITORY OF EAST URAL RADIOACTIVE TRACE (EURT)

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The knowledge of coefficient of soil decontamination is necessary for predictions the dynamics in development of radioactive situation, for retrospective estimation and reconstruction of the initial radioactive contamination levels of territories.

By effective coefficient of soil decontamination from radionuclides (CSD) we mean the multiple of a nuclide completely removed from the soil as a result of its physical decay, active removal of radionuclides from the ground into water ecosystems, as well as biogeochemical and wind migration beyond EURT borders. Its dimension can be expressed in the divisible relation (in how many times?), in percentage, as well as in periods of half-desintegration, if the process of desintegration is the exponent. The calculation is made on a complete nuclides supply in soil.

The objective of the study is to calculate the effective coefficient of soil decontamination from ^{90}Sr on the basis of the analysis of the experimental materials received at EURT territories in Sverdlovsk region.

Three sources of the information were taken.

1. Minimum and maximum data received by central plant laboratory of combinate No 817 (CPL), Leningrad scientific-research institute of radiation hygiene Ministry of Public Health of the USSR (LSRIRH) and Institute of Applied Geophysics AS of the USSR (IAG), as well as data of Experimental scientific-research station at PA "Mayak" (ESRS PA "Mayk") in 1957-1962 for the same settlements. For these settlements we have out own data for the period of 1992-1994.

The comparative analysis of these data for the same local sites is given by the following coefficients (tab. 1):

Table 1.

Coefficient of soil decontamination from ^{90}Sr received by the comparative analysis of various initial data

Statistical parameters	CPL, IAG LSRIRH (at minimum values)	ESRS at PA "Mayk"	CPL, IAG LSRIRH (at maximum values)	On the integrated data (CPL, IAG,LSRIRH ESRS at PA "Mayk")
Mean	4,53	3,74	6,91	4,96
Confidence level (95%)	3,13 - 5,92	2,86 - 4,63	5,09 - 8,74	4,09 - 5,82
Standard error	0,71	0,45	0,93	0,44
Minimum	1,28	1,03	2,56	1,03
Maximum	10,29	7,18	13,71	13,71
Count	14	16	13	43

On evidence of ESRS at PA "Mayak" - 3,7±0,7, on minimal values of CPL, LSRIRH and IAG - 4,5±0,5, on maximal values of CPL, LSRIRH and IAG - 6,9±0,9. Basing on the integrated data of all organizations the effective coefficient is equal to 5,0±0,4 (range from 1,0 up to 13,7; confidence level at P=0,95: from 4,1 up to 5,8, n=43). At defectation of the doubtful minimum variables giving coefficients less than 2,4 (i.e. coefficient on pure physical decay of ^{90}Sr, fig., curve 1), the CSD becomes equal to 5,5±0,5 (range from 2,4 up to 13,7, confidence level at P=0,95: from 4.6 to 6,4, n=37).

2.The second source of calculation - data from the analyses of practically annual selections of undisturbed soil samples from 1975 till 1992 on two far remote experimental plots of radiological laboratory of chemical station of the Sverdlovsk region. These coefficients were calculated for practical purpose, and in the table they are submitted in percentage of soil decontamination of ^{90}Sr per year. CSD depends on a long period of time after the accident (tab. 2). So, in the first year the coefficient decounts for 6,2 percents, within the first five years - for the average of 5,8% per year, in the first decade for 5,3 and throughout the whole 36 year-long period - on the for average of 2,7% per year, i. e. the migration of ^{90}Sr has the tendency of decrease in time, the process of decontamination is not submitted to the exponential law (Fig., curve 2). According to these obtained data the CSD for the period from 1957 till 1993 is more then 5.

3. The third source of the information - materials of Sverdlovsk regional Sanitary Epidemiological Service. The comparative analysis was conducted for *the same areas of radioactive contamination* (more than 250 km^2) on the 1958 map (more than 10 Ci/km^2) and on the 1968 map. At comparing the levels of pollution it was revealed, that only in a compared decade the density of radioactive contamination by ^{90}Sr has decreased 5 times. Taking into account, that this coefficient concerns only to the first decade and not to the whole 36 year-long period, as we have calculated from the above reasoning it clear that of soil decontamination was much more, than five times.

Table 2

Coefficient of soil decontamination from ^{90}Sr, percents per year.

Years	Coefficient	Years	Coefficient	Years	Coefficient
1958	6,21	1970	4,97	1982	3,73
1959	6,11	1971	4,87	1983	3,63
1960	6,00	1972	4,76	1984	3,53
1961	5,90	1973	4,66	1985	3,42
1962	5,80	1974	4,56	1986	3,32
1963	5,69	1975	4,45	1987	3,22
1964	5,59	1976	4,35	1988	3,11
1965	5,49	1977	4,25	1989	3,01
1966	5,38	1978	4,14	1990	2,92
1967	5,28	1979	4,04	1991	2,84
1968	5,14	1980	3,94	1992	2,76
1969	5,07	1981	3,83	1993	2,68

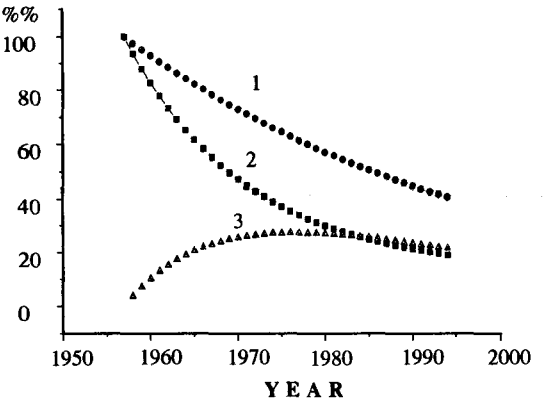

Fig. Dynamics of soil decontamination from ^{90}Sr.
Residual contamination, 100 per cent in 1957: 1 - On the physical decay, 2 - On the effective decay of soil decontamination. 3 - On the cumulative biogeochemical decontamination

Thus, the most probable coefficient of soil decontamination from ^{90}Sr in the Urals region for a 36 year-long period after the accident is equal to five.

In the following discussion of the received data we would like to call attention to the following:

On a data (Ionizing Radiation Levels and Effects, 1972) the soil loses in a year up to 2,5% ^{90}Sr at the expense of mechanical processes. Proceeding from these given during 37 years quantity of strontium has decreased in 2,8 times, we have this value equal 2,6. It should be noted, that the coefficient 2,8 takes into account not complete migration of nuclide from soil, as far as in it has a share of ^{90}Sr, entering of vegetable cover, with further partially coming back in soil.

Here we are dealing with the effective coefficient of soil decontamination from ^{90}Sr, which includes, as was stated above, its physical decay (this coefficient is equal to 2,4 per 36 years, fig., curve 1), active removal of radionuclides from the ground ecosystems into water ecosystems (lateral migration at the bottom of the lakes, bogs, rivers and ocean), as well as biogeochemical and wind migration beyond the EURT borders.

The correctness of application of the calculated coefficient of soil decontamination should especially draw attention to the analysis of highly-contaminated continuous of plots at EURT-territories (in "head part" of trace, for example) with virtually closed biogeochemical radionuclides cycle.

It should be especially noted that the biogeochemical decontamination in the first years after the accident was higher and in the subsequent years the availability of ^{90}Sr has sharply decreased. The dynamics of ^{90}Sr migration is so, that already since 1983-1985 the active biogeochemical soil decontamination in EURT zone is accounted for by the physical decay of ^{90}Sr (fig., curve 1, 3). It does not mean, that ^{90}Sr penetration to animals (including humans) can be relatively low, on the opposite, migration of ^{90}Sr can be higher than the legal normative statements, that should be taken properly into account in practical activities.

THE PROBLEM OF INHALATION DOSE ESTIMATION IN CHERNOBYL NPP 30 km-RADIUS ZONE

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Compared to initial accident period, dose rate of external exposure had reduced by today tens and hundreds times as much; after accident release products had fallen out of atmosphere on underlying surface and ^{131}I had decayed, sharp inhalation exposure has been discontinued. Nevertheless the problem of radiation safety securing of personnel within 30km-radius zone of ChNPP remains sufficiently actual. Considerable contingent of persons, working (and living) constantly on radioactively contamenated territory is subjected to chronic or episodic exposure on the part of uncontrolled sources.

The difficulty of the individual inhalation dose determination is caused by (i) spatial and temporal nonuniformity of the air radioactive contamination, (ii) presence of the radioactive fuel aerosols, (iii) radionuclide mixture content in the human body.

AIR MEDIUM RADIATION STATE CONTROL SYSTEM

Air is controlled at 24 stationary posts and also with the aid of movable sampling apparatuses on automobile roads and in places of the most intensive works with the solid radioactive wastes. Every month 76 samples of air and 62 samples of fallout are analysed for determination of gamma-ray emissing nuclides content and also 25 samples of air and 10 samples of fallout - for determination of ^{90}Sr and isotopes of $^{238,239,240}Pu$.

CONTENT OF RADIONUCLIDES WITHIN LAND LAYER OF AIR

The main source of joining radionuclides with air within the zone of ChNPP is radioactively contaminated underlying surface. Considerable similarity as to nuclide ration (%) in soil and air witnesses this (the data on ChNPP 5 km-radius zone in 1995):

	^{90}Sr	^{134}Cs	^{137}Cs	^{238}Pu	$^{239,240}Pu$	^{241}Pu	^{241}Am
Soil	20.7	1.1	46.6	0.23	0.64	30	0.65
Air	17.0	1.6	69	0.09	0.25	11.8	0.25

At present according to data of measurements at the stationary control posts resuspension factor is estimated on the average over the range from $2 \cdot 10^{-10}$ to $5 \cdot 10^{-9}$ m^{-1}. During the period of 1986-1988 concentrations of radionuclides in air were quickly decreasing but over the last five years they have been remaining almost unvariable. Peaks of concentrations are not infrequently in step with dry windy weather and dips - with humid weather. Seasonal changes take place as well: concentrations grow in spring, after snow cover had vanished and soil had dried, and diminish in winter. Radioactive contamination of air increases in the event of forest fires. As a whole, during post-accident period concentration of radionuclides in air lowered a million times as much. Depending on the position of measurements the total average annual concentration of the radionuclide mixture ranges from 0.078 to 21.5 mBq/m^3. According to data of 1987-1990 an activity median aerodynamic diameter (AMAD) is over the range of 5-7 μm: ^{134}Cs - 5.0-7.2; ^{137}Cs - 5.3-7.2; ^{144}Ce - 5.5-6.5; $^{239,240}Pu$ - 5.0-5.5(1). In due course AMAD of aerosols almost does not change.

According to the data obtained from the stationary air samplers on the average the inhalation doses do not exceed 1% of the established annual limit of 20 mSv.

FUEL AEROSOLS INFLUENCE UPON INHOMOGENEITY OF PERSONS EXPOSURE

A great part of common activity of aerosols in the 30km-radius zone of ChNPP is concentrated on so-called fuel particles. These particles are met in air quite seldom therefore it should be expected that inhalation doses are distributed to contingent of persons unevenly even if these persons are under equal production and life conditions.

Specially realized researches show that 1000 m^3 of air contain on the average no more than 34 radioactive particles. About 33% of the particles contain alpha-radiating nuclides (1). Using these amounts and deposition factor for particles with AMAD of 5 μm we will obtain that mean number particles forming inhalation dose, makes up, for example, in Chernobyl 0.15 particles per year and in industrial zone of ChNPP - 1.1 particles per year.

Distribution of such rare events has been accepted to describe by the Poisson law. Then, for example, under conditions of the Chernobyl town two particles have been laying in lungs for a year, by probability 0.01 (or at 1% of persons), that is 13 times the mean number; in industrial zone of ChNPP four particles have been laying in lungs for a year, by probability 0.02, that is almost 3.6 times the mean number for this point.

SPACE AND TEMPORAL IRREGULARITY OF CONCENTRATIONS' FIELD

Source intensity of secondary contamination of the land-layer of the air depends on many parameters: contamination density, physical and chemical forms of radionuclides, weather conditions, state of surface.

Average annual concentrations of radionuclides in air within periphery of the 30km zone and in its central part differ more than by two orders.

The local concentrations in the ChNPP zone that happen by mechanical or fire influences the contaminated surfaces are tens and hundreds times the concentrations measuring with the stationary samplers. Especially high concentrations were monitored by simple operations (transportation, loading) with slightly solid radioactive wastes. (See Tabie).

The local concentration (C_L) of $^{239,240}Pu$ in air in comparison with the concentration (C_B) measured in the stationary sampler, mBq/m^3 and estimates of the commited dose equivalent (H_T) and committed effective dose (H_E) from the ingalation of local contamination air during 100 hours

Working place	Concentration		H_T, mSv			H_E, mSv
	C_B	C_L	Bone surface	Lungs	Red bone marrow	
The roadway	0.008	1.6	0.64	0.092	0.028	0.043
Loading the ground	0.052	13.0	5.09	0.73	0.23	0.34
The solid waste repository dumping	0.052	340	134	19.2	5.94	8.9

Changeableness of weather conditions and technological regimes in the ChNPP zone, considerable differences in physical processes of forming background and local concentrations put obstacles in the way of establishment of correlation relationships between these concentrations.

Inhalation doses calculated by taking into consideration the assumption that a person without means of individual protection is being in the field of the local concentration during 100 hours, that is, 5% of annual working time, are shown in Table also. Under these conditions the principle dose limits are not exceed but it is obvious that in individual cases inhalation doses may be quite considerable and as a whole pronounced inhomogeneity of personnel's inhalation exposure takes place.

INCORPORATED RADIONUCLIDES CONTROL

Within the ChNPP zone inhalation doses are caused mainly with transuranium nuclides but there are the certain limitations of possibilities of their registration, especially in the presence of ^{137}Cs.

In 1994 26% of workers within 30km zone had content of ^{137}Cs over 1500 Bq, 9% - over 3700 Bq, 2% - over 11000 Bq according to data of measurement on the whole-body counter.

Content of ^{137}Cs in organism of persons being within the ChNPP zone is hundreds times the significant content of ^{241}Am; the latter is estimated by a magnitude of several Bq. For example, in 80 days after inhalation entrance on the level of one ALI (200 Bq) about 9 Bq of ^{241}Am were laid in all bone tissues (2).

Practical measurements at the modern measuring complexes confirm availability of difficulties when determining transuranium nuclides against a background of gamma radiation of ^{137}Cs. In the work of the Nuclear Safety and Protection Institute (France) the content about 20 Bq of ^{241}Am and about 1500 Bq of ^{239}Pu in the presence only 300 Bq of $^{134,137}Cs$ was determined. ^{241}Am registration limit by the content 2300 Bq of $^{134,137}Cs$ is estimated by the magnitude of 6.5 Bq by the time of measurement 60 min (3). The content of ^{241}Am on a high level 4...8 Bq by the content of ^{137}Cs on a low level 400...600 Bq was determined at the Ukrainian Scientific Center of Radiation Medicine also.

NORMATIVE SUBSTANTIATION OF INHALATION DOSES LIMITATION

The calculations of ^{137}Cs content in lungs and a body by the content of nuclides mixture on the authorized level were made (4) and was demonstrated that in nuclides mixture sizeable more low (8...2000 times as little) ^{137}Cs content, compared to regulated magnitudes. Thus, even background content of ^{137}Cs under conditions of the ChNPP zone by itself does not guarantee strict keeping established dose limits.

Normative documents do not reflect all specific properties of Chernobyl release radionuclides. According to data of 1993 about 42% of ^{137}Cs containing in the air has been in fuel aerosols (particles). This so-called fuel caesium is hard soluble and belongs to the lung Y class. Created by this ^{137}Cs dose in lungs is 80 times as high as proper dose of ^{137}Cs of D class (5), and only the latter is standardized in the ICRP Publication 61.

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INCORPORATION OF RADIOCESIUM BY THE POPULATION OF REGIONS IN BELARUS, RUSSIA AND THE UKRAINE AFFECTED BY THE REACTOR ACCIDENT IN CHERNOBYL

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SUMMARY
This paper reports on the results of a 3-year humanitarian aid project in the CIS funded by the German Federal Ministry of the Environment. The main objective of the project was to supply the population with independent information on radiation burdens and thus improve their psychological situation. More than 317000 measurements of post-Chernobyl body burdens of Cs-137 were performed in Belarus, Russia and the Ukraine from 1991 to 1993. In about 90 % of the cases annual doses derived from the body burdens did not exceed 0.3 mSv. Less than 2% of the measurements resulted in annual doses higher than 1 mSv. Open communication of the results helped to reduce fears in the population.

INTRODUCTION

On behalf of the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety, Forschungszentrum Jülich GmbH (KFA) organized and supervised a measurement programme at a total cost of DM 13 million to determine the radiation exposure of the population in the environment of Chernobyl after the reactor accident. This measurement programme was the Federal Government's response to a relevant request for help by the former USSR. The declared aim of the measurement campaign was to contribute towards objectively informing the population about their actual radioactive exposure. As was known from experience, this helped to eliminate many unjustified fears.

In order to examine the incorporation exposure mainly due to radiocesium taken up with food, a total of 317,000 measurements were performed on persons in 241 settlements in the period 1991 - 1993. Other γ-emitting nuclides were not observed.

EQUIPMENT

Three semitrailers were deployed in towns and major localities. The vehicles were each equipped with two nuclide-resolving whole-body counters of the FASTSCAN type from Canberra for incorporation measurements. This counter is equally suited for adults and children. During a measurement time of one minute it is still possible to detect 250 Bq Cs-137. For babies and very small children a special whole-body counter developed by the KFA was available from 1992 with which even 150 Bq Cs-137 could be detected. A further radiometric measuring system from Herfurth served as a standby for days with peak demand. The data were recorded by means of PCs. The results were stored in a data base.

In villages and youth camps, four smaller measurement vehicles equipped with radiometric measurement systems (QBM-1, Nuclear Enterprises) were used in 1991.

They were replaced in 1992 by a box-type van in which an incorporation monitor HM13010 (Herfurth) was installed.

RESULTS

Effective communication of the measurement results was an important prerequisite for achieving the humanitarian goals of the measurement campaign. A form with a registration number and the most important personal data was prepared for every person measured. The original was handed to the test person and one copy each was given to the local health authorities and the Research Centre Jülich. On this form, the measured value was confirmed and the radiation hazard assessed. For this purpose, the measured persons were classified into three categories (see *Table 1*). The threshold between categories 1 and 2 corresponds to an internal radiation dose of 0.3 mSv per year. The threshold between categories 2 and 3 corresponds to a radiation dose of 1 mSv per year.

The results are compiled by states in *Table 2*. In the first year, the measurements were completely concentrated on Russia. Measured values in category 3 were only obtained in just under one per cent of 163,000 cases. In the second year, the measurement campaign was extended to Belarus and the Ukraine. The cases in category 3 increased to 1.5 %, which should be attributable to a reduced number of test persons in less exposed regions of Russia and to the inclusion of further Russian measurement sites in zones with high soil contamination. In the third year, the number of cases in category 3 further increased since measurements were performed at locations with higher exposure in Ukraine and the measurement campaign in Russia concentrated more on settlements where high measurement values had already been found in the preceding years. Averaged over all three states and all three measurement years, however, the fraction of measurements in category 3 remained below 2 %. We recommended that this group of persons should be treated as is common practice in the Federal Republic of Germany for occupationally exposed persons, which means that they should be thoroughly examined at least once a year and that the incorporation profile should be controlled at regular intervals.

Table 3 shows average ingestion doses for adults in 1993 at locations in the Gomel region (Belarus). The doses were calculated from the arithmetic mean of measured activities. The highest dose of 2.2 mSv/a was observed in Kirov, a village near a forest, which was been partially evacuated. At this location, the highest activities were indeed measured during the entire three-year measurement campaign. Exposures of up to 770 kBq Cs-137 were determined corresponding to an ingestion dose of 29 mSv/a. The current annual dose limit of 50 mSv for occupationally exposed persons in Germany was thus not exceeded in this extreme case either. Average annual doses around 0.3 mSv/a, as observed in Leltschitzy, Narovlja and Svetilovitschi, must still be regarded as elevated. In Narovlja and Svetilovitschi this is due to elevated soil contamination. In Leltschitzy, on the other hand, soil contamination is fairly low. At this location, soils with a high transfer factor for the transfer of radiocesium from soil into plants are to be assumed as the cause for elevated exposure.

TABLE 1: CLASSIFICATION INTO CATEGORIES AS A FUNCTION OF MEASURED CS ACTIVITY

	children	adults
category 1	< 4 000 Bq	< 7 000 Bq
category 2	< 15 000 Bq	< 25 000 Bq
category 3	≥ 15 000 Bq	≥ 25 000 Bq

TABLE 2 : RESULTS OBTAINED ON INCORPORATION MEASUREMENTS IN 1991 - 1993

	Total	cat. 1	cat. 2	cat.3
1991 Russia	163033	93.7%	5.3%	1.0%
1991 Total	163033	93.7%	5.3%	1.0%
1992 Russia	49858	85.8%	11.7%	2.5%
1992 Ukraine	11373	100 %	0 %	0 %
1992 Belarus	29229	95.2%	4.4%	0.4%
1992 Total	90460	90.6%	7.9%	1.5%
1993 Russia	14836	70.0%	22.9%	7.1%
1993 Ukraine	36126	84.5%	11.9%	3.7%
1993 Belarus	12556	83.8%	12.4%	3.8%
1993 Total	63518	81.0%	14.6%	4.5%
1991-1993	317011	90.3%	7.9%	1.8%

TABLE 3: MEAN ANNUAL INTERNAL DOSE 1993 OF ADULTS IN THE REGION OF GOMEL (BELARUS)

Settlement	contamination zone [Ci/km^2]	mean body burden [kBq]	mean annual dose [mSv/a]
Bragin	15-40	3.2	0.12
Kirov	15-40	58	2.2
Korma	5-15	2.1	0.08
Leltschitzy	1-5	7.4	0.28
Narovlja	5-40	8.9	0.34
Shitkovitschi	1-5	2.7	0.10
Svetilovitschi	15-40	8.3	0.31
Tschetschersk	5-40	3.8	0.14

ENVIRONMENTAL MEASUREMENTS IN RUSSIA, BELARUS AND UKRAINE FROM 1991 - 1993

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INTRODUCTION

In the years 1991 to 1993 the Research Centre Jülich organised and conducted environmental measurements in the areas of Russia, Belarus and Ukraine which were contaminated by the reactor accident of Chernobyl. Within the scope of the "Measuring Program of the Federal Republic of Germany", investigations of soil contamination, area dose rate and measurements of food produced in these regions were performed. Some aspects of these investigations are to be discussed.

MATERIALS AND METHODS

More than 7000 environmental measurements have been conducted within the contaminated areas. Many teams from different institutions in Germany have participated in the three measuring campaigns each between three or four weeks a year.

The soil contamination was determined by taking soil samples and measurements with a gamma spectrometer in a mobile laboratory (measuring van) or by in-situ-gamma spectrometry. The depth profiles changed within short distances. This means that at each position the depth profile had to be determined separately which strongly extended the measuring time.

The gamma area dose rate was measured by small portable equipment.

But the main aim of our measurements was the determination of the contamination of food locally produced. For this purpose, food was brought to our van by the public after the announcement by the local authorities one day before. Immediately after the measurement a certificate in Russian language was handed over to the people which included the true measured value and a comment in regard to the Russian contamination limits which are similar to the international ones. Sometimes the measurements were difficult because the mass or the volume of the single samples were too low to be measured within the 10 minutes used normally or the geometry of the samples (e.g. meat) was different from the one used during calibration.

RESULTS

Each single measurement is listed in the reports (1 -3) of the different years or are available as data files. Therefore the single measurements shall not be reproduced in this paper but the ranges of the food measurements are given in Table 1.

In general it can be stated, that only a few samples in a few settlements are above the Russian contamination limits; e.g. self-produced milk in some settlements in the north western part of the Klincy district (Briansk region, Russia) showed high contamination levels.

DISCUSSION

1. MILK CONTAMINATION

Fig. 1 shows the mean values of Cs-137 milk contamination of different settlements in the Klincy district. It must be stressed again that these are the contamination values of the milk produced in the settlements and not the milk which could be bought in the local shops. The last one is much lower contaminated (about 20 Bq/l). From 1991 to 1992 the decrease of the milk contamination is obvious. But from 1992 to 1993 there was an increase in the milk contamination in some villages (Guta Korezkaja, Lopatni and Blisna). This can be explained by

- the lack or reduction of fertiliser given to the meadows or
- the measurements were not statistical representative

If the annual mean values of all single measurements carried out in all settlements in the Klincy district in 1992 and 1993 are compared, a small decrease in the milk contamination remains for 1993 in regard to 1992.

In Fig. 1, there are also given mean values for different settlements for the year 1994 which were made available by the Novozybkov Branch of the St. Petersburg Institute of Radiation Hygiene. The last measurements show that the general decrease with time of the milk contamination is going on in 1994 even if there are exceptions e.g. Beresowka, Guta Korezkaja.

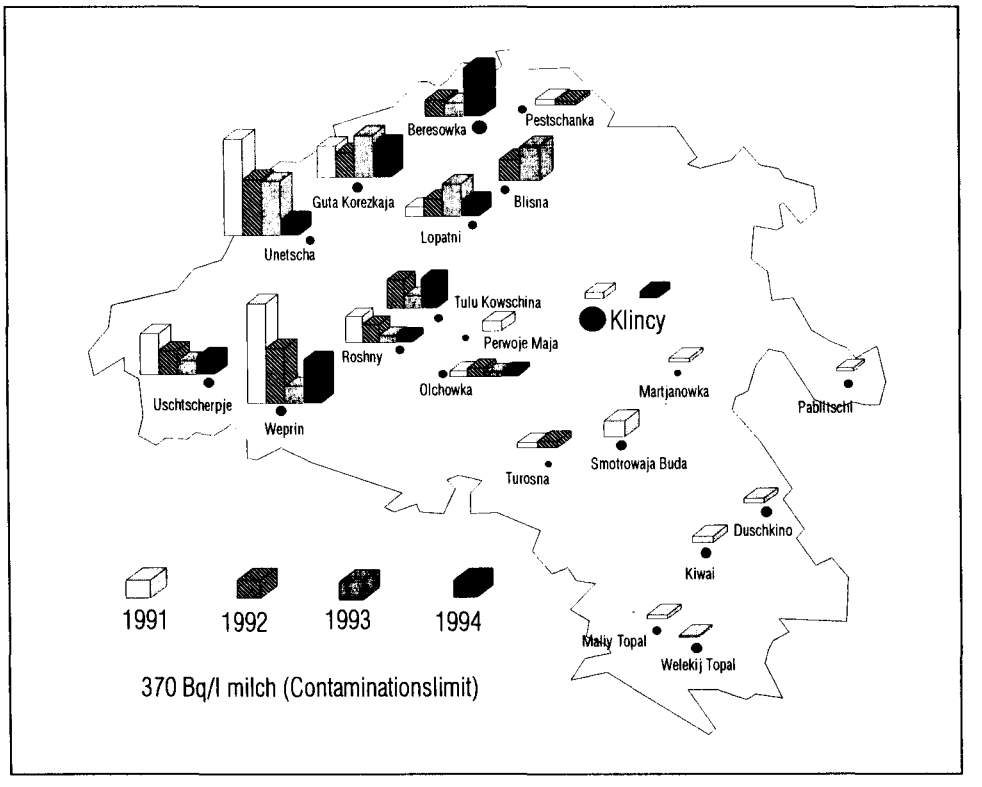

Fig. 1: The mean values of Caesium milk contamination in different settlements

2. AREA DOSE RATE

A correlation between Cs 137 soil contamination and the area dose rate measured 1 m above ground (same position) has been assumed. The area dose rate is due to contributions from mainly Cs 137 soil contamination as well as other nuclides like Cs 134, K40, the nuclides of the decay chains of the natural uranium and thorium families and the cosmic radiation. In Fig 2 the cross area dose rate is given as a function of Cs 137 soil contamination.

The good correlation regarded separately for meadows (undisturbed ground) and acres (disturbed ground) is most surprising (single measurements!) because each point in the figure is a composition of two independent determinations of the area dose rate and the soil contamination each of which has an uncertainty. These correlations enable us also to prove if there are incorrect single measurements.

The value of the area dose rate for the Cs 137 soil contamination "zero" is determined only by the contribution of the natural background and the background of the instrument. It results for the two correlations, (disturbed and undisturbed ground) in the same value of 0.20 μSv/h. For the Cs 137 soil contamination the net area dose rate above meadows is about twice the one over acres. For 1000 Bq Cs 137/m^2 we get an cross area dose rate of 0.69 μSv/h over acres and 1.28 μSv/h over meadows.

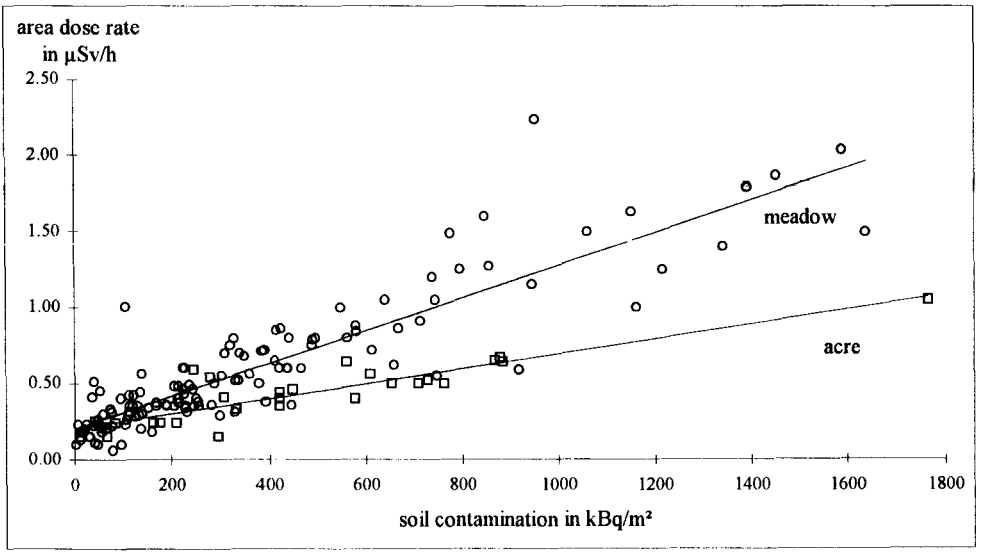

Fig. 2: Area dose rate as a function of soil contamination

3. FOOD CONTAMINATION

A comprehension of the single values determined in food measurements (1 - 3) is given in Tab.:1.From the table it becomes obvious that the food grown in gardens or acres is less contaminated than berries, mushrooms or meat of wild animals from the forest. Fish of prey is also highly contaminated

Tab.: 1 Measuring Ranges of Food Groups

area	fruits of the forest (Bq/kg)	fruits of the garden (Bq/kg)	cereals and their products (Bq/kg)	meat (Bq/kg)	fish (Bq/kg)
Klincy	<18 - 12320	<2,5 - 1270	79	4 - 4590	23 - 1170
Gordcjewka	89 - 18000	<4 - 514		<20 - 820	71 - 86000
Krasnaja Gora	119 - 16900	1,6 - 758	<3 - 8	27 - 500	28 - 2925
Belarus	<10 - 64000	<4,1 - <350	14 - 25	23 - 7311	61 - 607
Ukraine	<7,6 - 304	<1 - <10		<1	

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CONTRIBUTION OF THE SHORT-LIVED ISOTOPES INTO RADIATION SITUATION IN BELARUS AFTER CHERNOBYL NPP ACCIDENT.

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The reconstructed data about the radiation contamination of the some regions of Belarus was studed. The individual character radiactive fallout in verious regions has been found out. The contributions of short-lived isotopes in the integral activity and generation of the exposure dose rate was estimated. For reconstruction and systematization the contamination was used statistical treated data of the integral radiation-ecological data bank of the Institute of Radioecological Problems of the Academy of Science of Belarus. The data bank contain the results of gamma-spectrometric mesurements of the early soil samples (May - June, 1986), wherein La-140,Ba-140, Ce-141,Cs-136,I-131,Te-132 and I-132 radionuclides have been identified.

DATA BANK FOR THE DATA AT EARLY MEASUREMENTS OF RADIOACTIVE CONTAMINATION IN BELARUS AFTER CHERNOBYL NPP ACCIDENT

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INTRODUCTION

Radiation and ecological data bank of the Institute of Radioecological Problems/Academy of Sciences of Belarus (till 1992 the Institute of Nuclear Power Engineering/Academy of Sciences of Belarus (INPE ASB) was formed in June 1986 [1]. In 1986 the Institute of Nuclear Power Engineering was a leading institution in the Academy of Sciences of Belarus on studying radiation situation where the data of radiometric measurements of different organizations for analysis, forecasting the situation and mapping the contamination of the territory of Belarus were concentrated. For classification, verification, storage and timely processing of the information accumulated at the INPE ASB the data bank was created at computer ES-1061. As a software a program complex ASPID-5/ES was used which was developed at the Institute of Mathematics/Academy of Sciences of Belarus [2] . The initial data bank included only the results of gamma-spectrometric and radiochemical measurements of ring samples of soil which were selected and treated according to regulation documents developed and approved by INPE ASB and Goshydromet USSR on June 13, 1986. The very first results of gamma-spectrometric measurements of samples of the objects of the environment (soil, water, vegetables, etc.), obtained in May 1986 were not included into the data bank, because the distinguishes available of the techniques of their selection and measurement did not allow a unique data processing to be carried out. These data were only partially used for the analysis of radiation situation and mapping in 1986-1987.

Owing to active participation of many organizations in the works on selection and measurement of soil samples, as well as to timely processing of the data with the help of the formed data bank, in August 1986 the map of the contamination with ^{137}Cs of the whole territory of Belarus was drawn up at the Institute, and in November 1996 - the maps of contamination with ^{90}Sr and isotopes of Plutonium. For mapping more than 11600 gamma-spectrometric measurements of the objects of the environment from 1386 settlements and about 2100 radiochemical measurements were used [3].

Further the data bank was modified to some extent and added with a package of applied programs what made the preliminary processing and verification of the data be easy [4, 5].

ORGANIZATION AND STRUCTURE OF DATA BANK OF EARLY MEASUREMENTS OF RADIOACTIVE CONTAMINATION IN BELARUS

Last years an interest to reliable estimations of doses, obtained by the population during the first year after the Chernobyl NPP accident when they were the largest ones, was increased. It stimulated a further development of the data bank for more full presentation of early results of gamma-spectrometric measurements of samples of the environment as well as for more convenient processing of theirs. For that purpose in 1991 a new data base management control systems for computers PC AT/286/386 were developed at the Institute [6]. In this system every data base is represented by a set of files of relational type of DBF structure, but there are some distinguishes as to the description of fields formats in comparison with a traditional presentation of DBF files. An additional format was introduced for presentation of digital data with a float point "E". For the operation with such files an original interface was developed. The files created with the help of this interface are recognized by all similar systems (DBASE, FOXBASE, etc.), which process the files of DBF structure, but here the fields of "E" type are presented as symbolic ones.

Main functions of the system, providing with the access to the data, processing and modification of data bases, formation of complicated queries for data retrieval, generation of reports, etc., are realized in the form of menu tree.
Data bank presents a set of data bases of various assignment. The information about the measured samples of the objects of the environment (type of sample, technique and place of sample's selection, date of selection and

measurement of sample, exposure dose rate at the place of sample's selection, who selected and measured the sample, isotope activity in the sample and others) is stored in 18 files of DBF structure. Official information necessary for functioning of the system is stored in 17 files of DBF structure. Moreover, there are data bases of reference and normative information necessary for users.

APPLICATION PACKAGE FOR DATA PROCESSING AND CONTENT OF THE DATA BANK

For analysis, classification and verification of the data of early measurements a package has been created for data processing, which are presented in files format of DBF structure. The package includes the following main programs:

- programs for carrying out statistical analysis;
- programs for the calculation of exposure dose rate according to the isotopic composition of gamma-emitted radionuclides measured in the sample as well as recalculation of isotopes activity as of the specified date;
- verification programs for measurements of activities of genetically related radionuclides (for example, ^{140}Ba and ^{140}La, ^{95}Sr and ^{95}Nb);
- verification programs for measurements of activities of "similar" radionuclides (for example, ^{141}Ce and ^{144}Ce, ^{103}Ru and ^{106}Ru);
- verification programs for measurements of soil samples as to the ratio between dose rate of gamma-radiation above the surface of soil, measured when selecting the sample, and calculated according to the measured isotopic composition;
- verification programs of activities of non-volative radionuclides;
- programs of activities' reconstruction of genetically related isotopes with small periods of half-life-decay.

For investigation of space distribution of radioactive fall-outs on the territory of Belarus a geographic information system GEOS has been developed which is connected with the data bank [7]. The system GEOS has its own cartographic data base where there are the digital maps structured in a vector form, describing in equivalent form topographical maps of 1:200000 scale of the territories of Gomel and Mogilev regions. The system GEOS gives the possibility to carry out an additional analysis of the data when studying space dependence of activities' ratios in samples, their correlation connections as well as to draw various thematic maps for graphic supplements [8].

At present the data bank contains the results of radiochemical and gamma-spectrometric measurements of soil samples and grass samples selected from 1532 settlements of Gomel, Mogilev and Brest regions in 1986 and in 1987. The number of gamma-spectrometric measurements of samples constitutes:

12190 - ring samples of soil;
795 - "non- ring" samples of soil;
918 - samples of grass.

Short-lived isotopes were identified in a considerable part of these samples. Table gives the data about the number of samples of soil measured in 1986 and in 1987, where the activity of short-lived isotopes was measured.

Table

Number of soil samples in the data bank where short-lived radionuclides were identified

Radionuclide	Gomel region		Mogilev region		Brest region	
	1986	1987	1986	1987	1986	1987
^{132}I	66	-	-	-	2	-
^{132}Te	137	-	1	-	5	-
^{131}I	1075	-	605	-	52	-
^{136}Cs	651	-	276	-	24	-
^{140}Ba	158	-	1	-	2	-
^{140}La	717	-	257	-	2	-
^{95}Zr	4044	938	1454	186	495	59
^{95}Nb	4452	1194	2131	271	638	157
^{103}Ru	4647	321	2524	203	740	92
^{141}Ce	3742	198	932	186	375	89

At the nearest future the data bank is planned to be supplemented with available early measurements of samples of soil, grass and food-stuffs from the other regions of Belarus as well as the verification programs of the data to be modified with due account of changeable isotopic composition of fission fragments depending on burnup of fuel assemblies of RBMK-1000 reactor.

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ESTIMATION AND FORECAST OF THE RADIATION SITUATION ON THE TERRITORY OF THE REPUBLIC OF BELARUS AFTER THE CHERNOBYL NNP CATASTROPHE

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INTRODUCTION

During 10 years after the accident at the ChNPP the large amounts of the data on the content of radionuclides in the soil have been collected, using the network of the radiation monitoring stations.

In the after-accident period, the aerial photography of the territory of Belarus has been done, all populated areas have been investigated, including the agricultural and forest areas.

The information has been processed with use of the geoinformation RECASS system of construction of maps of distribution of caesium-137, strontium-90 and plutonium isotopes on the territory of the Republic.

The maps of the distribution of caesium-137, strontium-90 and plutonium isotopes on the territory of Belarus after the accident have been presented.

RADIOACTIVITY CONTAMINATION

The territories of 46,45 thousands of square kilometers have been contaminated with Cs-137 radionuclides with radiation density more, than 37 kBq/m2. 27 towns and more than 3000 populated areas with 2 millions of people, or nearly 1/5 of the population of Belarus, are located at the contaminated territories (Fig.1).

The Gomel, Mogilev and Brest Regions are the most contaminted as a result of the accident The radioactive contamination is of uneven, "spotted" character. There are cases, when within the same populated area on the relatively small areas, practically "clean" and highly contaminated plots are found close by. The Kolyban populated area of the Bragin Area, the Gomel Region can be taken as an example, where the lewels of the contamination with caesium-137 are in the range 170-2400 kBq/m2. Maximum levels of caesium-137 in the soil of the populated areas of the nearest zone have been fount out in Kruyki populated area of the Bragin Area, at the distance of 23 km from the ChNPP and consist of 28000 kBq/m2, and in the distance of 250 km in the Chudyany populated area of the Cherikov Area of the Mogilev Region are 28000 kBq/m2. The contamination of the territory of the Republic with strontium-90 is of local Character (Fig.2). The levels of the contamination of the soil with strontium-90 of 5,5 kBq/m2 have been fount out at the area of 21.1 thousands of square kilometers, which amounts to 10 per cent of the total territory of Belarus. The maximum levels of strontium-90 at the populated areas of the nearest zone have been found out in the Orevichi populated area of the Chojniki Area which is located at the distance of 28 km from the ChNPP and consists of 1800 kBq/m2.

The contamination of the soil with plutonium isotopes of 0,37 kBq/m2 has been found mainly in the Gomel Region and the Chudyany populaxed area of the Mogilev Region. The levels of the contamination of the soil with plutonium isotopes in the range 0.37 - 3.7 kBq/m2 have been found out in the Bragin, Norovlya, Chojniki, Rechitsy, Dobrush and Loev Areas of the Gomel Region. The levels of the contaminatin of the soil with plutonium isotopes of 3.7 kBq/m2 and more are Characteristic of the 30-th km zone around the ChNPP.

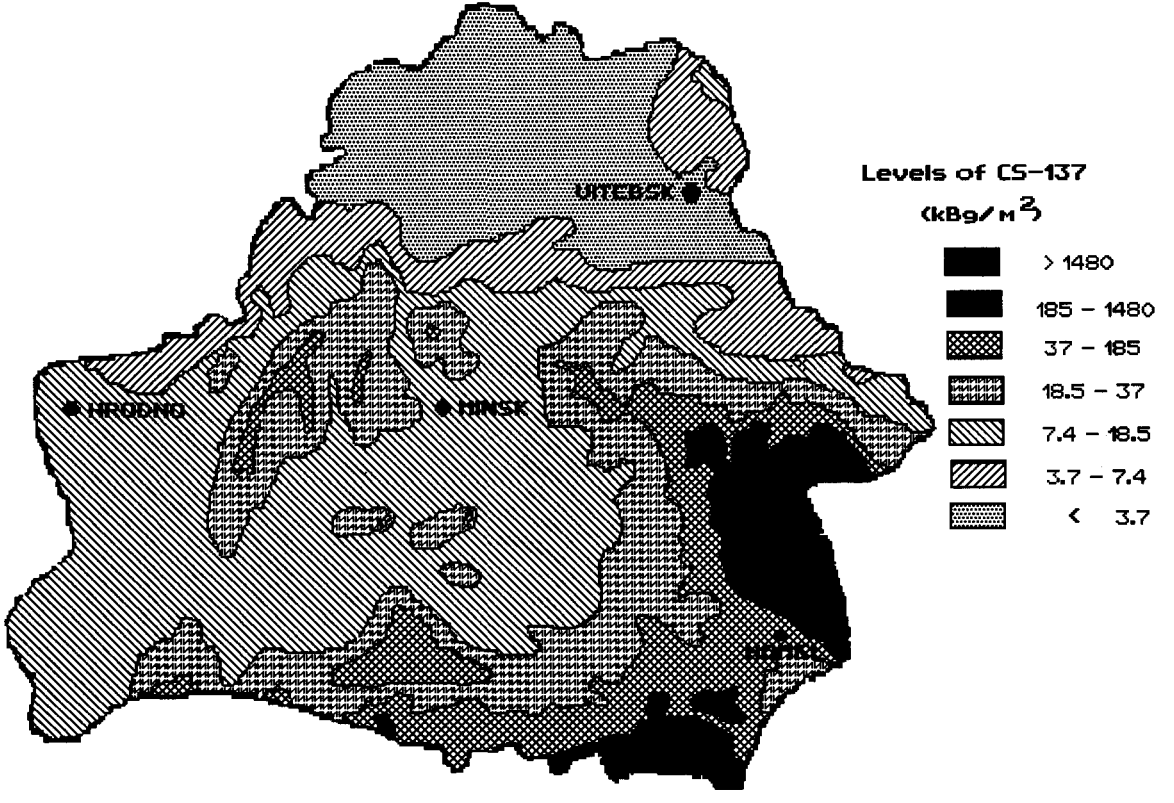

Figure 1. Radioactivity contamination of CS-137 in soil on the territory of Belarus (01.01.1995)

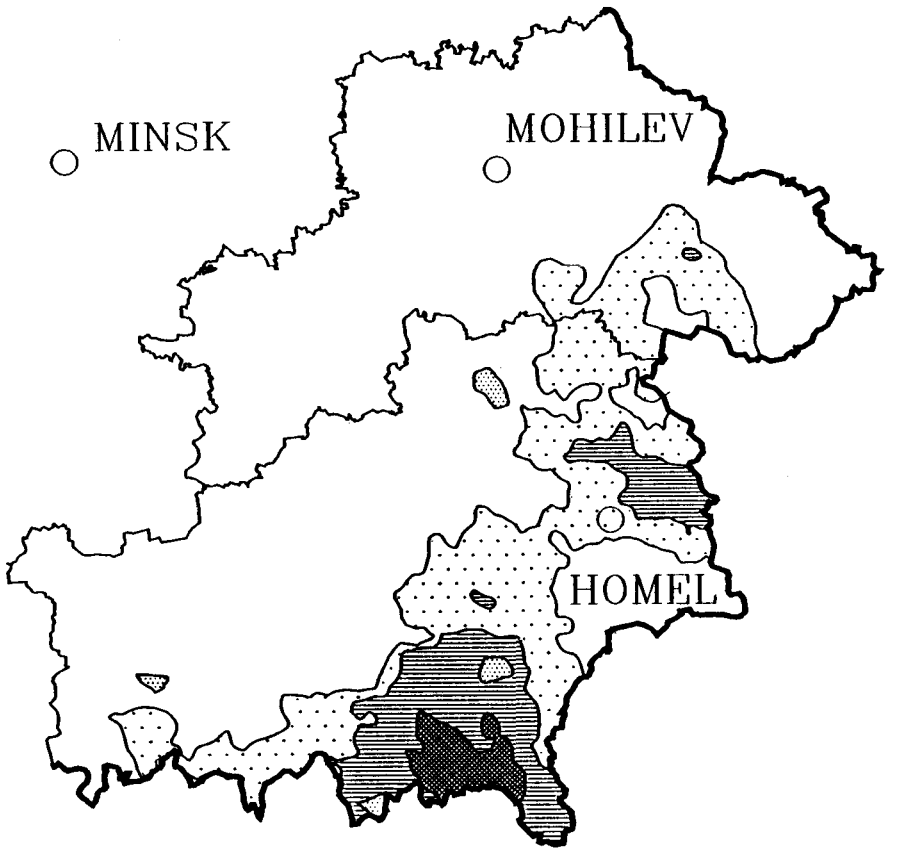

Figure 2. Radioactivity contamination of Sr-90 in soil on the territory Homel and Mohilev regions of Belarus (01.01.1995)

The forecast maps of the caesium-137 and strontium-90 inventory in the soil have been constructed with allowance for the natural decay of radionuclides for 2016 [1].

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THE RETROSPECTIVE ESTIMATION OF RADIATION SITUATION ON THE TERRITORY OF REPUBLIC OF BELARUS IN THE FIRST PERIOD AFTER THE CHERNOBYL NPP CATASTROPHE

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INTRODUCTION

The most complicated situation at the territory of Belarus has been observed in the first period after the accident at the ChNPP owing to the great quantity of the deposited short-lived radionuclides and the radionuclides with half-lives up to 1 year.

In 26 April 1986, the radioactive isotopes, including radioactive noble gases and iodine isotopes have initially reached the height of 1.8 km and began transport with air flows in the Nort-West direction through the western and central areas of Belarus. The analysis of the subsequent meteorological situation has shown, that the meteorological conditions of the movement of the radiation-contaminated air masses in 26 April to 10 May, 1986 have determined the radioactive contamination of Belarus, in the first phase with short-lived radionuclides.

RADIOACTIVITY CONTAMINATION

The reconstruction of the distribution of iodine-131 according to the situation for 10.05.86, ruthenium-106 and cerium-144 for 30.05.86 has been carried out (Fig. 1,2). The Methodical Approaches on evaluation of the contamination of the territory of the Republic with these radionuclides have been suggested for construction of the maps-reconstructions.

In April-May, 1986, the nearest zone (10-30 km) has been mostly contaminated with iodine-131, that is, the Bragin, Chojniki, Narovlya Areas of the Gomel Region (the levels of the contamination of the soil with iodine-131 have reached 37000 kBq/sq.m and more the dose rates of 25-100 mR/h, including the Chechersk, Korma, Buda-Koshelevsk, Dobrush Areas, where the levels of contamination of the soil with iodine-131 have reached 5550-11100 kBq/sq.m, 11100-18500 kBq/sq.m.

The Elsk, Lelchithi, Zhitkovichi, Petrikovichi Areas of the Gomel Region, Pinsc, Luninets, Stolin Areal of the Brest Regions have been highly contaminated with iodine-131.

In 1986-1989, the significant contribution into the radiation situation at the territory of Belarus have made such radionuclides as cerium-144 and ruthenium-106. The Narovlya, Bragin, Chojniki Areas are the most contaminated with cerium-144 areas (the levels of contamination are 555-1480 kBq/sq.m). In the zone of the relocation of the population, the contaminations with this radionuclide have amounted to 1480-3700 kBq/sq.m. The Narovlya, Bragin, Chojniki Areas are the most contaminated with ruthenium-106 areas (the levels of contamination are 185-555 kBq/sq.m). In the zones of the levels of the contamination with these radionuclides have accounted for 555-1480 kBq/sq.m.

The restoration of the situation in the first period after the accident can be used for the evaluation of the doses, obtained in the period.

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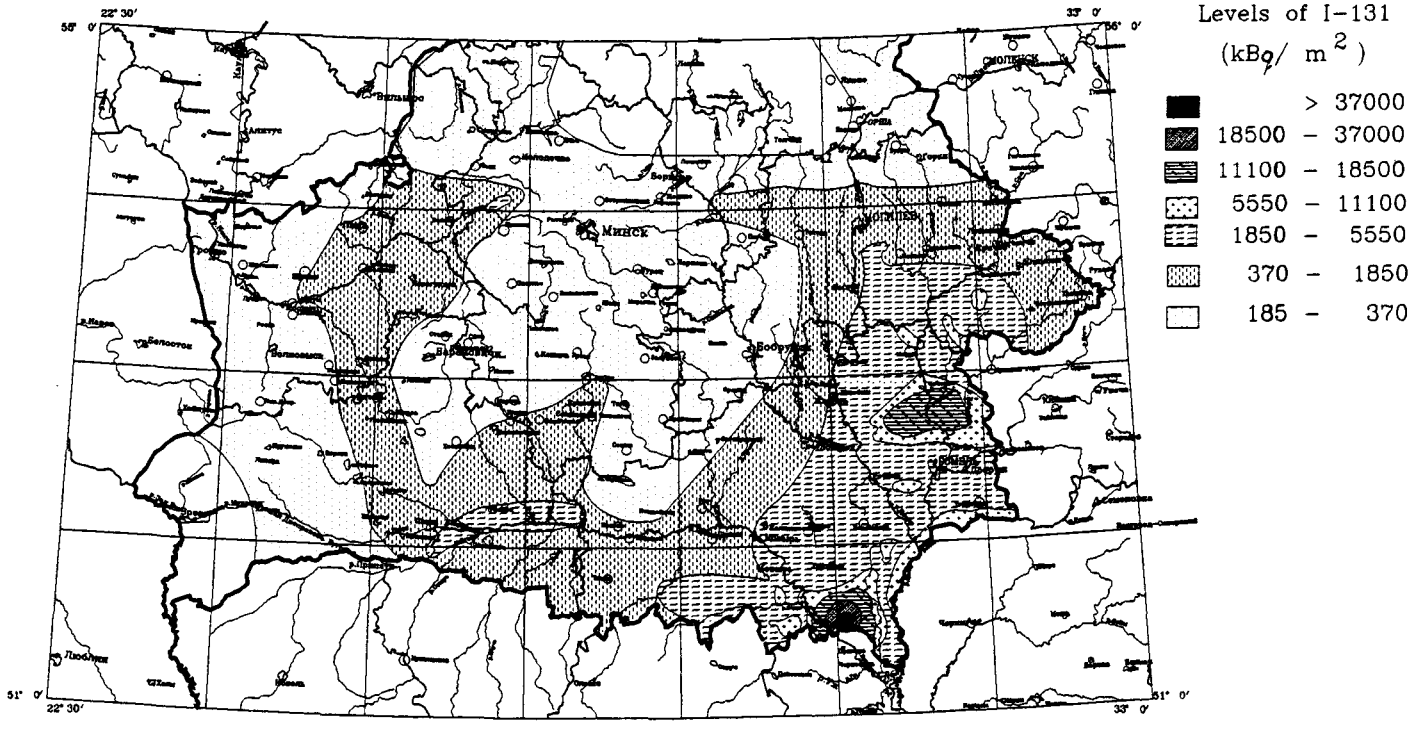

Figure 1. Radioactivity contamination of I-131 in soil on the territory of Belarus (10.05.1986)

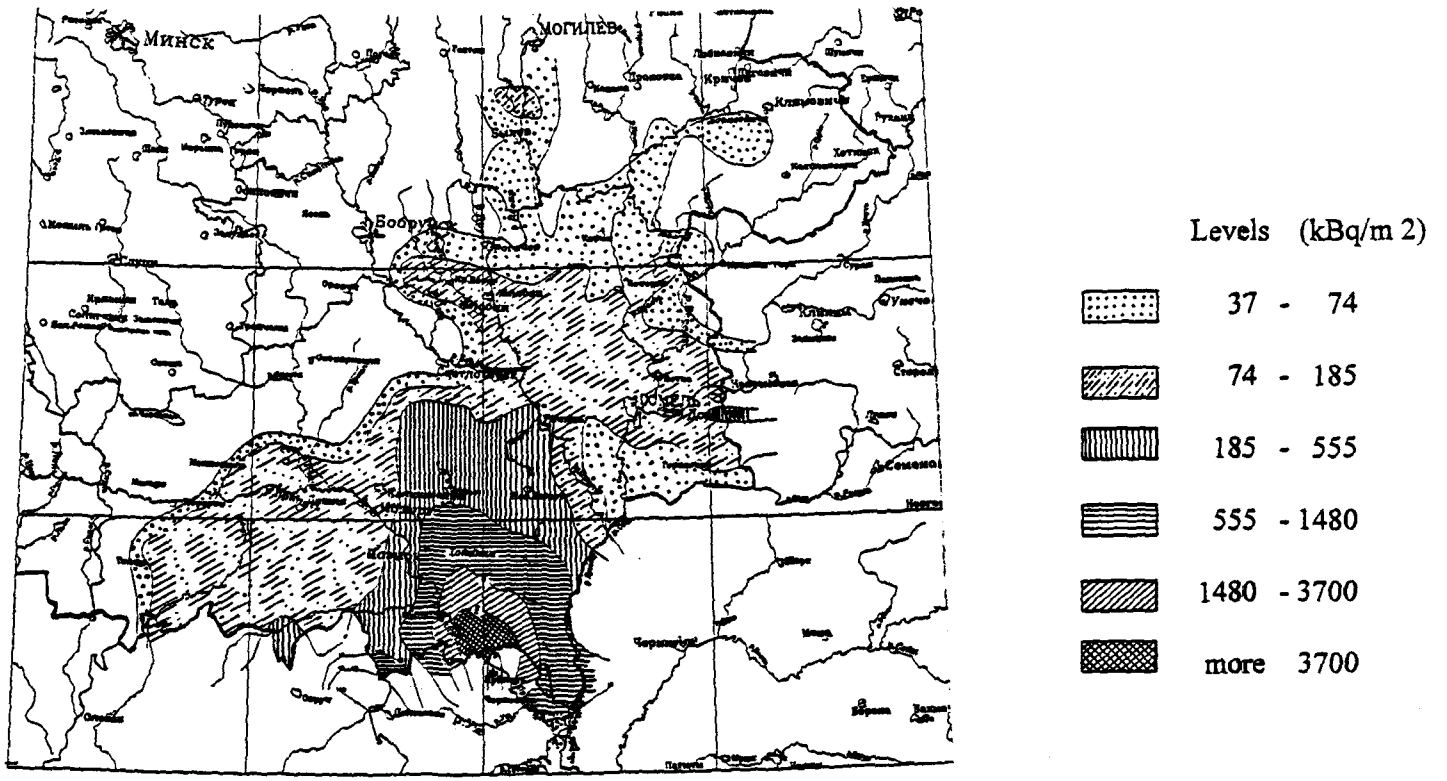

Figure 2. Radioactivity contamination of Ce-144 in soil on the territory of Belarus (30.05.1986)

ACTIONS ENTREPRISES PAR ELECTRICITE DE FRANCE EN BELARUS :

UNE SOLIDARITE ENTRE ELECTRICIENS

ACTIONS UNDERTAKEN BY ELECTRICITE DE FRANCE IN BELARUS :

A COMMUNITY OF INTERESTS FOR ELECTRICITY PRODUCERS

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L'accident de Tchernobyl a entraîné une très importante contamination en Ukraine, autour de la centrale accidentée, mais également en Belarus, république voisine de 12 millions d'habitants et en Russie (district de Bryansk).

Cette contamination a été secondaire aux rejets d'iodes et de césiums radioactifs. Son importance n'est pas seulement corrélée à la distance par rapport à Tchernobyl. Un bilan long et méticuleux publié par l'AIEA a permis d'en préciser, fin 1990, la géographie en "taches de léopard". La méconnaissance des niveaux de pollution radioactive dans les jours et semaines qui ont suivi l'accident n'a pas conduit à l'application de mesures prophylactiques et a entraîné une irradiation, à des niveaux variables, de la population biélorusse. Les enfants, dont la thyroïde est particulièrement radiosensible, ont reçu les doses les plus élevées.

De nombreuses initiatives nationales et internationales d'aide et d'assistance ont été prises dès 1986 dans un double but humanitaire et scientifique. Elles concernent essentiellement l'Ukraine et manquent cruellement de coordination.

C'est dans ce contexte et dans un souci de solidarité entre électriciens que les directions d'EDF ont décidé d'établir, en 1992, un plan d'aide et de collaboration avec les autorités

One of the consequences of the Chernobyl accident was the spread-out of an important contamination both Ukraine and Belarus (12 million inhabitants), as well as Russia (Bryansk district).

This contamination is correlated with the disposal of radioactive iodines and cesiums. The importance of this contamination is not only correlated with the distance to Chernobyl. A very long and thorough study, issued by AIEA specialists at the end of 1990, i.e. three and half years after the accident specified the precise geography of the contamination called "leopard spots". Since people discovered this contamination level as late as 1990, they did not apply the countermeasures and, as a consequence, the population, especially the bielorussian were severely irradiated. Children, whose thyroid is especially radiosensitive, received the highest doses.

Lots of national and international measures of help and assistance have been taken as early as 1986 both for scientific and humanitarian purposes. This help was essentially devoted to Ukraine and suffered from a deep lack of coordination.

In this context and to serve the common interests of electricity producers, the management of EDF decided in 1992 to set up a assistance and collaboration plan with the belarus authorities, having to face and deal with the medical consequences of the Chernobyl accident in their

biélorusses confrontées à la difficile gestion des conséquences sanitaires de l'accident de Tchernobyl sur leur territoire.

C'est ainsi qu'un contrat a été signé en février 1993 par EDF d'une part et par le gouvernement biélorusse d'autre part (Ministères de la Santé et de l'Energie, Comité Gouvernemental d'Etude des Conséquences de l'Accident de Tchernobyl et Académie des Sciences).

Ce contrat est actuellement établi pour une durée de 3 ans et porte sur une enveloppe globale de 10 millions de FF.

Ce contrat prévoit :

- participation française :

EDF a choisi d'inscrire cette collaboration dans le cadre d'une action concrète. Elle comporte 2 volets :

a) Fourniture de matériel dosimétrique et technique (*) :

- un spectromètre Gamma a été installé à l'Institut de l'Energie
- une soixantaine de dosimètres d'ambiance et 600 dosimètres individuels ont été fournis au Ministère de l'Energie.
- 80 balises de surveillance d'environnement ont été mises à la disposition des autorités biélorusses.

b) Mais l'essentiel de ce programme s'articule autour de la fourniture de matériel médical (*) et dans ce domaine, il s'agit surtout de répondre de façon rapide et utile aux attentes prioritaires des médecins confrontés à l'augmentation importante des cancers de la thyroïde chez l'enfant.

C'est ainsi qu'une gamma-caméra Sopha-Médical a été installée à l'Institut des Radiations à Minsk en 1994. Cet équipement est indispensable au diagnostic, au suivi et au traitement du cancer de la thyroïde, et jusqu'alors, il n'y avait pas d'installation de ce type en Bélarus. Cet appareil est à présent tout à fait opérationnel et pour en assurer une utilisation optimale, deux médecins ont été formés en France, au Centre Hospitalo-Universitaire de Créteil.

own country.

An official contract was set up in 1993 between EDF and the Belarus Government (Health ministry, Energy ministry, Governmental Committee on Chernobyl Consequences and Academy of Sciences).

The terme of this contract is 3 years and the costs for EDF are FF 10 million.

- French participation

EDF choose to provide concrete collaboration, by

a) supplying technical and dosimetric equipment :

- a gamma spectrometer has already been installated at the Energy Institute.
- 60 ambiance and 600 individual dosemeters have been supplied to the Energy Ministry.
- 80 environmental survey devices have also been supply to Belarus authorities.

b) But the main programme is devoted to medical equipment and, in this field, it appeared very urgent to respond to the inquiries of medical doctors involved in the treatment of thyroid cancers in chilren.

Within this framework a SOPHA MEDICAL gamma-camera arrived in Minsk (Radiation Institute) in 1994. This equipment is essential to diagnose, treat and control thyroid cancer. This EDF gift was the first device of this type installated in Belarus. It is now well operating ; two medical doctors from the Minsk Radiation Institute have been trained in a Parisian hospital (Médecine Nucléaire - Créteil) for 3 months.

Rappelons, en effet, qu'en Bélarus, le problème médical essentiel, suite à l'accident de Tchernobyl, concerne la pathologie thyroïdienne et, en particulier, l'augmentation importante de l'incidence des cancers de la thyroïde chez l'enfant (environ 500 cancers en excès depuis 1990). Ceci peut s'expliquer par l'importance des rejets d'iodes radioactifs (^{131}I, mais également isotopes à périodes plus courtes) lors de l'accident, la fixation élective de ces radioéléments sur la thyroïde et la radiosensibilité particulière de cet organe chez l'enfant.

- participation bélarus :

Elle s'inscrit, quant à elle, essentiellement autour d'une information en retour portant sur les divers aspects de cette catastrophe, en particulier et surtout sur le suivi sanitaire et médical des populations irradiées du fait de cet accident.

Il apparaît tout à fait fondamental de tirer le maximum d'enseignements de cette catastrophe, en particulier au plan médical. C'est là l'une des justifications de cette action.

(*) Il s'agit là des actions entreprises dans le cadre de la première année de collaboration.

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As we all know, the most dramatic medical problem in Belarus, after the Chernobyl accident, concerns the thyroid pathology, especially thyroid cancers in children : 500 cancers have been recorded since 1990. The increase of this pathology can be correlated with the amount of radioactive iodine released in may 1986 and the high radiosensitivity of thyroid in children.

- Belarus participation

We are, of course, very interested in any available information on the medical consequences of this accident.
It is more than ever essential to learn as much as possible from this disaster especially in the field of medical consequences.

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PAPER TITLE

Long-term uptake of cesium-134,137 by the body of inhabitants of Russian territory contaminated as a result of the Chernobyl accident

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MAJOR SCIENTIFIC TOPIC NUMBER 4.2. (see page 7)

ABSTRACT (See instructions overleaf)

To find out regularities of formation of internal dose of the Russian population living on the territory contaminated with long-lived radionuclides, we investigated food rations of over 2700 inhabitants of the Bryansk region and measured actual content of Cs-134,137 in the body of them in different periods after the Chernobyl accident| in 1987, 1990, 1993, 1994-1995. The paper presents the data about urban and rural inhabitants of both sexes. The questionnairy included information about annual food ration of a person before the accident, its changes connected with the accident, the results of radiometry of the body and information about individual countermeasures. We have also studied the content of Cs-134,137 in agricultural and natural food products. Thus, the internal exposure dose was estimated by two methods| as calculated with contaminated ration and according to the actual content of radionuclides in the body. The role of natural products (mushrooms, forest berries, fish)in the formation of the dose undoubtedly increases in remote terms, since natural decontamination of the natural ecosystem from transportable forms of radioactive cesium occures considerably slower than that of the agricultural ecosystem. Countermeasures in agricultural product ion and delivery of non-contaminated products decreased intake of Cs-134,137 with food ration by a factor of 3-10. The paper presents quantitative statistical analysis of the role of different food products and of different countermeasures in formation of the internal doseof the population in different periods after the Chernobyl accident.

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PAPER TITLE Detaled Radiological Survey of the Private Farmsteads as a Way of Development of Population Individual Dosimetry.

AUTHOR(S) NAME(S) E.E. Petrov , A.N. Zabudko , I.P. Sadokhin

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 (see page 7)

ABSTRACT (See instructions overleaf) The Association of Radiological Issues of Russia amalgamating Research Institutes in Obninsk carried out detailed radiological survey of more then 10000 private farmsteads in 125 rural settlements (about 170000 inhabitants) of Kaluga Region contaminated by the Chernobyl accident. The exposure dose power over the whole area of a farmstead, the amount of radionuclides (Cs^{134}, Cs^{137}, Sr^{90}) in soil and agricultural products and also in the surrounding's elements (pastures, forests, reservoirs) were measured.

The following results were obtained from the analysis of the data:

- statistical distributions with large dispersion of the exposure dose power, contamination of farmsteads soil and its agricultural products and also correlations between them;
- correlation of farmstead contamination with radicesium content in children bodies;
- radionuclides distribution across soil depth;
- radionuclides distribution in forest systems, including distribution in annual trunks;
- radionuclides content in air and in water.

The main conclusion is that inhabitants of one rural settlement live in different irradiation doses conditions. Information obtained allows to evaluate individual doses of external and internal irradiation for each inhabitant and to choose groups of risk to take special medical and financial support.

Presently the authorities prefer to use collective doses approach when strategic decisions are adopted. But in this case the interests of highest risk groups are taken into account incompletely.

WEATHERING OF RADIONUCLIDES DEPOSITED IN INHABITED AREAS

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ABSTRACT

When determining the long-term consequences of an accidental deposition of radionuclides from a nuclear power plant in an inhabited area it is essential to be able to predict the migration with time of the deposited radiocaesium.

Through the years that have passed since the Chernobyl accident occurred in 1986, the weathering effects on deposited radiocaesium on different types of surface in urban, suburban and industrial areas have been followed through six measurement campaigns to the Gävle area of Sweden. The weathering effects after the Chernobyl accident were also investigated in towns in the Ukraine and in Russia.

The radiocaesium level on asphalt and concrete pavements was found to decrease rather rapidly. It was found that the weathering effects over the first decade could be described by a double exponential function. Similar analytical functions were derived for the other urban surfaces. However, the weathering half-lives of radiocaesium on walls and roofs of buildings were found to be much longer.

Even in urban centres, the largest contribution to the dose-rate immediately after deposition often comes from the open grassed areas and areas of soil. As the dose-rate from such surfaces usually decreases slowly, depending on the soil type, the relative importance of these surfaces will often increase with time.

After a decade, the dose-rate from horizontal pavements will decrease by a factor of 10 or more, but the dose-rate from an area of soil or a roof may only be halved. Correspondingly, the dose-rate from a wall decreases by only 10-20 %.

INTRODUCTION

Typical relationships have been identified between the contamination levels on different types of urban surface shortly after the Chernobyl accident (1). These relationships change considerably with time, due to various 'weathering' processes (depletion of the retained material by such effects as wind, precipitation and mechanical impact of human activity) in the urban environment. An essential requirement for the process towards an estimation of the long-term consequences of an accidental deposit of radionuclides is therefore an adequate knowledge of the effect of weathering. For this purpose, investigations have been conducted in urban environments which received either a wet or a dry deposit of ^{137}Cs after the Chernobyl accident.

URBAN WEATHERING MEASUREMENTS IN SWEDEN

Since the Chernobyl accident, the weathering effects on different types of surface in urban, suburban and industrial areas have been followed in the Gävle area in Sweden, which received a high level of contamination with a heavy rainshower shortly after the accident. In situ measurements with germanium detectors shielded/collimated with lead were made in the centre of Gävle, in a light industrial area about 1 km east of the town centre and in a suburban/rural area about 13 km to the north-east of the town centre.

The radiocaesium levels were measured on various kinds of surface, including walls, grassed areas, pavements, walkways and roads. Six measurement campaigns were conducted in the Gävle area in the period 1987-1994. Table 1 shows a representative selection of the measured data.

Table 1. ^{137}Cs levels measured in Gävle, Sweden in 1987, 1988, 1990, 1991, 1993 and 1994

Location	Surface	Material	Orientation	^{137}Cs levels measured in Gävle, Sweden					
				1987	1988	1990	1991	1993	1994
City hall	wall	plaster	south-east	-	0.78 ± 11%	-	-	-	0.53 ± 12%
City hall	wall	plaster	south-west	-	0.55 ± 16%	-	-	-	0.21 ± 27%
City hall	pavement	flagstone	middle	-	9.71 ± 6%	4.26 ± 7%	3.13 ± 8%	3.10 ± 9%	-
City hall	pavement	flagstone	west	-	6.40 ± 6%	3.15 ± 7%	2.17 ± 8%	2.12 ± 9%	2.12 ± 6%
City hall	road	asphalt		-	1.49 ± 9%	0.44 ± 17%	-	-	-
Gevalia	wall	red brick	south, washing	1.93 ± 9%	1.65 ± 9%	-	-	-	-
Gevalia	wall	red brick	north, washing	-	3.93 ± 7%	-	1.85 ± 7%	2.13 ± 15%	1.89 ± 7%
Fire station	wall	yellow brick	north, dry deposition	-	0.42 ± 15%	-	-	-	0.14 ± 36%
Transformer substation	wall	yellow brick	south, washing	-	1.06 ± 11%	-	0.80 ± 10%	0.76 ± 10%	-
Magasin St.	wall	plaster	north, washing	0.99 ± 10%	-	-	0.98 ± 10 %	-	-
General Food	wall	plaster	east	-	1.14 ± 10%	-	-	-	0.59 ± 12%
Fire station	pavement	asphalt+ flagstone		-	-	3.21 ± 8%	2.38 ± 8%	-	-
Gevalia	car park	asphalt		3.17 ± 7%	3.28 ± 7%	-	-	-	-
Gevalia	cross-roads	asphalt		-	1.19 ± 9%	-	-	-	-

Although the first campaign was made in 1987, a comparison with other measurements made in the same area in May-October 1986 (2) made it possible to approximately relate the measured results to the initial radiocaesium deposition in the area.

The dose-rate reductions which were recorded in Gävle on grassed areas were found to be greatly dependent on the soil type, and the investigated soils displayed many different shapes of vertical profile of ^{137}Cs. Generally, the dose-rate from areas of soil was found to decrease by 40-60 % through the first 8½ years after deposition.

As can be seen from the table, very little, if any decrease in the levels of radiocaesium contamination on walls of buildings was identifiable 8½ years after the deposition. In one case, the level has actually increased at one point. This change is, however, not highly significant, but may have been caused by a wash-down of radioactive substances from the upper parts of the wall. Anyway, the measurement at this site the following year showed the expected decrease. Although it is difficult to distinguish between the weathering rates on the walls in Gävle previous efforts have shown that radiocaesium is retained most effectively by micaceous construction materials, especially those that have been fired at comparatively low temperatures, where the small openings in the mica structure are intact. A yellow brick wall of the fire station has a very low contamination level compared with the other walls because it was exposed to only dry deposition as the structure of the building prevents the wall from precipitation.

In contrast to the situation on walls, the levels of radiocaesium on asphalt surfaces have now decreased so much that less than 10 % of what was measured in 1988 is left. A comparison with the results of Karlberg and Sundblad (2) shows that this means that less than 2 percent of the initially deposited radiocaesium is now left on the road. These levels are now generally below the detection limit. This ties in with the results of laboratory investigations, which have shown that only a very little fraction of a caesium contamination is associated with the bitumen fraction of the asphalt. A large fraction of the caesium has been found to be associated with the more mobile street-dust, which usually contains micaceous substances weathered off various surfaces.

As for the concrete paved surfaces, the remaining 10 % of the initially deposited radiocaesium seems very firmly fixed. No significant decrease has been recorded over the latest 3 years. The weathering on horizontal hard surfaces was generally found to be faster in the more heavily trafficked spots.

Clay roof tiles which were contaminated due to the Chernobyl accident and exposed to the wind and weather in Gävle for four years were collected and brought to Risø, where the contamination level at this stage was assessed. The tiles were subsequently exposed to Danish weather after having been placed on a specially constructed scaffold. A decrease in the radiocaesium contamination level of between 28 and 35 % was recorded over the following 19 months period.

This is in reasonably good agreement with measurements on different types of roof contaminated by Chernobyl fallout at Riso, from which a semi-empirical model was derived (3). The weathering processes on both roofs, walls and horizontal pavings were found to have a slow and a fast component and could be adequately described over the first decade by two-component exponential functions. Some typical values of the parameters are given in Table 2.

Table 2. Typical weathering parameter values on different types of urban surface (slow and fast component)

Surface:	Fraction	Half-life	Fraction	Half-life
Roofs	0.7	$7.0 \cdot 10^3$ days	0.3	$3.5 \cdot 10^2$ days
Pavings	0.5	$7.0 \cdot 10^2$ days	0.5	$7.0 \cdot 10^1$ days
Walls	0.9	$2.0 \cdot 10^4$ days	0.1	$7.0 \cdot 10^1$ days

URBAN WEATHERING MEASUREMENTS IN THE FORMER USSR

Measurements of dry-deposited ^{137}Cs in the Pripyat area in the summer of 1993 with shielded germanium detectors showed that the contamination level on sandstone walls was in the range of 199-350 kBq/m^2 (the decrease since 1986 must have been very small), compared with 0.9-28 kBq/m^2 in other towns, where deposition occurred with rain. On the 'impervious' horizontal surfaces, such as roads and pavements, the caesium contamination level was found to be in the order of 30-350 kBq/m^2 in Pripyat, with the highest levels on concrete pavements and the lowest on asphalt.

From the contamination level and distribution of radiocaesium in local soil it was possible to estimate the caesium contamination level on pavings in the area shortly after the accident to about 1.5 MBq/m^2. This means that a much larger fraction remains on the paved surfaces in Pripyat than in Gävle. However, there has been very little traffic in Pripyat since the Chernobyl accident. If the weathering processes in the two towns had been the same, the remaining fractions in Pripyat would probably have been the smallest at this stage. Anyway, it was easier to remove caesium in Pripyat by forced decontamination trials than in other towns. The reason for this is believed to be that the contamination in Pripyat (near Chernobyl) took place with large (insoluble) core fragment particles.

CONCLUSIONS

The effects of weathering in towns and industrial areas have been investigated through field measurements in areas of Sweden and Russia, which were contaminated with ^{137}Cs after the Chernobyl accident. The radiocaesium levels on road pavings were found to have been halved after less than one year, and after about 3 years only about 10 % remains. The dose-rate contributions from roofs and soil areas were approximately halved in the first 8½ years after deposition, while measurements on walls showed little or no decrease in contamination level. Analytical expressions for the weathering processes on roofs, walls and pavings were derived and estimates of typical parameters were made. Measurements in Pripyat indicated that the impact of human activity on the decrease of the contamination levels can be great.

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INVESTIGATIONS TO THE DISTRIBUTION OF CS-137 AND K-40 IN A COW

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INTRODUCTION

It is of particular interest to study the distribution of radionuclides in domestic animals after a long-term ingestion of radionuclides. We performed a study to determine the specific activities of Cs-137 and of the naturally occuring K-40 in different tissues, organs and in the gastro-intestinal tract of a cow. As a consequence of the fallout following the Chernobyl accident in April 1986, the cow ingested chronically heavily contaminated forage during four years. It is the aim of these investigations to find out regions of high activity concentrations in the cow and to compare the activity concentrations of Cs-137 and K-40 in the samples of the whole cow. Of particular interest are the activities of radiocaesium in edible tissues (meat) and organs.

EXPERIMENTAL PROCEDURE

The cow was slaugthered in November 1992 and dissected in anatomical parts. Samples of the whole body of the cow were taken during or immediately after slaughtering. Most of the samples had to be deep-frozen and were thawed before measuring. We performed exclusively mechanical procedures to separate complex structures to get samples in correspondence to their anatomical sites and their physiological function (1,2). The activities of Cs-137 and K-40 were determined simultaneously by gamma counting with the help of semiconductor-detectors and the usual software. Marinelli-beakers or small cylindrical containers of perspex were used as containers for the samples. Some samples containing fat and vascular systems were molten to obtain the activities of pure fat. Other samples were cleaned with water to get the activities of the pure parts.

EXPERIMENTAL RESULTS

All activities are related to November 14, 1992, the day of slaugthering. Values for the specific activities, for the activities and their ratios are presented with a confidence level of 2σ.

Muscle tissues

Table 1. Highest specific activities in muscle tissues.

	Spec. activ. Cs-137 (Bq/kg)		Spec. activ. K-40 (Bq/kg)
M. masseter	389.5 ± 20.7	M. semitendinosus	129.2 ± 6.2
M. gracilis	381.8 ± 6.6	M. iliopsoas	122.9 ± 4.5
M. pterygoideus	371.5 ± 33.3	M. longissimus dorsi (b)	122.2 ± 4.5
M. temporalis	354.6 ± 31.8	M. gracilis	121.8 ± 3.0
M. obturator	352.9 ± 22.3	M. splenius	120.6 ± 10.5
M. semitendinosus	352.9 ± 14.1	M. latissimus dorsi	119.0 ± 6.6
M. glutaeus	352.5 ± 6.5	M. triceps brachii	116.1 ± 4.1
M. iliopsoas	352.5 ± 8.7	M. quadriceps femoris	115.7 ± 3.5

Table 1 shows the highest specific activities of Cs-137 and K-40 observed in the muscle samples. The samples were measured without any special preparation and contained pure muscle tissues, tendons and remainders of fat. The values of the specific activities of Cs-137 and K-40 range from 213.0 Bq/kg (M. levator labii maxillaris) to 389.5 Bq/kg (M. masseter) and from 62.8 Bq/kg (M. buccalis) to 129.2 Bq/kg (M. semitendinosus), respectively. In all muscles of the head we found low specific activities of K-40, whereas high specific activities of K-40 were observed in the important edible muscle tissues. These tissues contain only less tendons.
High values of the ratio of specific activities of Cs-137 and K-40:
M. masseter > M. pterygoideus > M. temporalis > M. zygomaticus > M. buccalis.
All these muscles are located in the head.
Low values for this ratio:
M. cutaneus maximus < M. extensor digitorum lateralis < M. splenius < M. extensor digitorum communis < M. extensor carpi radialis.

Pure muscle tissues
These samples consisted of pure muscle cells, connective tissues were separated. Highest values of Cs-137 were found for M. masseter and M. gracilis: 415.0 ± 37.1 Bq/kg and 408.6 ± 10.9 Bq/kg, respectively. The specific activities of K-40 showed a moderate variation.

Tendons
Table 2. Specific activities of Cs-137 and K-40 in cleaned (washed) tendons.

	Spec. activ.Cs-137 (Bq/kg)	Spec. activ.K-40 (Bq/kg)
M. gastrocnemius	72.7 ± 8.7	25.8 ± 4.0
M. longissimus dorsi (f)	88.8 ± 10.8	25.8 ± 6.9

It was observed that flat-spread tendons accumulate more Cs-137 than tendons of other compact shapes.

Fat
The samples contained fat and vascular systems. The specific activities of Cs-137 in fat, correspond roughly to the activities of Cs-137 in the pure muscle tissues. In the rank of the specific activities of K-40 structures with great contents of pottassium (Ligamentum nuchae, Glandulae suprarenales) are included. Some samples of fat were molten and then the activities of pure fat were measured. The activity levels of Cs-137 and K-40 of the pure fat were found to be at the detection limits of the spectrometers (0.5 Bq - 1.0 Bq).

Lymph nodes
The highest specific activities of Cs-137 and K-40 were found in the lymph nodes of the head: (326.2 ± 39.7) Bq/kg and (89.5 ± 29.8) Bq/kg, respectively. Rank of the specific activities of Cs-137 in lymph nodes:
Head > Muscle tissues > Rumen > Kidneys > Udder > Mesenterium commune.
The specific activities of K-40 in different lymph nodes are within the confidence limits and may be considered nearly constant.

GI-tract
Values of the specific activities of Cs-137 and K-40 in all digestive organs are summarized in a previous work (3).

Organs

Data in Table 3 are valid for the activities of the pure samples without connective tissues.

Table 3. Specific activities of Cs-137 and K-40 in selected organs.

	Spec. activ.Cs-137 (Bq/kg)	Spec. activ.K-40 (Bq/kg)
Heart	196.2 ± 11.8	77.4 ± 7.5
Lung	135.4 ± 4.3	63.0 ± 3.2
Liver	194.1 ± 5.4	102.9 ± 4.2
Gall bladder	43.8 ± 2.2	25.3 ± 2.0
Pancreas	274.6 ± 24.6	106.3 ± 13.0
Kidneys	301.6 ± 10.9	62.1 ± 3.7
Spleen	217.0 ± 13.8	111.5 ± 10.0
Skin	35.8 ± 0.4	20.0 ± 0.4

Total Activities of Cs-137 and K-40

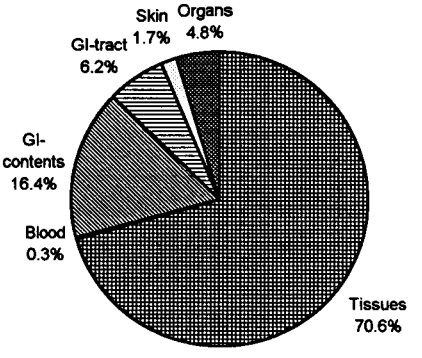

Fig. 1. Percentage of the activities of Cs-137 in different compartments.

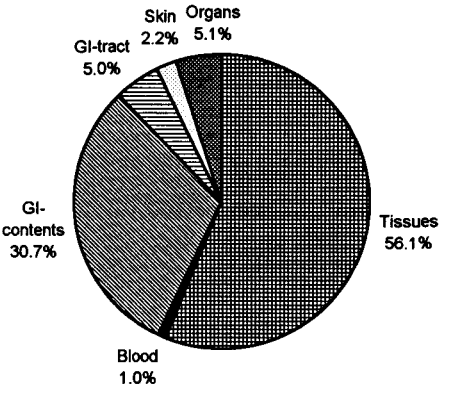

Fig. 2. Percentage of the activities of K-40 in different compartments.

The compartment "tissues" includes muscle tissues and connective tissues.. Total mass and total activities of Cs-137 and K-40 (without the compartment "GI-contents") were estimated to be 302 kg, 50.1 kBq and 21.7 kBq, respectively.

ACKNOWLEDGEMENT

The authors are grateful to Univ. Prof. Dr. Hj. Müller, Reaktorinstitut Graz, for his effective help and substantial financial support.

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RADIOACTIVITY DURING THE FIRST 5 YEARS AFTER THE CHERNOBYL ACCIDENT AND COMMITTED EFFECTIVE DOSES

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INTRODUCTION

The Chernobyl accident from 26th of April 1986 has had an impact on radioactivity in Romanian territory. Consequently, since second part of 1986, a long-term survey on effects regarding radioactivity of environmental and on health of the population was established in Romania.

METHOD

Data on radioactive contents of water and of food (milk and dairy products, meat, bread, vegetables and fruits) reported by the Radiation Hygiene Laboratories from Arges, Bacau, Brasov, Caras-Severin, Cluj, Constanta, Dolj, Galati, Iasi, Mures, Maramures, Prahova, Sibiu, Suceava, Timis and Bucharest were processed by the Institute of Hygiene, Public Health, Health Services and Management - Bucharest, using an INDEPENDENT computer. Knowing the annual consumption of the population from Romania and applying appropiate dose conversion factors (committed effective dose per unit intake), the individual and collective doses for the period 1987 - 1991 have been calculated.

RESULTS

Since 1987 to 1991, the Sr-90 and Cs-137 radioactive content continuously decreased, down to unsignificant levels.

In comparison with 1986, the Sr-90 content was maximum in some vegetables (in 1987, up to 0.98 Bq/kg) and it was minimum in drinking water (0.03 Bq/l).

The Cs-137 content was maximum in some dairy products. It decreased from about 62 Bq/l in 1987 to 5 Bq/kg during the following years (1).

The resulted individual and collective doses due by Sr-90 and Cs-137 content in drinking water and foodstuff are presented in figures 1 and respectively 2.

CONCLUSION

For an adult, the individual dose decreased from 0.331 mSv in 1987 to 0.018 mSv in 1991, mainly due to Cs-137 content. As Cs-137 content decreased, the contribution to the committed dose from Sr-90 increased.

For the whole interval (1987-1991), the resulted collective dose for the population of Romania due by ingeration of drinking water and foodstuff and by their content in Sr-90 and Cs-137 was 11,298 man·Sv, 62,7% being done by the radioactivity level from 1987.

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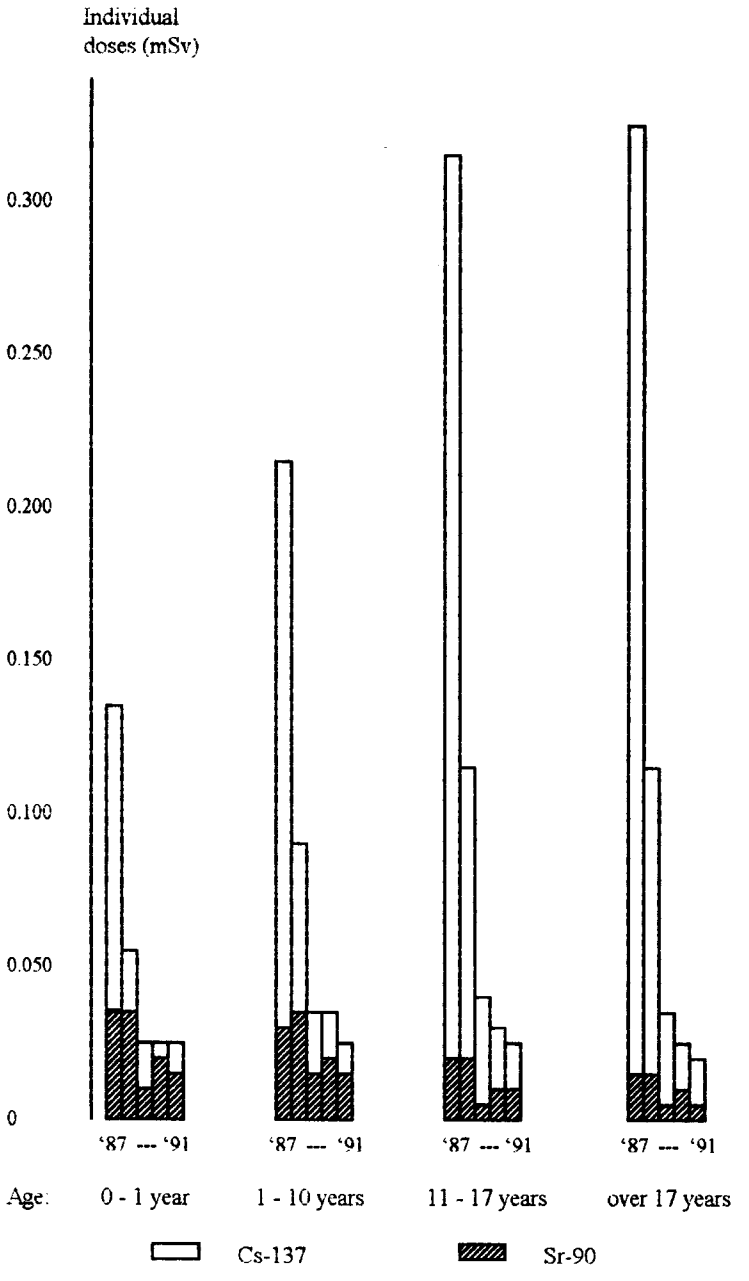

Figure 1. Individual committed effective doses for the Romanian population after Chernobyl.

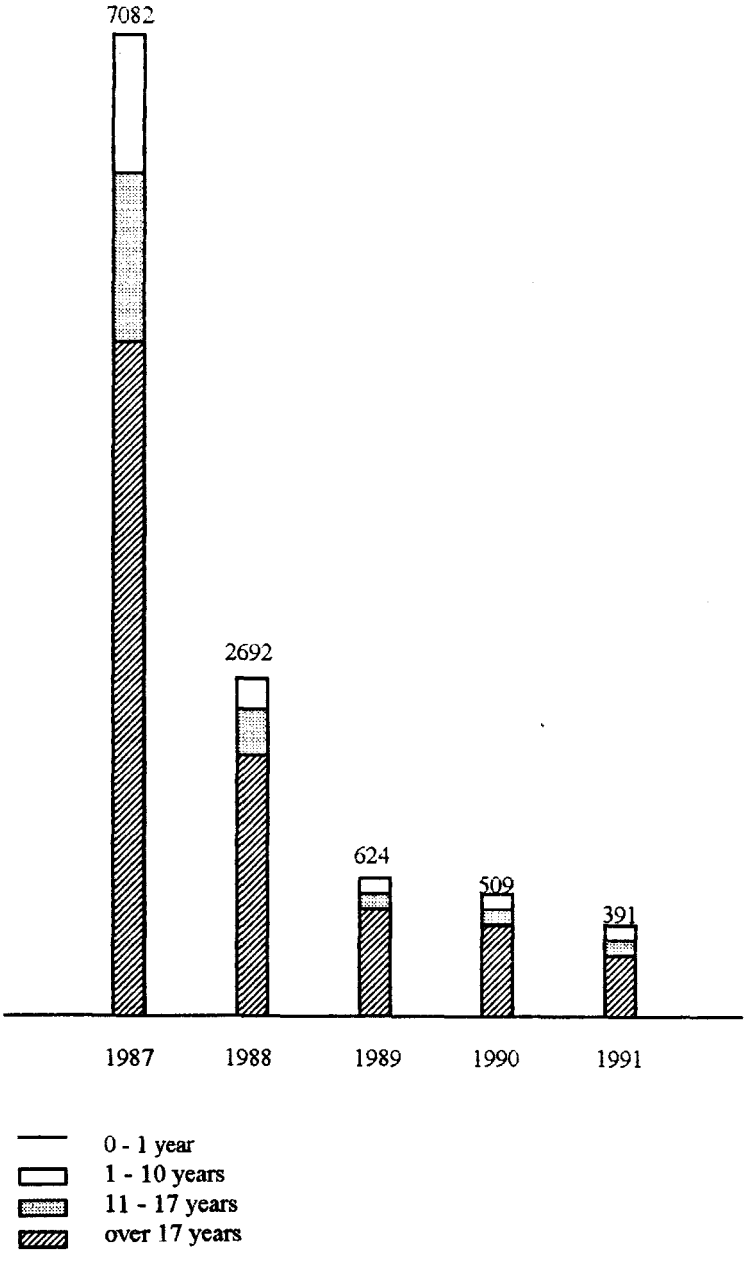

Figure 2. Annual collective commited effective doses after Chernobyl, for the Romanian population (man.Sv)

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PAPER TITLE Potential transfer of radiocaesium to man from Swedish forest ecosystems

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 (see page 7)

ABSTRACT (See instructions overleaf)

Transfer of ^{137}Cs to man from the forest ecosystem is high in Sweden. The main pathways are soil-plant/mushrooms-game animals-man, soil-berries-man and soil-mushrooms-man. Ericoid plants show high uptake of ^{137}Cs and are very important fodder plants for some game animals such as moose and roe deer. Even higher levels of ^{137}Cs is found in fruitbodies of fungi and mushroom picking is very common in Sweden. Mushrooms are also very important as fodder for game animals particularly for roe deer. We have determined transfer parameters for the various transfers and by using them, the mean ground deposition of ^{137}Cs in Sweden or in specific regions of Sweden and data of production and consumption of various food products from the forest ecosystem the potential transfer of radiocaesium to man has been calculated.

The ^{137}Cs levels in moose and roe deer has not decreased from 1986 to 1994 indicating a very long effective ecological halflife of ^{137}Cs in the forest ecosystem. We use the physical halflife as an estimation of the effective ecological halflife. The potential transfer of radiocaesium to man by game animals - moose and roe deer - correspond to a time-integrated dose commitment of about 2,000 manSv i Sweden. The corresponding value for berries is 1,500 manSv. Only rough estimation of the corresponding value for mushroosm will be presented due to lack of knowledge about the consumption pattern.

The conclusion is that a major fraction of the potential doses to the Swedish population due to the Chenobyl fallout will be due to transfer of ^{137}Cs from the forest ecosystem.

POSTER No.: ***P43-18***
POSTER SESSION No.: ***PS-4b***

ANALYZING Cs-137 AND Cs-134 CONCENTRATION IN MOSSES SAMPLES IN CAMPANIA REGION - ITALY AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

The Chernobyl accident caused an exceptional relapse of radioactive material. The consequent fall-out it in several European countries, including Italy (1).

Term and height of aeriform effluent radioactivity admitted into the atmosphere, the distance of place of origin and the variability of meteorologic qualifications existence (wind, rain, etc...), determined the level of contamination of radioactivity in the air with a consequent deposition of soil much variable depending on space and time.

Following the Chernobyl accident the situation of artificial deposition in Italy is considerably changed. In order to obtain arrange a new map of contamination of radioactivity in the soil. All over italian territory has been organized radioecological campaign to arrange a new zero point (2).

The Central Laboratory of Italian Red Cross (L.C. - CRI) is part of the National Network for the monitoring of the environmental radioactivity in Italy co-ordinated by National Environmental Agency (ANPA). The L.C. - CRI took part in the national campaign for the monitoring of the environmental radioactivity to evaluate the deposition of artificial radioactivity on Italian territory (3). Fig.1.

In Italy to obtain the radioecological map the gamma spectrometry technique is used. In fact this technique is more sensitive and is answering several kinds of information. The gamma radioactivity on the place is a more appropriate method, in reality isn't use due to the lenght of time.

In this work, in particular, living organism by *bioindicators* have been used as a mosses (bryophytes). The experience carried out at national and international level demonstrated that mosses can be used to map the radioactive fall-out (4). Studies were also performed on mosses which appeared to have some advantages with respect to lichens.

The mosses are organisms made up chlorophyll and therefore photosynthesis. Mosses generaly live in environmental sub-air, they have not roots and do not have a vascular system to allow the transport of material from substratum for diffusion and to absorb atmospheric water.

The present work was orientated to use specific types of mosses which act as a passive trap to collect radioactive particules and mosses are distributed in all mountain ecosystems whether in Italy or in other countries.

The monitoring of gamma radioactivity for Campania region has been delegated to the Section of Environmental Radioactivity of the L.C. - CRI. To this task the cooperation with the Ministry of Agricolture and Forestry and together with the help of the Department of the State Forestry Corp of Campania.

This work presents the data of the surveys carried in 1993-1994. The mosses were picked up in four different locations of Campania region and measured by means of ray spectrometry, according to a national standard procedure, to evaluate the content of Cs-137 and C-134. The content of K-40 was also measured.

MATERIALS AND METHODS

In this paper the measures concern n.44 samples of mosses; this mosses have been collected in four States Forest Corp posts in the provinces of Salerno, Caserta, Napoli. In particular, the localities are: Laviano, Piaggine, S.Gregorio Matese and Boscotrecase.

The species of the mosses picked up are *Hypnum cupressiforme* and *Homalotecium lutescens.*

The samples were gathered in late autumn (November - December) only on the rock and show horizontal growth, with formation of carpets thickness between 2 - 4 centimetres.

The samples have not received any chemical treatment and after drying up in the air have been utilized.

The samples underwent analysis of gamma ray spectrometry: the measures have been carried out using high purity Germanium detector (Hp), relative efficiency 25 %, in to a well of the lead 10 centimetres thick. Radionuclide concentrations were determined by automatic peak fitting using a spectroscopy applications software package (PCA II - GDR). Detector was calibrated using a calibration standard (359 RLM-ENEA) in the liquid form (Marinelli 500 cc) through participation in national intercomparison. For the samples, the counting error, expressed with an accuracy of ±1%. Consequently the samples placed in Marinelli Beakers in the volume of 500 cc. The count time was 72.000 sec (20 hours).

The concentration of mosses is reported on the 1st July 1993 and is expressed as Bq/m^2.

The counting time which is ratherlong allowed to obtain a standard deviation on peak of height intensity of Cs-134 of ± 10%.

Fig. 1
from : "Mapping of radioactive fallout using mossese as bioindicators"
International Conference Taormina (Italy) October 1993; pagg. 215-226 (3)

RESULTS

The table n.1 shows the average concentration of the Cs-137 and Cs-134 on relating to 44 samples of mosses examined in four places for cathering. The values of average concentration of mosses in the Campania region are: 374 Bq/m² of Cs-137 , 16 Bq/m² of Cs-134 and 151 Bq/m² of K-40. Tab.1

Tab. 1
Cs-137 - Cs-134 and K-40 activity in mosses

LOCALITIES	*Cs-137 (Bq/m²)*	*Cs-134 (Bq/m²)*	*M k-40 (Bq/m²)*
S.GREGORIO (TORNARE)	*505*	*22*	*175*
LAVIANO (CACARI)	*375*	*16*	*134*
PIAGGINE (COZZO)	*312*	*13*	*1163*
VESUVIO (TRECASE)	*303*	*14*	*133*
REGIONAL	*374*	*16*	*151*

Tab. 2
Cs-137and Cs-134 activity in mosses in differents regions of Italy (1992-93)

REGION	*Cs-137 (Bq/m²)*	*Cs-134 (Bq/m²)*
ALTO ADIGE	750	50
CAMPANIA	530	30
E.ROMAGNA	940	70
FRIULI V.G.	5400	380
LAZIO	500	35
LOMBARDIA	6000	400
PIEMONTE	5800	400
TOSCANA	330	20
TRENTINO	3500	250
UMBRIA	350	25
VALDAOSTA	1300	90
VENETO	3000	210

CONCLUSIONS

In this paper the results of the concentrations of the Cs-137 and Cs-134 in the mosses confirm an uniform distribution of the radioactivity post Chernobyl in the regional territory. The reluts are in accord to the previous date (1992-1993). Tab.2

The concentration of K-40, in all samples examined, has a costant value.

Currently we are investigating the possibile correlation between the concentration of the Cs-137 and Cs-134 in the soil and in the mosses of the same spots (5-6).

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FOLLOW UP OF THE CHERNOBYL ^{137}Cs CONCENTRATION IN BELGIUM BY MEANS OF *IN-SITU* GAMMA SPECTROMETRY

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INTRODUCTION

On Friday, May 2, 1986 a cloud containing radioactive particles from the Chernobyl accident reached Belgium during the morning. At noon, an activity in the air of 58 Bq/m^3 was measured in Gent. Together with short living isotopes (^{131}I, ^{132}Te, etc.) a concentration of 5 Bq/m^3 of the long living ^{137}Cs was registered (half live 30 y). One day later, on Saturday, May 3, the air activity was already reduced to a few percent of this peak value.
The aim of this research is to evaluate the situation concerning the residual radiocesium concentration in different parts of Belgium nearly 10 years after the Chernobyl accident by means of *in-situ* γ-spectrometry.

EXPERIMENTAL PROCEDURE

The *in-situ* γ-spectrometry technique using efficient high resolution Ge-detectors allows a fast, accurate and sensitive determination of radionuclides in the soil. This method is described by Beck et al. (1) and more recently by Miller et al. (2, 3). The ICRU report nb. 53 (4) (December 1994) gives a complete review on the subject.
In this work, the measurements were performed with two different measuring chains, one of the University Gent (RUG) and the other of the SCK-CEN Mol. The RUG detector is a 20.2 % p-type (energy resolution 1.7 keV) with a normal 30 litre dewar. The SCK chain uses an n-type HPGe coaxial detector (resolution 1.83 keV and efficiency 10%) mounted on a small multi-attitude cryostat. Both spectrometers were calibrated completely independent and intercompared during two in-situ measurements at locations in Mol and Tihange. All results are in good agreement within the total experimental error of about 15%.
Laboratory analysis at the SCK-CEN of soil samples confirms the in-situ data. Details about the calibrations and the intercomparison are given in internal reports (5, 6).
After some preliminary measurements in Gent, Mol and the Ardennes, a systematic survey of the Belgian territory (30507 km^2) by in-situ gamma spectrometry has been carried out. About 50 measurements, equally distributed across the country were executed (the western part by the RUG-team and the eastern part by the SCK-CEN group). For all measurements, pastures that we know as undisturbed during the last 10 years were selected. In the calculations of the ^{137}Cs concentration we use a depth profile with relaxation parameter α = 20 m^{-1}; for the natural isotopes we assume a homogeneous distribution with α = 0 m^{-1}.
This relaxation parameter is rather critical for the calculations of the ^{137}Cs concentration; with α = 33.33 m^{-1} as used in earlier measurements (5,6) the values given in fig. 2 should be reduced by a factor 1.356. To improve our knowledge on the actual ^{137}Cs distribution in the soil, further investigations based on spectrometry of soil samples are in progress. Our actual choice of α = 20 m^{-1} (relaxation length 5 cm) complies with the data for aged fallout (table 3.5 in (4)). Fig. 1 gives a typical spectrum of a measurement with the RUG-detector in a high activity region; one can easily see the 662 keV ^{137}Cs line together with the 1461 keV ^{40}K line and the typical lines of the natural U and Th series. A measuring time between 30 and 60 minutes allows an accuracy better than 15% at the actual remaining ^{137}Cs concentrations from Chernobyl in Belgium (typical values from .5 to 5 kBq/m^2). Even a quick 10 minutes test measurement can give a good indication on the concentration.

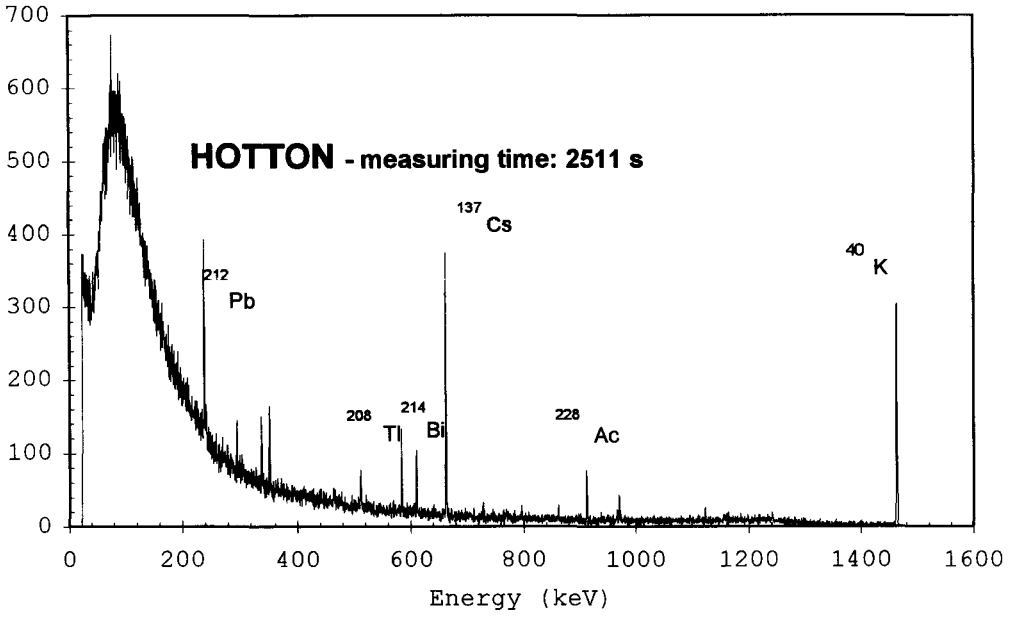

Figure 1: typical in situ gamma spectrum; ^{137}Cs concentration is 4600 Bq/m^2

RESULTS AND INTERPRETATION

The map (figure 2) summarises the results of our survey for ^{137}Cs; all values are calculated with $1/\alpha = 5$ cm. The results of 3 measuring points (indicated with *) are not available at the moment. The occurrence of regions with higher concentration (a peak value of 4610 Bq/m^2 was obtained near Hotton) coincides roughly with the zones of higher deposition as mentioned by Govaerts et al. (7). Some discrepancies between neighbouring points can be explained by the variations in the use of the grassland. Although our careful selection of the measuring sites, it can not be completely excluded that some pastures have been ploughed or modified in some way, disturbing the natural distribution of the Cs. A new control measurement at some locations will give a final value in some cases. At the moment, we are still studying the correlation of the actual concentration with the deposition, the rainfall and other meteorological conditions in the May 1986 period. More detailed information will be available on the poster during the conference.

ACKNOWLEDGEMENTS

The RUG-team is indebted to Prof. H. Dumont (Ecology Laboratory, RUG) for transportation facilities. The SCK-CEN authors wish to thank all their colleagues of the γ -spectrometry group who participated in the measurement campaign.

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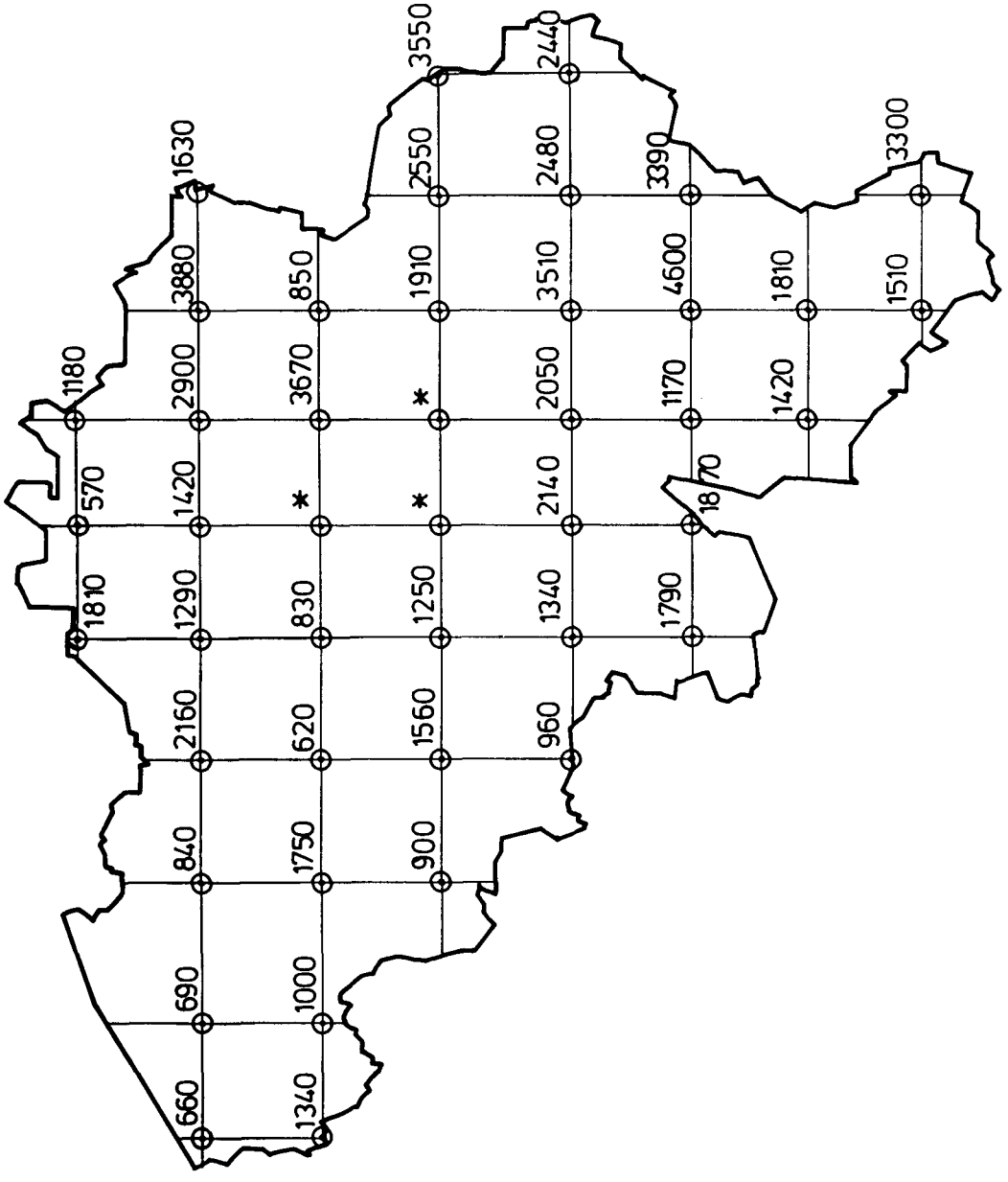
Figure. 2: map of Belgium with the ^{137}Cs concentrations (in Bq/m^2) at the measuring points



IRPA9
1996 International Congress on Radiation Protection
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Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE Findings of the First Comprehensive Radiological Study of the Republic of the Marshall Islands

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 (see page 7)

ABSTRACT (See instructions overleaf)

The Marshall Islands was the site of the United States atomic weapons testing program in the Pacific. Sixty-six atomic weapons were detonated in the island country during the years 1946 through 1958. For several decades, radiological monitoring by the U.S. Department of Energy (and its predecessor agencies) has been conducted on the two test site atolls and neighboring atolls. However, 70% of the land area of the 1200 islands was never systematically monitored prior to 1990. For a five year period (1990-1995), the Government of the Republic of the Marshall Islands has undertaken an independent program to assess the radiological condition throughout it's 29 atolls. Over 1300 in-situ high resolution gamma spectrometry measurements have been made on more than 400 islands at an average spatial density of one measurement per 0.15 km^2. Native foods including coconuts and other tropical fruits have been sampled as well as more than 200 soil profiles and more than 800 surface soil samples. The fruits and soil samples were analyzed for gamma emitters with an emphasis on ^{137}Cs, the surface soils were also analyzed for $^{239,240}Pu$. External exposure-rates from ^{137}Cs vary, depending on location, from 1 μSv/year to over 2 mSv/year. Estimates of possible internal doses from ^{137}Cs, depend on location and dietary habits, and can range from 10 μSv/year to 50 mSv/year. The objective of this presentation is to report the first comprehensive summary of the radiological conditions throughout the Marshall Islands.

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PAPER TITLE Radiological Consequences in the Environment Following a Venting-Procedure at PWR and BWR

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3 ..... (see page 7)

ABSTRACT (See instructions overleaf)

The release of radioactive material during venting procedures of the containment of BWR and PWR after a postulated melting of the core results in an exposure of the public in the environment. This exposure can be significantly reduced by disaster control. The methods and the extend of the disaster control have to be determined with suitable forecast-models based on measured activity-concentrations in the atmosphere of the containment. The dose caused by the release of radioactive iodine is caused by conservative release-assumptions. The release of radioactive noble-gases and the resulting dose hereby can be reduced significantly by the installation of delay-systems. The instruments to optimize venting procedures for the environment from the viewpoint of radiation protection are shown and discussed.

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PAPER TITLE Assessment of Dose Consequences from Iodine-129 and Other Radionuclides by Isotopic Equilibrium Analysis

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MAJOR SCIENTIFIC TOPIC NUMBER 4.3. (see page 7)

AB: Application of deterministic calculational modeling indicates that the most significant radiological impacts from low-level radioactive waste (LLW) disposal could result from long half-life components such as: I-129, Tc-99, U-238, Pu-239, etc. However, alternative assessment methods have demonstrated that this may not necessarily be the case. For example, application of isotopic equilibrium analysis (IEA) indicates that presence of stable iodine (I-127) in the environment would necessarily cause isotopic dilution of I-129 released from a waste disposal site. Since the total iodine content in the human body is regulated by homeostatic processes, and since the I-129 /I-127 ratio in the human body must equillibrate to that in its environment, any significant dose consequence from I-129 would be precluded except under very unusual conditions. Other evidence indicates that in the cases of U-238 and Tc-99, radiological consequences may be minor relative to the chemical toxicity effects resulting from ingestion of these materials. Alternative evaluation techniques such as IEA and probabilistic analysis have been applied in assessing potential impacts from the Savannah River Plant LLW burial grounds. The results can be used to augment and elucidate predictions determined by calculational modeling.

EXPERIENCE AND RESULTS OF COOPERATION IN RADIATION PROTECTION BETWEEN SWEDEN AND EAST EUROPE COUNTRIES

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INTRODUCTION

From 1991/92 until July 1995 the Swedish Government has allocated SEK 55.1 million (about 10 M $) for cooperation in radiation protection with countries in Central and Eastern Europe, particularly the Baltic states (Estonia, Latvia, Lithuania) and Russia but also to less extent so far Byelorussia, Poland, Ukraine and others. The Swedish Radiation Protection Institute (SSI) is in charge of this program, which is often referred to as Project Radiation Protection East. In addition, SEK 8 million have been reserved for SSI activities promoting safety during decommissioning of the former Soviet naval reactors training center at Paldiski, Estonia and the subsequent transfer of responsibility for the site from Russia to Estonia (the PIERG project).

Radiation protection covers a broad area of the society. Already at an early stage it was realized that supporting cooperation was necessary and of interest over the whole area. Therefore extensive discussions on priorities between parties concerned have been and are necessary. The choice of projects is based on a "Protocol", that is the eventual result of priority discussions signed by representatives of both countries (bilateral agreements).

Most of the projects within "Radiation Protection East" mean a longterm build-up of radiation protection in a broad sense. Some projects concern new problems, other projects concern problems that the country inherited from the earlier Soviet Union. Other projects may concern acute problems as e.g. lost strong radioactive sources, radioactive contaminations etc.

Even if most of the projects within Radiation Protection East are bilateral there is a substantial international input and cooperation. There are many projects that are run in cooperation with the IAEA, OECD/NEA, EU and together with other countries particularly the Nordic countries.

There is an increasing need of financial support as the many preparatory phases are passed and the follow up efforts are more expensive. The other party will also gradually be more susceptible for more advanced cooperation.

THE OBJECTIVES AND TERMS OF REFERENCE

The idea of activities within the Project Radiation Protection East is the following:

Efforts adjusted to the need of the country to improve the radiation protection and make it more effective and to solve acute radiation protection problems.

The cooperation means support to these countries but it is also of mutual interest. Uncontrolled releases of radioactive material and bad handling of radioactive sources in a cooperation country might in some circumstances mean a potential risk to health and environment also in Sweden. Therefore, a successful cooperation within this project might have a positive influence on radiation protection in Sweden as well. Furthermore, the cooperation is instructive for both parties and is very stimulating and productive for development of mutual understanding and future cooperation.

A condition that is to be fullfilled in the choice of a project is that it comes up to one or several of the following requirements:

- it is fundamentally important for the cooperation country
- it is expected to lead to decreasing unnecessary radiation doses in that country
- it is of interest for Sweden and the Baltic Sea region
- it means education and training of relevant persons of the cooperation country
- it has a common interest for the two countries in the areas of research or information.

Requirements are set on both parties - both Sweden and the cooperation country as e.g. preparatory domestic work, to inform, motivate, engage and fill concerned persons with enthusiasm. Share of the costs of the project is another way to show interest and create partnership.

SOME PREREQUISITES FOR COOPERATION

The cooperation is individually decided for each country after discussions and negotiations, the results of which are manifested in the Protocol. There are some components of the infrastructure of the cooperation country that are essential for continued cooperation in radiation protection and therefore have the highest priority in the first phases of the cooperation.

They are:

- radiation protection law and legislation that define the national actors, devide responsibilities between them and defines their duties and rights
- radiation protection authority(ies) with functioning organisation, competent personnel and modern equipment
- a national strategy for handling, storage and disposal of radioactive waste
- a national monitoring system with measurement stations as a part of an early warning system, coordinated with other countries
- a laboratory for measurements on environmental samples and a national program and strategy for environmental control in general for the purpose of radiation protection.

During the few years of cooperation within the Project Radiation Protection East some of the important components mentioned above have now been established largely as a result of the cooperation. Points of contact have been established, problems common for several countries have been identified and can be treated in a more rational and costeffective way.

Completed laws need directions, equipped laboratories need supplementary training of personnel and programs for sampling; when the strategies for radioactive waste are decided there is a need to realize the strategies, etc. There are many examples of continued and expanded cooperation e.g. concerning the radioactive waste management and disposal in Russia and the Baltic states, the decommissioning and dismantling of the earlier Paldiski submarine training center, the Sillamäe tailing ponds in Estonia and the emergency preparedness plans and training.

THE PROJECT MANAGEMENT

The projects are devided into the following main categories:

1 *Upgrading of national authorities, 18 projects*
Legislation, organization, information, computer support, etc.

2 *Emergency preparedness, early warning, 7 projects*
Organization, education, dose predictions, gamma monitoring, communication etc.

3 *Nuclear power and research reactors, 4 projects*
Safety at work, releases, organization and other plant related issues not dealt with elsewhere.
4 *Instrumentation, 12 projects*
Laboratories, mobile and stationary measurement equipment, education etc.
5 ~~*Decommissioning, waste, environmental control, 24 projects*~~
National waste strategies, plant specific programs (i.e. Paldiski, Sillamäe), radiological clean-up, measurement programs, education etc.
6. *General radiation protection, 40 projects*
Medicine, radiological protection of workers, dosimetry, radioactive lighthouses, radon, non-ionizing radiation etc including research support.
7. *Other project costs*
Projects not covered by any of the categories above.
8. *Project management and administrative support*
Includes resources needed to plan, organize and report on the various projects.

That means more than 100 projects since 1992. Some have been finished and some are still in the planning stage. Normally, radiation protection specialists of the staff of SSI are project leaders and also take active part in the projects. The management manpower of the Project Radiation Protection East is in total about 2 personyears per year.

Two examples of cooperation projects are given.

Emergency preparedness, early warning

Automatic gamma stations

Expansion of the Early Warning System in Estonia, Latvia and Lithuania with a central unit and four automatic gamma stations in each country. The expansion was made during December 1993 by Finnish ALNOR (RADOS). During 1994 the system was tested and found to function properly (although some stations had problems with the telecommunications). The test period is thus finalized and the one year warranty period started in 1994-12-01. Bilateral agreements were made during Jan-Feb 1994. The systems were formally handed over during 1995.

Decommissioning, waste, environmental control

National waste strategy (in Lithuania)

A study is made to present a background for decisions on a national waste strategy for handling all kinds of radioactive waste - from the Ignalina NPP, from hospitals, research establishments, industries etc. Parties involved are identified and their roles, mandates and liabilities are assessed together with the legal framework. Available methods for transport, conditioning and intermediate and final storage of the waste are assessed. Problems around the Ignalina waste are specially considered. The report is presented to the Lithuanian Minister of Energy. Additional follow-up projects with high priority are expected.

CONCLUSIONS

The cooperation between Sweden and the countries in the eastern part of Europe has developed well. Good contacts and exchange of information with other western countries as regards planned and ongoing cooperation with the countries in eastern Europe is essential to avoid duplication and gaps. The cooperation has already lead to some improvements of the conditions for good radiation protection in these countries but there is still much to do. The cooperation has also a mutual, stimulating effect upon those persons and organisations involved in it and it opens channels for further contacts and cooperation. It will therefore continue.

Cs^{137} AND Sr^{90} DIETARY INTAKE AND URINARY EXCRETION FOR CHILDREN, AFTER THE CHERNOBYL ACCIDENT

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INTRODUCTION

Since the accident from Chernobyl, an important number of studies were focused on the effects of the accident but, nine years after the accident, we still don't know enough about its impact on public health and environment.

A major problem after the Chernobyl accident was to asses the effects of the irradiation for different age groups, especially for children. Our group measured Cs^{137} and Sr^{90} dietary intake and urinary excretion for children of different ages (between 4 and 12 years), at different time intervals after the accident. From the intake data, a trend of the annually committed effective doses was deduced. The paper presents the dose values for different age groups, as well as the balance of the intake and excretion, given here as the "observed ratio", defined as:

$OR = (\frac{Cs-137_{urine}}{K_{urine}}) / (\frac{Cs-137_{intake}}{K_{intake}})$ for Cs^{137}, and

$OR = (\frac{Sr-90_{urine}}{Ca_{urine}}) / (\frac{Sr-90_{intake}}{Ca_{intake}})$ for Sr^{90}. Both Ca and K are given in grams, while the Cs ans Sr contents are in Bq.

MATERIAL AND METHODS

The group which undergone the study consisted of children, separated in three age subgroups : 4-6 years old, 7-9 years old and 10-12 years old . These children were living in the kindergardens and schools hostels. The data for the dietary intake and urinary excretion for the children was compared with similar data for the adult women working as teachers and educators at the schools, and taking their meals together with the children. These women were aged between 24 and 49 years. The food samples were prelevated from the schools' cafeterias. The 24 hour urine samples were prelevated the next day after the prelevation of food samples.

All the measurements were performed by radiochemical methods of analysis [1,2]. After the radiochemical separation of caesium and strontium, the radiometric measurement was performed with a low-level anticoincidence beta counting system, with high efficiency.

RESULTS AND DISCUSSION

During 1986 - 1995, we have selected four time periods in which we have studied the dietary intake and urinary excretion of Cs^{137} at children, for all three age groups, in comparison with the adult women group. We will present here the differences between the first period, (caharacterised by a strong contamination), and the last period of study.

The first period was September 1986 - December 1987. From the point of vue of the ingestion and excretion processes, this was the most complex period. During this period, an intake peak occurred in October 1986 at all the age groups, including the women. After this date, Cs^{137} dietary intake is continuously decreasing, but the slope of the decrease is age-dependent. The OR for this period is presented in Fig. 1.

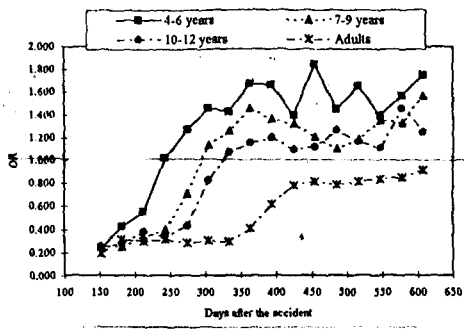

Fig. 1: OR for Cs^{137}, during September 1986 - December 1987

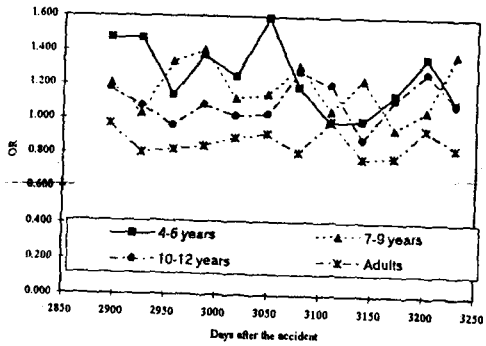

Fig. 2: OR for Cs^{137} during April 1994 - March 1995

At 4-6 years and 7-9 years children, the OR reaches a plateau in April 1987 while at 10-12 years old children the plateau occurs in May 1987. The values of the OR, in the plateau zone, are also different, the value for 4-6 years children being the highest, and the value for adults being the lowest. The OR for the last period (April 1994 - March 1995) took values situated in the plateau area , for all age groups, as can be seen in Fig. 2.

As it can be seen, within the same age group, the OR tends to decrease as the time goes by; for example, at the 4-6 years old children, the OR decreased from an average value of 1.643 in 1988 - 1989 to 1.256, nine years after the accident.

During 1986 - 1994, we have selected three time periods in which we have studied strontium-90 dietary intake and urinary excretion for the same three groups of children compared with adult women. The first interval of study was September 1986 - September 1988. The maximum strontium intake in children was 1.198 Bq/(gCa.day), reached in October 1986. The observed ratio, has different values for the two age groups studied in the first period, as can be seen in Fig. 3.

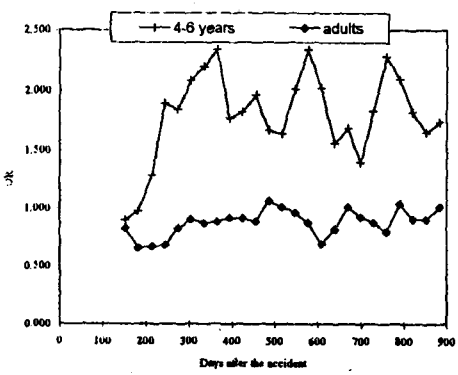

Fig. 3: The OR for Sr^{90}, during September 1986 - September 1988

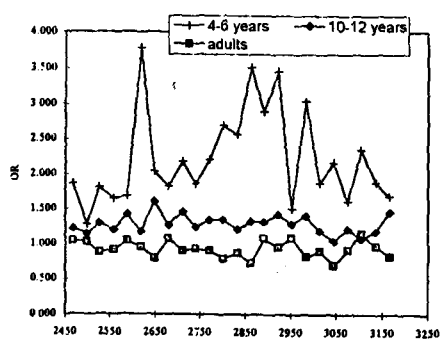

Fig. 4: The OR for Sr^{90}, during January 1993 - December 1994

The last period of study - since January 1993 until December 1994 - included measurements of Sr^{90} dietary intake and urinary excretion for two children groups - 4-6 years and 10-12 years - and the group of adult women. The OR evolution is shown in Fig. 4. This period is caracterised by low values of Sr^{90} dietary intake and urinary excretion. The average amount of ingested strontium for this period was 0.073 Bq/(gCa.day) for 4-6 years children and 0.102

Bq/(gCa.day) for 10-12 years children. As it can be seen from Fig. 4, Sr^{90} urinary excretion had higher values than the dietary intake, for both children groups (OR>1), while ce for adult women the two values are roughly equal (OR~1).

The intake values for Sr^{90} and Cs^{137} resulted in the effective doses committed annually. In order to calculated these doses, we used the dose factor recommended by ICRP 67 [3]. The effective doses committed due to Cs^{137} intake, in each time interval studied, are presented in Fig. 5. As it can be seen, the lower the age, the lower the dose and, for all age groups, the doses are decreasing as the time goes by. The doses committed due to Sr^{90} intake are presented in Fig.6.

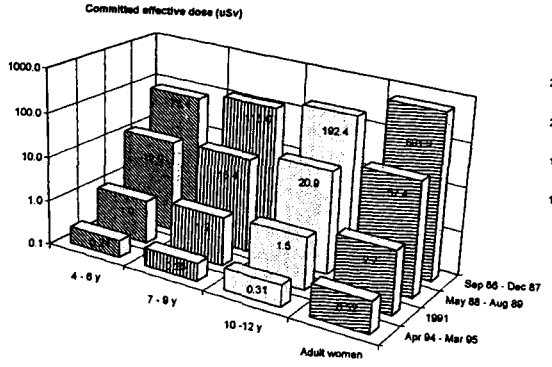

Fig. 5: Effective doses committed due to Cs^{137} intake

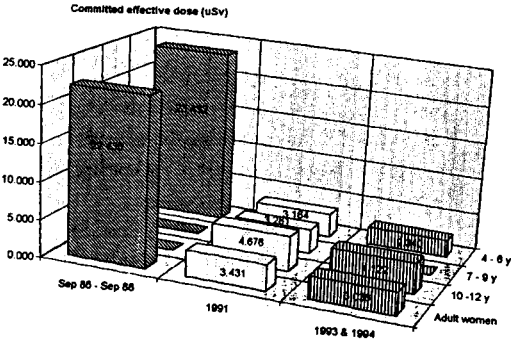

Fig. 6: Effective doses committed due to Sr^{90} intake

These doses were obtained by suming the doses to red marrow and bone surface (Toader and Vasilache, 1995 [4]). In this case, there is no more linear age dependence of the dose. Fig. 6 seems to indicate that the highest doses were committed by the 10-12 years old children, but the data are not suficient to consider this hypothesis as sure. However, the doses committed due to Sr^{90} are decreasing as the time elapses, as the doses due to Cs^{137} do.

CONCLUSIONS

The results of this study allowed us to estimate the effective doses committed due to Cs^{137} and Sr^{90} intake. The effective doses committed by the children , due to Cs^{137} intake, are lower than the doses committed by the adults. As in the case of the doses committed by the adults, the doses committed by children decrease as the time goes by. In the case of the doses committed due to Sr^{90} intake, the values for children are comparable to the values for adults.

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ASPECTS OF THE RADIOACTIVE CONTAMINATION OF FOOD AND HUMANS IN ROMANIA, AFTER THE CHERNOBYL ACCIDENT

Ion Chiosilă, Constantin Milu

Romanian Society for Radiological Protection

INTRODUCTION

The radioactive contamination of the environmental factors in Romania, was shown four days after the accident, by a massive increase in the radioactivity of the air, deposits, soil, and vegetation. It was found that some areas were more contaminated, due to the currents of air that favoured the movement of the masses of contaminated air, and to the fall-out driving the radionuclides toward the ground. Areas with low levels of contamination were identified in the western part of the country, while the most contaminated areas were the mountain regions and the areas in the north-east of the country.

Twenty-one radionuclides were identified by gamma spectrometry, to which Sr-90 and Sr-89 were added, that were detected by radiochemical analysis.

The contents of Cs-137 in the atmospheric deposits did not exceed 80 kBq/m^2, which places the level of contamination in Romania below the levels reported in Ukraine and Belarus (over 200 kBq/m^2), or in Sweden (over 100 kBq/m^2)[1,2] .

MATERIALS AND METHODS

The samples of drinking water and foodstuffs were collected from the whole area of Romania, once a month during 1986-1987, and once a trimester between 1988 and 1992[3] .

Cs-137 and Sr-90 were chemically separated out of large volumes of water (100-300 l), then measured by spectrometry and beta radiation of the Y-90, respectively [4,5] .

The samples of foodstuffs, after being weighed and calcinated, were measured by gamma spectrometry, to determine their contents of Cs-137 and Cs-134. From the same calcinated samples Sr-90 was separated, then measured by means of its daughter Y-90.

The gamma spectrometric measurements were performed by means of Canberra analysers with 4096 channels, endowed with Ge(Li) detectors. The beta measurements were carried out with anticoincidence facilities, Nuclear Enterprises (U.K.)[4,5] .

The concentration factors (CF) in the trophic link soil-plants were also computed, as the ratio between the activity of a radionuclide in a mass of 1 kg of fresh plant, and the activity in 1 kg of soil.

The human contents of Cs-137 and Sr-90 was measured by antropogammametry. The human contents of Cs-137 and Sr-90 was also estimated by computation, as a function of the contents of the radionuclides in the ingested food [2,4-6] .

RESULTS AND DISCUSSIONS

The values of the contents of Cs-137 and Sr-90 in drinking water increased during May 1986, but did not exceed 1 Bq/l. During the following months, the activity of the two radionuclides decreased continuously, until, in 1990, it reached values of only 2-10 mBq/l that were also found prior to the nuclear accident.

Throughout May-July, the contents of Cs-137 in cultivated vegetation was of the order of hundreds of Bq/l, but decreased quite rapidly during the second half of the year 1986 and during the following years, until, in 1989, it reached values lower than 1 Bq/kg. When comparing the activity of this radionuclide in wheat grains to its activity in corn grains, the contents of Cs-137 in wheat was higher by at least one order of magnitude than that in corn (up to 500 Bq/kg).

In 1986, the contents of Cs-137 in animal products (milk and dairy products, meat and derivatives, etc.) showed high values, hundreds of Bq/kg of fresh mass. During 1987, the values decreased by at least one order of magnitude, and in 1990 they reached less than 1 Bq/kg. Isolated peaks of increased contents of Cs-137 were detected at the beginning of the year 1987, mainly in dairy products, due to the use of contaminated vegetation that was stocked during the summer of 1986.

The values of the contents of Sr-90 in vegetation and foodstuffs of animal origin were much lower (with one to two orders of magnitude) than those found for Cs-137.

The contents of Cs-137 in daily menu in the southern part of Romania was about 100 Bq/day during 1986 and the first half of 1987, reached values below 10 Bq/day during the second half of 1987 (the highest intake was due to the wheat flour ground from the crop of the previous year), was only 1 Bq/day since 1988, and decreased even further after 1990. The contents of Sr-90 was lower than 10 Bq/day during 1986, and decreased to about 1 Bq/day in 1988. The two radionuclides detected in the menu during 1987 were mainly due to the flour ground from the crop of the previous year. When it was replaced by the flour from the crop of the year 1987, the contents of Cs-137 in the menu decreased to about 2 Bq/day.

The concentration factors of these two radionuclides in plants, computed for the interval 1987-1992, showed values below one (between 0.01 and 0.11), which explains the rapid decrease of the contents of the radionuclides in plants, then in animals and products of animal origin.

The contents of Cs-137 in human body was maximum during the years 1986 and 1987, approximately 10,000 Bq, and continuously decreased since 1988 due to the low intake from food, reaching, in 1991, values below 100 Bq. Sr-90, computationally estimated as a function of the degree of contamination of the ingested food, was below 100 Bq even in 1986, but decreased much slower than Cs-137.

The relatively rapid metabolic process of Cs-137, and the low intake of this radionuclide from food even since the end of the year 1987, resulted in a quite rapid decrease of its contents in human body, to values below 100 Bq in 1991. As for Sr-90, the intake from food was also low starting with 1987, but the much slower metabolic process of this radionuclide resulted in a much smaller decrease of its radioactive contents, as compared to Cs-137.

In Romania, the level of human and food contamination with Cs-137 and Sr-90 was close to the levels reported in other European countries (Hungary, Poland, etc.), and much lower than those found in Ukraine and Belarus [1].

CONCLUSIONS

1. The radioactive contamination of food reached high values for Cs-137 (hundreds of Bq/l or kg), and lower values for Sr-90 (10 to 100 times lower than Cs-137), during the months May-July 1986.

2. The values of the contents of the radionuclides Cs-137 and Sr-90 in water and food decreased rapidly, even during the second half of 1986, and in 1991-1992 they reached the levels recorded for the periods prior to the accident.

3. The concentration factors for Cs-137 and Sr-90 in plants had low values, usually below one, even since 1986, which explains the rapid decrease of food contamination.

4. The evaluation of the human contamination indicated maximum values by the end of 1986, about 10,000 Bq Cs-137, and less than 100 Bq Sr-90.

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Vienna, Austria

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PAPER TITLE Dosimetric Basis for the Risk Analysis of Thyroid Cancer Morbidity in the Bryansk Region of Russia.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 3.1, or 3.5; 4.3

ABSTRACT (See instructions overleaf)

Average thyroid dose in children under three years of the Bryansk region, the most suffered territory of Russia after the Chernobyl accident was about 2 Gy with maximal individual values up to 10 Gy in the most contaminated settlements. Prognosis of medical consequences for population of Bryansk region showed that thyroid cancer would be the most significant effect of people exposure due to the Chernobyl accident. Now about 50 thyroid cancer incidents were registered among children and teenagers of Bryansk region irradiated in 1986.

Thyroid dose estimations for population of the Bryansk region were based on the results of 17 000 measurements of I-131 content in thyroid of inhabitants (10 000 - children and teenagers under 18 year) in May-June 1986, data of questionnairies of the inhabitants about their life-style and food consumption, data of the regular environmental monitoring that has being fulfilled in the Institute of Radiation Hygiene from May 1986 till now.

Dosimetric models used for the reconstruction of group and individual thyroid doses are discussed in the report. Comparison of different parts of I-131 measurement database showed quite satisfactory agreement of dose estimations made independently by different groups of investigators with different equipment. About 20% of measured people got thyroid doses more than 0.3 Gy, 5% - more than 1 Gy. Among children under 8 years 44% and 15% of the investigated children had thyroid doses more than 0.3 and 1 Gy.

Using of this dosimetric data base for a epidemiological study of thyroid cancer morbidity in inhabitants of Bryansk region are discussed in the report. Preliminary estimations of radiation induced thyroid cancer risk coefficients are presented.

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PAPER TITLE Radioecological Characteristics Determining Radiation Exposure of the Population around Mayak Facility

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The paper discusses quantitative estimations of different radioecological factors which determine exposure doses to population residing on the territory contaminated with Sr-90. The internal doses of Sr-90 result from Sr-90 intakes with diet and depend to a great degree on the content of this radionuclide in the food which, in its turn, is governed by many radioecological factors: level of contamination of the territory, distribution of radionuclides, topography, land-use practices. Additional factors influencing Sr-90 intakes with food are chemical composition of the soil (depending on soil type), soil moistu - ring, radioactive surface water run-off, accumulation of dung and ashes from firewood at certain sites, etc. In terms of food it has been established that the main contributor is milk which is due to using hay from contaminated lands to feed the animals. The peculiarities of forage production, the contribution of pastures and haylands and other ecological factors to Sr-90 ingestion are analyzed.

RADIOECOLOGICAL STUDY OF FRESH WATER ECOSYSTEMS INFLUENCED BY THE OPERATION OF NUCLEAR CYCLE FACILITIES IN THE URALS

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Nuclear cycle facilities have been functioning in the Urals for over four decades which resulted in significant impacts on the environment.

The researchers of the Department of Continental Radioecology of the Institute for Plant and Animal Ecology (IPAE) has for long years conducted radioecological studies of the river Techa and water basins in the East-Urals Radiation Trace (EURT) area, and in the vicinity of the Beloyarsk APS.

The present paper only contains integral indices, such as inventory of the radionuclides contained in the ecosystems studied.

Radioecological conditions in the Urals region were to a significant extent a result of operation of the Mayak Industrial Association (MIA) and a major accident that occurred at the plant. It was in the South Urals that the USSR's first reactor (1948) and first radiochemical plant (1949) were started. Their task was to make the first Soviet atomic bomb.

Releases of radioactive wastes from the radiochemical plant into the river Techa were going on from 1949 through 1956. Seventy-six million cu m of effluents with total beta-activity of 2.8 million Ci were disposed into the river in the period indicated. About 95% of total activity entered the river from March 1950 through November 1951. During that period the average daily discharge of activity constituted 4.3 thousand Ci and the radionuclide composition of sewage waters: 0.8% of Sr-89, 11.6% of Sr-90, 26.8% of rare element radionuclides, 13.6% of Zr-90 + Nb-95, 25.9% of Ru-106. Radioactive releases into the river system were sharply decreased during the next 5 years and made up 9.5 thousand Ci/yr in 1952 and from 0.5 to 2 thousand Ci annually in the period of 1953-1965 (1).

Currently, 0.7×10^{14} Bq of Sr-90 and 1.3×10^{14} Bq of Cs-137 are contained in the river Techa according to measurement data, the bulk of the activity being deposited in the bottom sediments of the river. The Techa floodplains contain 1.3×10^{14} Bq of Sr-90 and 1.7×10^{14} Bq of Cs-137 (Table 1).

Inventory of Sr-90 and Cs-137 in the Techa River and its Floodplains

Components	Sr-90	Cs-137
The river	0.7×10^{14}	1.3×10^{14}
Floodplains	1.3×10^{14}	1.7×10^{14}
Total inventory (river+floodplain)	2.0×10^{14}	3.0×10^{14}

According to the calculations made by the researchers of Department of Continental Radioecology, IPAE, and the National Laboratory, RISO (Denmark), under the supervision of Dr. A. Aarkrog and on the basis of our own results and literature data, approximately 1×10^{15} Bq of radioactive releases (Sr-90 + Cs-137) were migrating from the river Techa into the river-system Iset-Tobol-Irtysh-Ob over 4 decades (2).

The explosion of a tank of volume 300 m^3 which contained 70-80 tons of high-level waste (mostly in the form of nitrate-acetate compounds) occurred on September 1957 on Mayak's production site

About 2 million Ci (out of 20 million Ci of radioactive substances from the tank) were released into the air to the height of up to 1 km which led to the formation of a plume. The deposition of radioactive substances from the plume led to the contamination of an area of 23 thou km^2 in Chelyabinsk, Sverdlovsk and Tyumen Regions. The contaminated area was designated by the term East-Urals Radiation Trace (EURT). The dispersed radioactive substance consisted mainly of short-lived radionuclides. However, it was the long-lived Sr-90 in the mixtures (2.7% of total activity) released that presented the main long-term hazard.

Radioecological monitoring of three lakes (Tygish, Bolshoi Sungul and Chervyanoye) was being carried out by our laboratory for several years in the EURT area, close to the boundary between Sverdlovsk and Chelyabinsk Regions. The results of this work was summarized in an inventory drawn for Sr-90 contents in lakes.

Maximum Sr-90 contents were found to be contained in the Tygish Lake: 374 GBq; while the contents for the Sungul and Chervyanoye were 240 and 167 GBq, respectively (Table). The data used in calculations were only those that were included in our monograph [3].

Table 2

Distribution of Sr-90 in the main components of some lakes on the EURT in Sverdlovsk Region (1992)

Lake	Sr-90 contents, GBq		
	In water	In bottom sediment	Total for the lake
Tygish	35.1	338.8	373.9
B. Sungul	21.8	218.3	240.1
Chervyanoye	15.2	151.7	166.9

Beloyarsky APS named after Kurchatov, the first electric station in the USSR was put on line in 1964. At present one of the blocks FN-600 is functioning on fast neutrons generating 600 Megawatts of power. Two other blocks, one for 100 and the other for 200 megawatts, were shut down.

The scientists of the Department for Continental Radioecology, IPAE, assessed the contents of the principal artificial radionuclides, Co-60, Sr-90, Cs-137, released by the APS into the Beloayrsk Water Reservoir [4].

The cooling pond of the APS contains 244 GBq of Co-60, 144 GBq of Sr-90 and 668 GBq of Cs-137. The concentration of these radionuclides in the bottom sediments constitutes 90%. From 2 to 8% of the total radionuclides indicated above are registered in water and less than 0.01% in plants (Table 3).

Table 3

Distribution of radionuclides among components of the Beloyarsky Water Reservoir

Components	Co-60		Sr-90		Cs-137	
	MBq	%	MBq	%	MBq	%
Water	6600	2.7	9900	8.0	12000	1.8
Sediments	238000	97.3	114000	92.0	656000	98.2
Aquatic plants	26	0.01	12	0.01	15	0.002

It should be mentioned in conclusion that the impacts of the radioactive contamination in the Urals resulting from the operation of a nuclear facility had a regional and local aspects. Thus, the discharges of radionuclides into the Techal led to the pollution of the entire Ob-Irtysh basin.

The accident of 1957 at Mayak IA resulted in the scatter of radioactive matter over the area of 23,000 km^2 . Although the radioecological conditions in the area around Beloyarsky APS are much safer, there is a local pollution of the Beloyarsk Reservoir. Taking into account the unique character of the environmental situation observed in a number of water reservoirs, it would be necessary to conduct further in-depth investigations.

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CHERNOBYL IS NOT EVERYWHERE - A CRITICAL REVIEW OF EMERGENCY PREPAREDNESS 10 YEARS AFTER CHERNOBYL

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The Chernobyl shock had positive consequences for emergency preparedness. But have the right conclusions been drawn and the lessons really learnt? Some countries which were not ready when Chernobyl happened, have based their new concepts too closely on the Chernobyl scenario and do not seem to realize that a next accident may be completely different. Many concepts and developments are too academic and not adapted to the severe operational requirements of real emergency conditions. The information of the public still shows most of the deficiencies which were the cause of the only real "catastrophic" consequences of Chernobyl outside the former Soviet Union. the information failure.
A critical review of the developments in the various aspects of emergency preparedness, based on international comparisons (a.o. from NEA International Emergency exercise INEX-1 and both NEA and FS Workshops [1-7]) shows both real improvements and mistakes or weak points and leads to the following ☺ *positive* and ☹ *negative observations*, ➔*conclusions and proposals for future action*(given in short-hand form due to limited space):

Characteristics and trends of emergency preparedness

characteristic	before Chernobyl	after Chernobyl	optimal concept
attitude	egocentric	(more) cooperative	flexible
range of events	own NPP only	cross-boundary	all radiological events
hazards considered	external exposure	mainly internal exposure	all exposure paths
decision support means	manual + brain	computers academic software	common sense + user-friendly operational software
information of public	"his master's voice" from ivory tower	problems unsolved media dominating	take back initiative use everyday language

Concepts
☺ Cross-boundary and far-field consequences are considered. Some concepts include not only large-scale releases but also design base accidents. ICRP 63 and IAEA-BSS / SS 109 [7] serve as guidelines. Even non-nuclear countries introduced emergency preparedness. Eastern European countries are willing to catch up and to learn, but are hindered by old structures and little transparency. Regional comparison and harmonization of concepts (Scandinavia, D-CH) [4,6]
☹ Many late-coming countries prepare only for Chernobyl-type accidents with emphasis on internal exposure via food path.
➔ Concepts should include all types of events with radiological consequences or hazards. Agreed concepts for optimal monitoring and for forecasts of consequences are needed. Distinguish between emergency phases and return to normal situation. Most late phase and long term problems should be handled by normal competent authorities and means, not by emergency organisations. Test concepts on several scenarios in table-top exercises. [1,3,4]
-> Optimization is best application of available information at the right time (Potter [4])
Criteria
☺ Scandinavian harmonization of intervention criteria. Dose-action-concepts with predetermined derived criteria and Intervention levels (IL) for range of realistic scenarios. [4]
☹ Criteria expressed in non-operational quantities. Discussion and decision on criteria delayed until emergency will happen.
➔Operational criteria depend on national circumstances -> general harmonization not possible, but protective goals should be comparable. Derive operational, scenario-specific Intervention Levels for qualities that can be directly measured during emergency.
Organization
☺ Legal basis for emergency organisation in radiation protection legislation (CH). Coordination of all available national and regional means. Same EO for all types of radiological events. One national contact point for all event types. [3-6]
☹ Reorganization in emergency = chaos. Different (and competing) lead authorities for different event types. Missing coordination if several provinces are affected. Several not corresponding contact points for one country.
➔ As normal as possible, as extraordinary as necessary, as simple as possible: involve normal authorities, hierachies and emergency services as far as possible. Exchange liaison officers between the main partners and with neighbouring countries.
Monitoring
☺ Most countries have installed or planned nation-wide automatic monitoring networks. Some countries begin to optimize their entire stationary and mobile monitoring organization. International monitoring data exchanges are in test phases. [3-6]

☹ Data graveyards of unverified and not representative data. Sampling and monitoring methods not standardized, quality control insufficient. Data exchange prevented or made difficult by not standardized data formats and transmission protocols, different specifications of national telecom hard- and software etc.

➔ Develop concepts for optimal monitoring. Standardize, quality-control and intercompare methods. Define minimum required types and amounts of monitoring data. Do not flood partners with unwanted data, but make data easily retrievable on request.

Data Management (assessment, processing, storage, presentation) [4]

☺ A few countries have developed operational, user-friendly and field tested systems suitable for various event scenarios. Close interactions between data management and information of public have been recognized. Exchange of experiences has started.

☹ Too many systems answer post-Chernobyl requirements but are not suited for other emergencies. Large academic system development programs without operational experience and consultation do not prevent countries to develop their own operational systems. Some systems are pure tv games. Simulators have training value but limited operational use. History writing analysis systems such as developed to analyse Chernobyl data are not suited for operational purposes except in late phase.

➔ Design goals: simplicity, modularity, reliability. Operational data management systems must be tailored to the needs of the particular organizations and users and tested under field conditions. Clearly define for which emergency phases and users a system is designed. Centralise data bases and data evaluation, make only verified and representative data available to users. Fight NIH (not invented here) syndromes and tendencies to re-invent the wheel, intensify open exchange of experiences, also of problems and mistakes. Keep (manual) backup methods operational, use rules of thumb for quick estimates and checks. The products of data management must be suitable for use in information of the public and must allow black/white fax transmission. Color codes of monitoring and contamination maps and headings (coordinates etc) should be standardized.

Decisionmaking [3,6]

☺ Recognition of necessity to make decisions based on uncomplete information, e.g. plant parameters. Preparation of sets of probable decisions for selected scenarios. Protective early countermeasures initiated before release starts. Catalogue of countermeasures as decision basis.

☹ Source-term fixation: source term and forecasts based on source terms are never available in time for early decisions. Political decisionmakers rarely participate in exercises. Normal limits are misused as intervention levels. Reasons for decisions are not clarified or explained. Political or ideological motives are disguised as "health protection".

➔ Decisionmaking cannot be replaced by expert systems. Sophisticated systems alone do not provide good emergency management if concepts and organization are not appropriate. Do not rely on early availability of source term or forecasts. Train for timely decisionmaking based on early incomplete and fuzzy information. Delegate decision for early countermeasures to a team available round the clock and connected to contact point. Provide legal basis for countermeasures and emergency actions. Differentiate between limits for normal use and investigation levels for emergency use. Distinguish between health protection and economic or political reasons. Decisionmaking on countermeasures is a two-step process: Step 1: yes/no decision on need for countermeasure based on dose expected without it. Step 2: if several choices, select the action with the highest dose averted. But: Dose averted is only one of several parameters in selecting the appropriate countermeasure, some others are feasibility, timeliness, negative side-effects, economic costs, psychological and political acceptance.

Countermeasures [1,3,4-6]

☺ Wide acceptance and similar application of major early countermeasures such as sheltering, evacuation, iodine prophylaxis. Emergency planning zones where early countermeasures are considered usually have a radius of about 10 km.

☹ Agricultural and food countermeasures often academic, not realistically feasible or accepted by customers and market. Old lessons from nuclear weapons fallout have been forgotten. It is difficult to overcome established customs and to implement up-to-date concepts.

➔Concentrate on efficient early phase countermeasures. Late phase rehabilitation is not emergency business. Learn from experiences of other types of emergencies (chemical accidents, earthquakes, floods, food poisoning, plagues, firefighting, large traffic accidents), as many problems and solutions are similar. Motivate people to take care of themselves and to understand reasons for countermeasures. In preparing and using countermeasures cooperate with specialists from the relvant fields, radiation protection is only part of the total problem. Even if only a few real cases with health problems are expected, a large part of the affected population will need some kind of psychologic and medical counselling and needs to be calmed down by at least a contamination check. Exit control of affected region should be combined with such services.

Communications [3,6]

☺ Modern technology allows quick transmission of data to several receivers in parallel.

☹ Communications easily overloaded; insufficient backup systems, insufficient testing under stress conditions.

➔ Use only thoroughly tested and instructed systems. Provide backup systems that are less overloaded. Use dedicated lines/systems between key installations. Make full use of modern technology, provided the users are familiar with it.

Information [4,6]

☺ Most countries have recognized that information of the public is decisive for success of emergency

management. and are attempting to coordinate information. Local safety committees (Sweden) are an excellent solution for continuous information of the public and for gaining confidence.
☹ Uncoordinated information by several authorities produces contradictions and loss of confidence. Language is too scientific. Media take initiative from the beginning and dictate rhythme and style of information. Information to tourists and guest-workers in other than the national language(s) is mostly insufficient. Information sources talk about topics outside their competence.
➔Advance information and instruction of population on planned countermeasures is indispensable. Integrate media in emergency organization. Do not fight the media but make sure that official announcements are not mutilated or modified. Do not hide anything, full transparence is the only choice. Use an unambiguous terminology and a clear everyday language. Try to regain the initiative from the media.
The Danish concept of a centralized information database containing both basic and event-related information accessible by all information services, should be internationalized [4]. Information must be coordinated and compatible, delegated down to the front and to the region of the event. Each staff or authority informs only about the topics within its competence and makes its information available to the other information services. Hotlines for answering questions from the public need backup by specialists and by staff with detailed knowledge of the affected region. Include local opinionmakers such as medical doctors, pharmacists, teachers etc. into information concept.

Training and Exercises [2,3,6]

☺ Most countries with an emergency organization have also started training and exercise programs. Especially valuable are the bilateral, regional and international exercises (e.g. CH-D, Nordic, NEA INEX-1 and 2) [2,3].
☹ Many exercises stick to the Chernobyl scenario or to some standard scenario with large release and neglect the large variety of other, more realistic scenarios. Some countries do too many exercises and have no time to evaluate them, draw conclusions and implement the necessary improvements. The real political decisionmakers are rarely actively participating in exercises and receive no training in emergency management. Media pressure and information of the public are rarely part of exercises. In large-scale command-post and field exercises the individual participant sees only a frog's view of the scenario and the organization and has difficulties to define his own place and role.
➔ Motivate key people and show the general picture and the interactions in table top exercises which have a high benefit/cost ratio. Use command-post exercises to test specific phases of a scenario, the communications, decision support and decisionmaking. Field exercises (monitoring) can be run separately from command-post exercises.
Systematic training not only of the teams forming the emergency organizations but also of the political decisionmakers and of their staffs and, if possible, of journalists, is indispensable. Check the cost/benefit ration of exercise programs. Carefully evaluate each exercise, draw conclusions and implement the necessary improvements which then are tested by the next exercise.

International Cooperation [3-6]

☺ Chernobyl intensified international cooperation. Most countries participate in IAEA notification and assistance agreements, Europeans in ECURIE. IAEA and NEA provide useful guidelines and promote exchange of experiences and treatment of important topics in workshops and working groups. The NEA International Nuclear Emergency Exercises are a success and attract more and more countries, also from Eastern Europe, Asia and the Americas. Bilateral and regional cooperation increases. The FS has a useful bilateral (CH-D) Working Group on Emergency Management.
☹ Cooperation on the level of the radiation protection and emergency management specialists is better than on the level of decisionmakers. Cross-boundary assistance may be delayed or blocked by stupid administrative obstacles (customs, prestige etc.). Many of the means listed in the IAEA assistance tables would not really be available in a large scale emergency because they would be needed by the country itself.
➔ Continue and intensify international cooperation in working groups, workshops, seminars and joint exercises. Provide training and practice opportunities for emergency managers and staff from countries which are in the process of forming an emergency organization.

Comprehensive Optimized Emergency Preparedness

should provide for: coordination of all available national means, both civilian, civil defense and military ones / key monitoring and data management systems used routinely for environmental monitoring / one contact point and one front-line emergency management center and team available around the clock , dealing with all types of events with radiological hazards or consequences, having competence to alarm and protect population and to manage emergency and information during first 1-2 days of an event / coordinated information concept. [5, 6]

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PAPER TITLE DEVELOPING EMERGENCY MANAGEMENT RESOURCES

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ABSTRACT (See instructions overleaf)

To effectively direct an emergency response, an emergency manager needs to be able to quickly obtain the magnitude of an event that involves the release or potential release of radioactive or hazardous materials. By using facility-specific hazardous material and meteorological data, dose calculation methods of the International Commission on Radiological Protection, and the seven emergency classification levels of the International Atomic Energy Agency, an emergency assessment resource manual or summary (EARM or EAS) can be developed as a tool for rapid response to local site emergencies. Such tools provide a summary of the facility's operations, hazardous material inventories, worst-case consequence analyses (maximum impact distances) for all probable events, and distances to potentially impacted populations. These manuals/summaries allow the emergency manager to make immediate, conservative, and accurate decisions during an emergency and also provide an excellent pre-planning tool for development of exercise scenarios.

This paper describes the process and input materials used to prepare these emergency management resources for widely divergent facilities and sites within the U.S. Department of Energy (DOE) complex. Their implementation at the local sites and at DOE Headquarters during exercises has rapidly aided the evaluation of questions such as: What are the upper bound consequences to be expected from an event? What are the consequences most likely to be? What else can or could happen?

OFF-SITE RADIOLOGICAL EMERGENCY PLANNING JUSTIFICATION AND CO-ORDINATION

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INTRODUCTION

The nuclear power reactor accidents at Three Mile Island (TMI) (1979) and Chernobyl (1986) have influenced off-site emergency planning world-wide. Despite very limited environmental radiological consequences of the TMI accident and a limited need of response actions of the emergency organisation outside the plant, the accident caused a higher attention and new conditions regarding the bases for off-site emergency planning. After TMI in many countries more attention was also given to even more severe reactor accidents than so called `design base accidents´, i.e. accidents were now considered which could lead to emergency situations with large scale land contamination's.

The Chernobyl accident some years later made it quite clear that existing national emergency preparedness and international agreements were not sufficient to cope with an accident leading to a large scale radioactive contamination with an impact in several countries. There was an obvious lack of bilateral or multilateral agreements regarding early warning, preparations for exchange of information and measurement data, or shortcomings in existing agreements due to inadequate preparedness and training.

In response to the experience of these two accidents, the awareness and resource spending in the field of radiological emergency planning have increased nationally as well as within the various international organisations. Increased efforts were clearly necessary, and by and large they have improved emergency preparedness considerably.

However, we wish to raise the following questions:

- To what extent could the various planning measures be considered as justified from the radiation protection point of view? Are there instances where public opinions or even `expert opinions´ forced decision makers to take some actions which could be questioned?

- A lot of actions have been taken, and new ones can be expected in the future, to improve emergency preparedness in Central and Eastern Europe and the newly independent states of the former Soviet Union. Currently, there is a need for a co-ordination of all the various ongoing assistance activities, in order to avoid duplication of efforts and waste of limited resources. Which initiatives ought to be taken to make such activities as effective as possible?

1 EARLY NOTIFICATION OF A NUCLEAR ACCIDENT

The IAEA convention on early notification of a nuclear accident and information exchange has got a world-wide acceptance. A communication system for notification and information in case of a radiological accident (ECURIE), which is to some extent parallel to that of IAEA, has been established within the European Union in accordance with a Council Decision of Dec. 1987. For the Member States of EU, these two systems lead to duplication regarding exercises and communication tests. Furthermore, to some extent the two communication systems use different means of communication, thus increasing workload but hardly improving reliability. Communication tests are performed on more or less routine bases in accordance with the IAEA convention and between the Member States of EU through various ECURIE exercises. In this area we believe that there is a need for an extended

international co-operation with common exercises arranged by IAEA and the Commission of the EU. The purpose would be to avoid the use of procedures and formats which are not harmonised. Otherwise, such unaligned procedures can in practice obstruct or prevent the involved Member States from fulfilling their obligations in case of an accident.

2 INTERNATIONAL NUCLEAR EMERGENCY EXERCISES

The accidents at TMI and Chernobyl prompted various international organisations to start or intensify their activities within the field of emergency planning and preparedness during the late 1980s and early 1990s. The reason for that is mainly the different international and transborder aspects in the event of a severe nuclear reactor accident. One of these new activities launched by the Nuclear Energy Agency of OECD was the arrangement of international off-site emergency exercises as INEX 1 (1993) and thereafter followed by the ongoing planning of the INEX 2 regional exercises of which the first exercise will take place in Switzerland 7th November 1996. However, since several years bilateral exercises have been performed on more or less routine bases in different regions where nuclear reactor sites are located close to borders.

With the background of our participation in INEX 1 and our experience in the field of emergency planning we interpret that the overall purpose and the specific objectives of the INEX 1 set by the organiser were reached, even if the final analysis (evaluation report) of the exercises was delayed some years.

It was particularly valuable that the accident scenario considered the two different roles of countries in emergency response , depending on whether the country has a nuclear power reactor site or not. The exercise encompassed both an accident country and a neighbouring country, both of which affected by radioactive fallout. The exercise scenario also covered the post-release phases of an accident, which is an advantage as most exercises only deal with the initial phase of an accident.

Seen from our experience the scenario of INEX 1 was very well developed and useful. For the future we encourage such an international co-operation regarding the development of new exercise scenarios intended to be used in connection with national and bilateral exercises. The development of scenarios in the framework of an international expert group will stimulate participating organisations and emphasise the need to consider the various international aspects involved. It will also give an internationally accepted shaping and quality to the exercise scenarios. Not least for countries with only limited experience and resources in the field of peacetime radiological emergency planning such an international co-operation can be expected to be of great value.

However, we believe that most of the evaluation work of an exercise can and must be performed in the respective countries as many of the aspects considered are very country-specific. Large scale international co-operation in this respect seems therefore not to be of the same priority as the development of exercise scenarios.

3 MONITORING ARTIFICIAL RADIOACTIVITY

In case of an accident which is expected to cause or has caused fallout of radioactive substances the authorities in the involved countries need fast and reliable information on the location and characteristics of the fallout. Therefore most countries have established national strategies to map their territory in case of an inland as well as an accident abroad.

In addition to the IAEA convention on early notification and the ECURIE system, automatic gamma monitoring stations form the most common part of a national early warning system. Concerning the number of these monitoring stations expressed per unit area there are variations between the various countries from 1 station per 100 km^2 to 1 per 10 000 km^2. This means that countries as Germany has thousands of stations, Finland more than two hundred, Norway and

Sweden some tenths. To some extent these numbers also reflect a difference in the resources used for this purpose.
There are also extensive programmes in most countries for field measurements and laboratory analyses of environmental and food samples. Normally such a programme also includes high resolution measurements of airborne radioactivity using air filter stations and resources for airborne fallout mapping.
In accordance with the various international conventions and bilateral agreements an international exchange of measurement data is required in the event of an radiological emergency. To warrant a good quality and a comparability of the data exchanged there is a need for an extended international harmonisation regarding calibration procedures, instructions for the collection of environmental samples an the presentation of the values in connection with reporting of data. Which of all available data will be of most interest to communicate between the countries? To what extent shall data be excluded to avoid overloading of the communication systems or of the emergency response organisations in the various phases of an accident?
As we see it these are examples of questions which need an international discussion to reach some form of international standard or recommendation, in order to express the extent to which different measurement resources ought at least to be included in a national radiological emergency planning. The aim is to achieve a possibility to check whether a national emergency planning can be considered to fulfil an international agreed acceptance level concerning measurement resources. One further purpose with such a standard is to give a background for decisions to what extent a spending of resources can be considered as justified from the radiation protection point of view, including the need to have a background to give reliable information to the public in case of a radiological emergency.

4 CO-ORDINATION OF ASSISTANCE ACTIONS

In recent years a lot of assistance actions have been taken to improve the radiological emergency preparedness for peacetime accidents in Eastern Europe and the newly independent states of the former Soviet Union.
Stationary automatic gamma monitoring stations have been set up. National emergency plans have been reviewed by national teams. Various measurement and communication equipment , including computers, have been introduced to upgrade national laboratories, authorities and organisations involved in the various national emergency response organisations. Workshops on the establishing of a strategy for the information of the public, scientific visits and training courses have been organised.
Our experience is that in this field there is an urgent need to try to co-ordinate all different initiatives for assistance actions which now are going on or planned by international organisations and individual countries. Most of the assistance actions so far seem to have been performed without any co-ordination or attempt to prioritise contributions.
Recently an assessment study regarding the priority of needs for assistance in the field of radiological emergency planning has been done by the Commission of the European Union. Hopefully this study will be followed by an initiative to also co-ordinate future assistance actions to reach an effective use of available resources spended. Our belief is that a better co-ordination in the future of the various assistance actions will be of great benefit to the deliverer of assistance and not least for the receiving country.

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1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE *THE FRENCH CONTRIBUTION TO THE KOLA EXERCISE*

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MAJOR SCIENTIFIC TOPIC NUMBER *4-4* (see page 7)

ABSTRACT (See instructions overleaf)

The Russian ministry for Civil Defence (EMERCOM) took the initiative of a major exercice played around the consequences of a simulated major release of radioactive matters. This release was supposed to be emitted by the Polyarnie Zory nuclear power plant, in the Kola peninsula. The Nuclear Safety Institute of the Academy of Sciences of the Russian Federation (IBRAE-RAN) was the technical support of EMERCOM for the implementation of all the stages of this exercise.
The Nuclear Safety and Radioprotection Institute (IPSN) provided its assistance to IBRAE all along this process. The exercise was played on May 30th, May 31st and June 1st 1995 in simulating respectively the 3rd day, the 15th day and the 30th day after the release.

The Department of Human Affairs (DHA) of the UN and the IAEA sponsorised this exercise. About fifteen countries participated to it.

IPSN placed its Emergency Center of Fontenay-aux-Roses (Paris region) at the disposal of its Russian counterparts. A real "distant expertise" has been so implemented. Satellite and radio links between Apatity, site of the exercise, and Paris were operated by the French Ministry of Interior. Self sufficient national means for transportation and stay of the experts and the equipement, including a governmental aircraft, have been used. The mission lasted from May 25th to June 3rd, 1995. About fifty French experts contributed to this exercise.

Such a participation is a concrete and operational answer that France could give in the frame of the IAEA Convention on Assistance in the case of Nuclear Accident or Radiological Emergency.

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PAPER TITLE *STRENGTHENING THE FRENCH EMERGENCY PROVISIONS DEALING WITH POST-ACCIDENTAL SITUATIONS*

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ABSTRACT (See instructions overleaf)

The set of the actors who intervene in the management of a nuclear or radiological emergency varies very much according to the events it can be referred to and the different stages of their evolution. The problems which has to be dealt with are of various nature and require a large spectrum of skills to be solved.

In France, the off-site emergency plans address mainly the decision making and the implementation of the classical short term countermeasures in case of nuclear emergency. The response to the threat phase or to the very early stage after an accidental release of radioactive matters is set up by the centralised organisation of well-known non numerous actors. The efficiency of these plans is reasonably validated by a strong policy of exercises.

Things change when further stages after a release are considered. A lot of other actors will play major roles. Some of them are not really identified. For instance, representatives of the world of Justice or associations of victims have not yet been included in the current attempts of such radiological situations assessment. In addition, new difficult questions, strongly linked between them, can be raised, such as :

- what is "return to normality" ?
- which trust put to which actors ?
- which strategy for compensation ?

All theses issues are currently subjects to concrete debates. Full scale interministerial national exercises are foreseen within the next three years. They aim to strengthen the provisions made by the public authorities. International collaboration overtures are from now on foreseen, particularly with Russia, in some of these important projects.

CURRENT MANAGEMENT APPROACHES IN THE EVENT OF RADIOLOGICAL EMERGENCIES

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ABSTRACT

Since the accident at the nuclear power plant at Chernobyl in 1986, most European countries have undertaken major efforts to implement international conventions and recommendations, develop real-time emergency response systems and upgrade early warning/monitoring networks. Despite of these improvements several areas (logistics, hardware, software) are still in need for further R & D.

THE ISSUES

The accident at the Ukrainian Chernobyl nuclear power plant on April 26, 1986, reveiled severe emergency management deficiencies in Europe in the following areas:

- lack of adequate international agreements addressing the transnational consequences of such a major nuclear accident.
- problems with the degree of comparability of information obtained at the local, regional or national levels, describing the environmental radioactive contamination situation.
- incoherent national regulations concerning trade import and export, especially food, food products and animal fodder.
- inadequate public information systems dealing with perceived fears of members of the public as well as dealing with the media.

These issues can be addressed in terms of monitoring networks and communication, emergency response systems, and international conventions and recommendations. In this paper, examples of the international and national responses to improve radiological emergency management in Europe over the last 10 years is briefly described.

MONITORING NETWORKS AND COMMUNICATION

The arrival of the Chernobyl-fallout in Western Europe in three major plumes during 26 - 30 April 1986 resulted in the upgrading of several national early warning- and environmental monitoring networks. In most countries where a large network has been established, the routine environmental monitoring mode (usually gamma dose-rate) will trigger the emergency response operating mode when elevated activities are detected, but in some cases manual operation is still used. Monitoring for atmospheric aerosols and iodine have also been considered to be part of the overall national systems.

If one takes into account the existing (and ready to be deployed) gamma dose-rate and aerosol monitoring coverage per unit population density (persons/km^2), the ten countries that rank highest in Europe are shown in Table 1. The number of stations per population density of the country can be grouped into three general categories of monitoring capabilities of > 10 stations/pop density, between 1 and 5 stations/pop density, and less than 1 stations/pop density. Finland, Spain (when the stations are deployed) and Germany fall into the category with greater than 10 stations per unit population density. Austria, Ukraine, Sweden, France and Norway fall between 1 to 5 stations and leaving Ireland and Hungary with the group of other European countries that have fewer than 1 stations per unit population density.

The density of radiation monitors is irrespective to the size of the national nuclear program (for example France and Germany compared to Austria and Norway), and these results illustrate the importance given to establishment of monitoring networks. This is important due to the relatively small distances between European States leading to the increased probability of transboundary contamination. Further standardization efforts are necessary to improve the comparability of data obtained with these systems:

- hardware: radiation detector systems; measurement conditions (height above ground, collimators, energy response, time-averaging periods, data transmission frequency), early-warning alarm-discriminator;
- analysis: quality assurance and -control of input data and data evaluation systems (e.g. system status, formal and technical compliance with standards or database, consistency between data pools).

At the IAEA (Vienna) a 24 hour-communication centre has been established for the exchange of data in standardized formats. An International Nuclear Event Scale (INES) has been devised. The seven classes of accident scenarios take into account off-site and on-site impacts, worker exposures and degradation of safety systems. It is hoped that with the use of INES misunderstandings can be avoided among all parties envolved in the management of a large scale nuclear accident.

EMERGENCY RESPONSE SYSTEMS

European capabilities for the real-time assessment of accident consequences and emergency response have improved since 1986 due to the development of computer-based support systems, such as: SPADE, MC31 and 3-DRAW for modelling atmospheric dispersion at short-range, mesoscale ($\leq$ 200 km from the source) or long range; STEP and START as feedback-models for estimating the source term and releases; EURALERT as dose assessment programme for different pathways, using data from modelling and measurements.

The RODOS project, funded by the Radiation Fission Safety Program of DGXII of the European Commission, is a major effort to integrate real-time monitoring and emergency response decision-making on a European scale. The large cooperative effort with Eastern European countries adds to the importance of this project.

Despite of these improvements several issues remain to be addressed in the future, e.g.:

- quantification of the uncertainties for the different models, particularly in view of limited respresentativeness of the data available for model validation;
- sensitivity analysis on the significance of contributing factors, such as: topography, agricultural practices or complex meteorological conditions (e.g. deposition in fog; variable wind conditions in stagnant anticyclones), gravitational settling.

INTERNATIONAL RECOMMENDATIONS AND CONVENTIONS

An essential element of confusion during the post-Chernobyl period was the heterogeneous international approach to the **limitation of the radionuclide content in foods**. A major step towards harmonization, thereby facilitating international trade, was the development of the *Codex Alimentarius Commission's guidelines* (1). It is to be noted that only 4 different values for limiting the activity concentration of 10 nuclides in all food, milk and drinking water for general consumption and for infants are used, e.g. 1000 Bq/kg for ^{134}Cs, ^{137}Cs, ^{103}Ru, ^{106}Ru, ^{89}Sr. This results in an increase of the ingestion dose for those countries with previously lower limits (e.g. for ^{137}Cs in milk in the EU: 370 Bq/kg; in Austria: 175 Bq/kg, resp. 12 Bq/kg for infants).

In the area of **early notification and mutual emergency assistance** in a nuclear accident two IAEA-Conventions reduce the shortcomings noticed after the Chernobyl accident (2,3). However, some topics of ambiguity remain, such as:

- what information about the accident is considered „relevant"? What time-period meets the request for information-supply „as soon as possible"?
- what uncertainty is acceptable in the modelling-based forecast whether a release „may occur", whether this „may result" in a transboundary contamination, which State „may be affected"?
- if the resources of a State are insufficient to respond „adequately" to a large accident, can it still refuse external assistance (because of security issues or pride), thereby increasing the radiological consequences in another State?

The issue of timely provision of relevant information about the Chernobyl accident by the former USSR and the adequacy of some European countermeasures were subject to debate. It is foreseeable that divergent interpretations of the above topics can cause further dispute in another large scale nuclear accident.

CONCLUSIONS

Major improvements have been achieved in emergency management since 1986. Nevertheless, comparability of the assessment of the radiological situation and the adequacy of countermeasures are still affected by remaining ambiguities in the interpretation of conventions, insufficient model validation for real-time assessment and lack of standardization concerning monitoring networks. Besides reducing the difficulties that can result in the application of emergency response measures, a more standardized approach would facilitate consistent communication of countermeasures and actual risks to the general public.

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Table 1: Ranking of top-ten values of a list of 19 States with early warning monitoring systems (4,5,6)

Country	No. of operating nuclear power plants	Rank $\left[\frac{\text{total no. of stations}}{\text{population density}}\right]$
Austria	0	4
Finland	4	1
France	56	7
Germany	21	3
Hungary	1	10
Ireland	0	9
Norway	0	8
Spain	9	2
Sweden	12	6
Ukraine	15	5

EMERGENCY MANAGEMENT IN SLOVENIA

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Abstract: Accidents may occur during the production, transport, use or storage of the radioactive materials. Experience has shown that advance emergency planning is essential in order to mitigate the consequences of the accidents. The paper presents the general philosophy applied in Slovenia to the problem of emergency response; background information on the Slovene use of nuclear energy; and finally legal, organizational, planning and reviewing aspects of the Civil Protection applying to nuclear or radiological emergencies.

INTRODUCTION

Changes that occurred in Slovenia as a result of independence required a new assessment of the situation in the field of disaster protection, as well as the coordination of policies and goals. Slovenia is at present in the process of reconstructing its system of protection against natural and other disasters.

The system for protection and rescue is organized as an overall system established for the protection and rescue of people, material and other goods in the case of natural and other disasters, in war and in other extraordinary circumstances. The most important points for legal and system-related regulation in the field of disaster protection are determined by the National Assembly. Civil protection and other disaster protection activities are humanitarian and non-military in nature.

In this general context nuclear or radiological accidents fall under industrial accidents which in turn are treated as "other disasters".

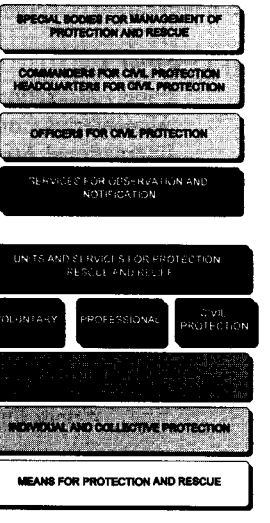

Figure 1. The system of disaster protection

GENERAL PHILOSOPHY AND BACKGROUND INFORMATION

Protection against disasters is undertaken by those bodies, services and organizations, whose regular activities already include such operations. If these are not sufficient, additional forces and means for protection, rescue and relief are brought in. These are mostly forces and means of Civil Protection (CP) which are organized as a special, designated section of the overall safety an protection system in the country. It is obvious that state and local authorities cannot provide sufficient collective assistance during disasters. Therefore the initiative of the local population and their capabilities for individual protection are of vital importance. The Slovenian system of disaster protection is shown in Fig. 1. and disaster protection duties in Fig. 2.

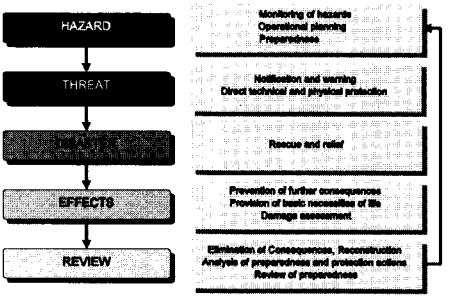

Figure 2. Disaster protection duties

On the basis of the new Constitution entirely new legislation is currently under preparation. Two fundamental laws already exists.

Operational implementation of protection, rescue and relief in the event of natural and other disasters, in war time and other extraordinary circumstances is planned through protection and rescue plans. Protection and rescue plans are based on risk assessments, studies of the vulnerability of the living environment and other studies and investigations relevant to protection and rescue. All of the above information is used to determine which local communities or economic associations or other organisations require protection and rescue plans. Forces of protection, rescue and relief at the state level are shown in Fig. 3.

Slovenia has beside more than 400 radiation sources also one nuclear power plant (a two-loop PWR, 632 MW electric power) and one research reactor. Construction at the Krško site began in early 1975. The plant was synchronized to the national grid in October 1981.

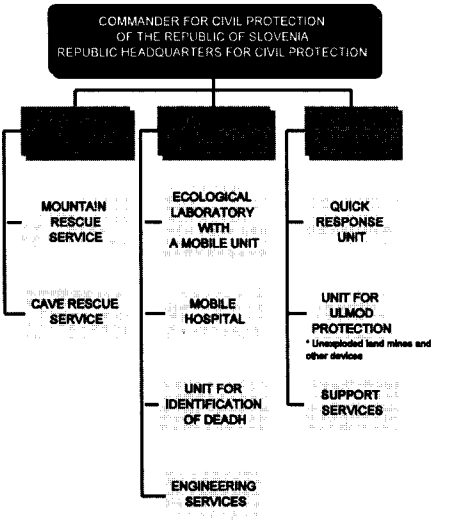

Figure 3. Forces of protection, rescue and relief of the Republic of Slovenia

PLANS OF PROTECTION AND RESCUE IN THE CASE OF NUCLEAR ACCIDENT AT THE KRŠKO NUCLEAR POWER PLANT

Measures to be taken in emergency situations are stipulated in a number of plans, from the national level to the local community level and the plant itself.

According to the law Administration for Civil Protection and Disaster Relief (ACPDR) is responsible for planning and preparedness off-site at the national level. The ACPDR prepared, amended and maintains Emergency Plan in the Event of a Nuclear Accident at Krško NPP. The plan postulates three emergency planning zones (EPZ): (i) urgent protective actions EPZ within 10 km radius, (ii) food-chain and long term protective actions EPZ within 25 km radius and (iii) general preparedness EPZ covering the whole Slovenia.

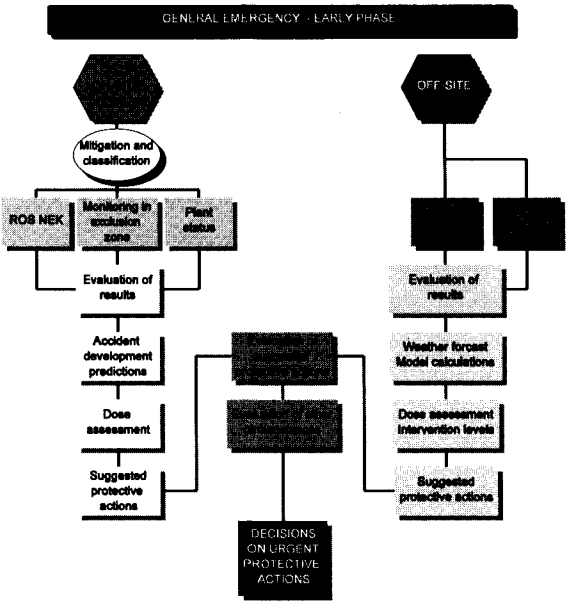

Figure 4. Emergency response in early phase of the accident

The emergency plan at Krško NPP defines an organised and effective response by personnel at the NPP to ensure protection, health and safety and to reduce the consequences of any accidents. Emergency situations in the facility are divided into four categories: abnormal occurrence, abnormal situation, site emergency and general emergency. Appropriate actions are planned for each category.

The Administration of Nuclear Safety (ANS) has its own plan of actions, harmonized with the national plan. In the event of nuclear or radiological emergency the ANS functions as a professional support and advisory body to the Civil Protection Headquarters. Three expert groups are organised for analysis of the accident, dose assessment and for support

and preparation of information for public and the IAEA. The ANS is responsible to notify the IAEA and the countries which could be affected by the accident through international conventions and bilateral agreements. The embassies of neighbouring countries are informed through the Ministry of Foreign Affairs.

Emergency response strategy in early phase of the accident and structure of off-site emergency monitoring are shown on Fig. 4 and 5 respectively.

To test the plans and to remove shortcomings an overall exercise was prepared and conducted in 1993. The exercise showed that the planned on-site and off-site actions form a sound basis for taking emergency actions.

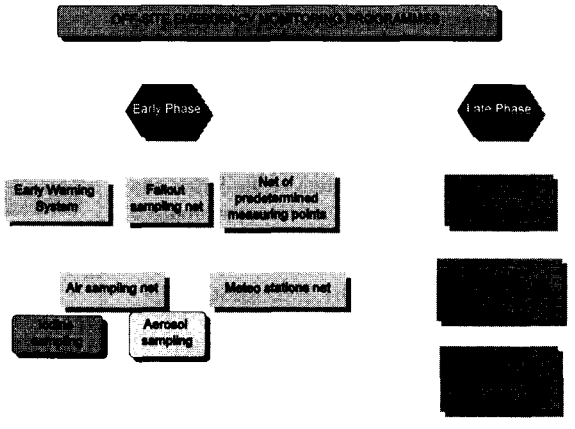

Figure 5. Off-site emergency monitoring programmes

CONCLUSIONS

- Slovenia has Civil Protection for coping with the emergencies.
- The CP Headquarters at national level established highly professional emergency units in different fields of emergency response.
- Additional efforts to increase the preparedness of the CP staff and units to cope with radiological emergencies have paid off handsomely and have led to a general improvements of the performance of CP, also for other types of emergencies.

Finally, a reasonable degree of preparedness has been achieved. However, there always remain some questions and problems yet to be answered and solved. Among these one might choose the following ones. In near future Slovenia has to:

i. complete its own regulations,
ii. implement a new national early warning system,
iii. improve adequate and effective mean for public information and
iv. update the equipment and training of CP in dealing with radiological emergencies.

SOME CONSIDERATION ON DECISION-MAKING IN A NUCLEAR EMERGENCY

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1. Introduction

In various phases of a disaster, the result of decision-making should finally reach the local public of interest, and the detriment among the public should be minimized/optimized according to the proper response by the public who receive the result. The decision involves the proper selection among countermeasures, and should reach each one of the public. The expression to be informed to the public should be quite understandable in its meaning.

After Hanshin big earthquake (Jan. 17, 1995), the Basic Plan for countermeasures against disasters which is the foundation for the basic law for disasters has been largely revised and reissued in July 1995. For preparedness and countermeasures outside a nuclear facility, there are many useful experiences that have to be learnt from natural disasters. In Japan and China, there have been no major nuclear accidents affecting the public in the environment. However, preparedness for nuclear emergency derived from natural disasters is important.

2. Guide lines for sheltering and preparedness

In Japan, if some trouble occurs in a nuclear facility, leading to serious situation and possibly developing to an emergency, the judgment on whether emergency or not will be made at the Central Committee on Countermeasures and the Nuclear Safety Committee. The result of the judgment will be immediately transferred to the local government of the nuclear facility. The headquarters for countermeasures against disasters will be activated in the local government. And information necessary for countermeasures will be collected. Simultaneously, the experts on nuclear reactor engineering will be dispatched to the facility at emergency and the experts on radiation safety will be also dispatched to the headquarters by the Central Government.

In order to support to the experts, COSTA (a computerized support system for the emergency technical advisory body in Japan) has been developed[1]. In operation (activation) of the system, the commencement of the system depends on the first notification on the trouble in a nuclear facility. And it is assumed that the first notification should not be considerably delayed.

The precise estimation of the reactor situation needs information on the plant 3-4 times every hour [2-3]. Since some uncertainties are accompanied on the estimation of the amount of radioactivity released and prediction of doses to the public, decision-making on countermeasures should pay attention to the uncertainties. [2-3]

The system for prediction of environmental doses is also operated for the

decision of countermeasures, the dose predicted would be accompanied by uncertainty of a factor about 2.

The decision-makers in the headquarters will make decision for selection of countermeasures based on the data from emergency environmental monitoring at first stage, site specific conditions (feasibility for countermeasures, risk in practical case, number of people influenced, avertable dose, and so on) being taken into account. Some flexibility would lay on the decision-makers.

Some uncertainty which will accompany the decision from only predicted dose is listed in the following table.

	situation in practical case
predicted dose: at minimum	• some countermeasures be made unnecessary in real case
predicted dose: at maximum	• at first, a countermeasure be not made necessary in real case. • later, a countermeasure be made, some confusion among the local public would be accompanied in real case.

The delayed countermeasure would be accompanied with some confusion (in allocating buses for evacuation, for example) and with unnecessary detriment among the local public by increasing dose.

3. **Lessons learnt from natural disasters**

Many lessons we should learn from natural disasters. Some of them would be quite useful to nuclear emergency.

1) Information and communication

There are, in general, (a) loudspeaker car and (b) local simultaneous broadcasting system. Their failures in natural disasters are briefly summarized in the following table.

	car speed		information communication on tsunami after an earthquake
	fast	slow	
(a)	local public cannot hear	can hear only in a small area	
	in result, both are not very helpful to countermeasures		
(b)	• misunderstanding by receivers of the information • vagueness in expression of the information • depending on skill by a reporter		

TVs and radios are the fastest communication systems among the mass-media. However, the extent of information arrival to the local public is restricted to an audience rating. There points out the concrete, correctness and appropriate in the expression of information.

2) In obtaining information

At present, telephones are needed as convenient communication means, however there are some overcrowding in communication by telephones in a disaster. Some radio cars are available in a disaster in obtaining in formation on damage. The mass-media use also more powerful radio system, there are happend the sensitivity suppression phenomena on the car local government used in the case of the near radio frequency assigned.

The results from emergency environmental monitoring at first stage are sent to the headquarters via a radio car. There needs some measure in the case where the assigned frequency of a radio car is close to that of mass-media.

3) The public as receivers of information

There found many cases in natural disasters, the receivers did not act after information on evacuation communicated. This is because the receivers have prejudice taking real risk as not so dangerous, namely they would not want to accept psychologically the dangerous situation about themselves [4]. The decision-makers in the headquarters should pay attention to such a problem.

4) Problems about the headquarters

It was a general way to start the headquarters after collecting information on the damage. After Hanshin big earthquake, it is pointed out to be too late for some disasters. The headquarters should function even in the half number of the members, not after all members gathered.

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THE IMPLEMENTATION OF THE IAEA ACCIDENT RESPONSE PLAN IN YUGOSLAV PRACTICE

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ABSTRACT

One of the important lessons from the Chernobyl accident is the necessity of existence of operational national emergency response plan. Summarising consequences and experiences after Chernobyl accident, expert groups from IAEA, ICRP and other international scientific organisations, have been extensively worked on reviewing old ones, and preparing new radiation protection and nuclear safety principals and codes. One of the important issue is national emergency response plan for radiological accident.

The nuclear accident response plan in Yugoslavia is presented in this paper. It is essentially based on IAEA model national response plan for radiological accident (8). This model has to be adjusted to the specificity of member states. The optimum society organisation for emergency management in the case of accidents in ionising radiation sources practices is suggested in this paper. Specific characteristics of Yugoslav state organisation relating to accident response are emphasised.

INTRODUCTION

Large accidents at nuclear fuel cycle plants in neighboring states could create big problems for Yugoslavia and other counties. As for prevention, effective protection and lowering the consequences, an accident must be detected as soon as possible. Therefore, the prompt gamma monitoring system is a must. Besides, the population, material goods and environment would be protected to the maximum and the consequences decreased if during the accident people work by appropriate plans.

In Yugoslav radiation protection regulation which is numerous there are several regulation concerning large accident problems. Problems of accident detection, declaration and interventions are dealt with in documents (1, 2, 3, 4). Limits in (5) should be considered as adequate only in accidental conditions. There is three nuclear objects in Yugoslavia, (two research reactors and one temporary radioactive waste storage center). Their monitoring is precisely defined in (6).

But there is still not an appropriate emergency response plan. Instead of the basic act (7) the new one is in preparation. Therefore, the new regulation must be imposed with emergency response plan for radiological accidents including measures for consequences prevention in accordance with the international recommendations (8).

Nevertheless, in Yugoslav practice, especially for Republic of Serbia, exists the response plan (Fig.1) developed by radiation protection experts several years ago (9). In this paper the basic characteristics of that plan are presented. It is in a good agreement with IAEA plan (8). This similarity is discussed too. These two plans will be the basis for developing a new Yugoslav accidents response plan.

YUGOSLAV RESPONSE PLAN CHARACTERISTICS

The subject of the existing response plan (YUPLAN) are:

1. accidents that could happened abroad and endanger Yugoslavia
2. accidents on domestic nuclear facilities which consequences or potential consequences exceed facility bounds
3. accidents at domestic nuclear facilities which consequences or potential consequences do not exceed facility bound but there is not possibility for consequences preventing.

The goal of YUPLAN was a society organisation establishing as well as infrastructure and cadre and instrumentation defining.

Response organisation is defined by:

1. existing state organisation
2. early detection of nuclear accident
3. preventing the consequences in accordance with organisation and possibility of society
4. efficient averting the consequences for people and the environment

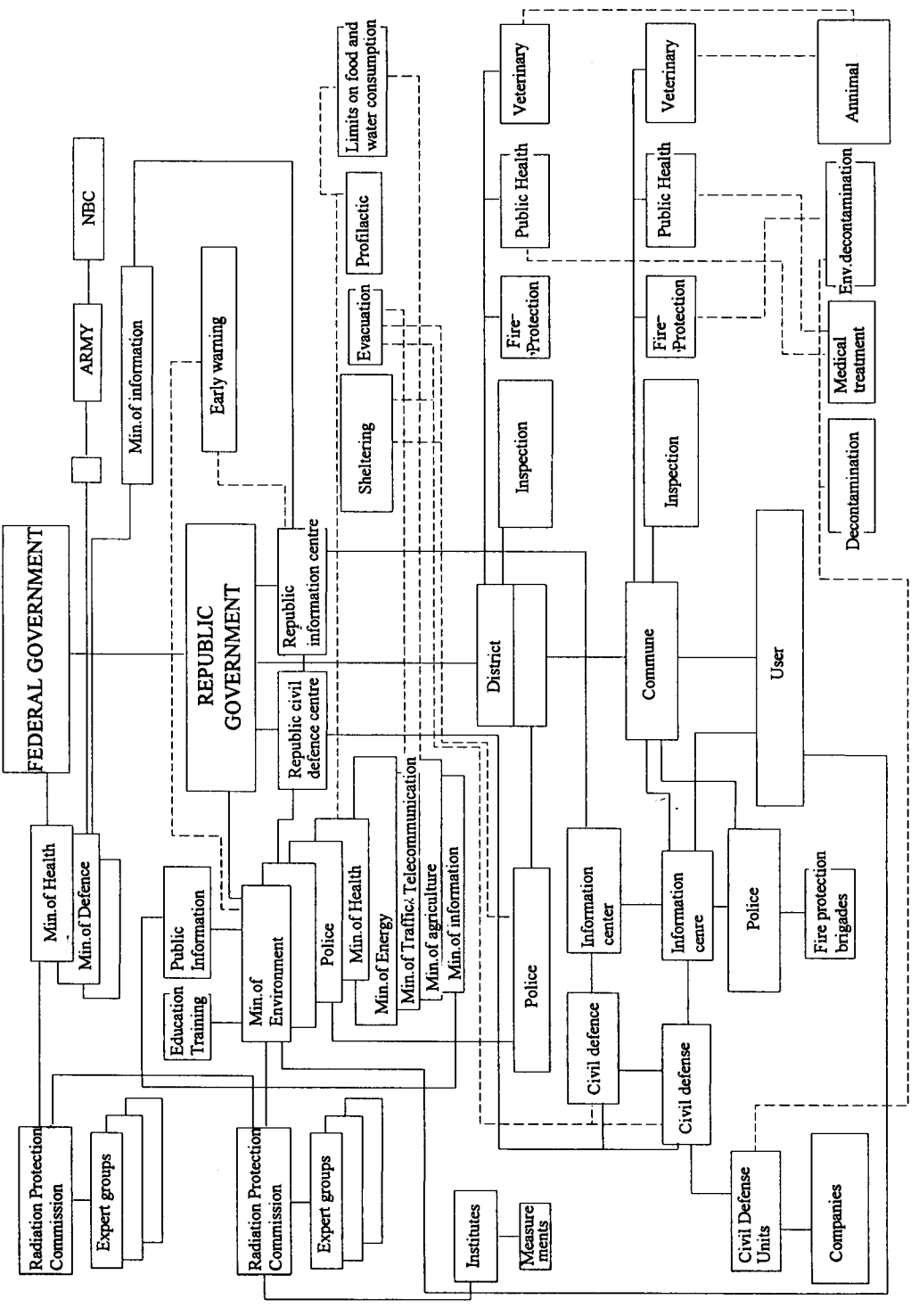

Figure 1. Scheme of the existing Yugoslav emergency response plan

The main organisation subjects are:

1. authority (federal, state, local) with different organs
2. owner of nuclear object (for domestic accidents)
3. prompt monitoring system monitoring
4. measurements (authorised institutions)
5. preventing measures (2)
6. averting the consequences
7. public relation
8. exercise
9. public education

How these organisation works? The system activation begins with nuclear accident notification from the government. The basis for notification is: direct call from nuclear facility authority, prompt system monitoring response, or by diplomatic way. The government activate Radiation protection commission which exist at federal and state level. The commission is well prepared for accidental situation (software, models, measurement). It ascertains the characteristics of the accident and according to severity of accident proposes concrete governmental measures. One of the first measure is environmental monitoring system activation. After those measurements the commission propose the most appropriate measure. One of the most important commission job is defining intervention levels and derived intervention levels for that accident. The commission take care of accidents in all phases and propose the end of accident.

The role of other institution involved in YUPLAN is the same as in other countries and in IAEA plan (8).

YUPLAN AND IAEA PLAN COMPARISON

At the time of the Chernobyl accident, Yugoslav accident detection system was based on absorbed dose rate measurements on several selected locations, manually done in predefined time intervals. Although it was considered as adequate for countries without nuclear power plants, practically it presented itself as very slow and inertial one. Anyway it was very good basement for the implementation of the IAEA Emergency Response Plan. The basic idea is the same in both plans. Some specific elements, concerning political state organisation had to be incorporated.

Differences can be seen in the precision of defining responsibilities of various scientific and technical authorised organisation ivolved in YUPLAN. This differences are naw to be inplemented in new version of Yugoslav Respoce Plan for Radiological Accidents.

CONCLUSION

The Chernobyl accident had significant versatile influence on regulatory changes. One of the aspects is an adequate society organisation in emergency conditions. The necessity of the existence of well co-ordinated plan of action in the emergency is proved, even for non nuclear countries. In the new Radiation Protection Act it has been implemented the governmental responsibility for the existence of the Emergency Response Plan in Yugoslavia. The important task still remained to be done is to make it effective and operable.

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ON THE USE OF THE NORMALIZED LEAST-SQUARES METHOD TO RECONSTRUCT THE SOURCE TERM FROM TRACER EXPERIMENTS

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INTRODUCTION

Should an accident in a nuclear power plant (NPP) result in a major release of radioactivity into the atmosphere, a number of protective actions must be issued to avoid excessive exposure of the population to ionizing radiation. Early countermeasures will be based on the current status of the NPP such as burned-up fuel, cooling time, release pathways , etc. However, once the release has begun, off-site emergency actions will strongly depend upon one's ability to collect in a minimum of time enough data to characterize the accident, and to use these data to reconstruct the source term, i.e. the amount and nature of the material being released. New estimates of the source term are then fed into a dispersion-deposition model to obtain an updated view of the affected areas downwind, providing the decision makers with new information.
This paper concentrates on the use of the normalized least-squares method as a tool to reconstruct the source term used during several tracer experiments, as well as on a discussion of its applicability considering the time and data constraints of a real situation.

MATERIALS AND METHODS

Experimental and Modeling

The data set used in this work was generated during several (11) experiments conducted during the early eighties at the location of Mol, Belgium. These experiments consisted of releasing a known amount of a tracer gas (SF_6) from a height of about 2 m. Samples were collected at a number of stations situated downwind on arcs around the point of release, and covering a distance of a few kilometers. For details on the sampling technique as well as on the analytical procedures used to determine the air concentrations, the reader is referred to [1]. The release rate was kept constant throughout the experiment, and the time span between release and sampling was long enough to ensure that the plume had reached the observation point.
A data set of model estimates was computed for each tracer experiment using the segmented-puff dispersion deposition model ATSTEP [2]. This model has been chosen due to its degree of complexity which lies between a simple bi-gaussian model and a particle model, the reduced number of input parameters required, and overall speed, which are all elements of extreme importance in real-time response to an emergency. Model predictions were done assuming a stationary meteorology, i.e. wind direction, speed and atmospheric stability did not change throughout the experiment, and a constant release rate of 1 g s^{-1} .

Normalized least-squares method (NLSM)

This approach does not differ significantly from the one used in [3] for the source term reconstruction using the deposited activity after a nuclear accident, and it is based upon the assumption that the modelled quantities, airborne concentrations in this case, satisfy the equation:

$$L_i = Q\,L_i(P)$$

where the subscript i represents a given point, Q is the source term, and P summarizes all the parameters such as release height, wind speed, atmospheric stability, etc. Bearing this in mind, one can define an objective function F such that,

$$F = \frac{\sum_i (QL_i - Z_i)^2}{\sum_i Z_i^2}$$

with Z_i being the observed quantities, e.g. airborne concentrations and L_i the modelled quantities. The normalization to the sum of the Z_i ensures that all possible values of F are between zero and one, which makes the interpretation of the results somewhat easier. The source term (Q) can be determined by minimizing F, which is a straightforward procedure.

RESULTS AND DISCUSSION

A summary of the results obtained for 11 tracer experiments, is shown in the Table 1 below:

Case	Qreal	Qnlsm	Qinvgauss
1	2.2	1.1	1.2
2	3.2	6.7	6.5
3	3.2	8.0	10.4
4	3.2	3.6	4.1
5	3.4	3.5	4.4
6	3.4	2.9	2.8
7	1.0	0.8	1.1
8	3.3	3.1	4.7
9	3.3	12.0	13.1
10	3.5	1.6	1.4
11	4.2	2.7	3.4

Table 1. Comparison of the real source term (g s^{-1}) with the reconstructed using NLSM and that obtained by simply inverting a bi-gaussian model.

It can be seen from this table that the NLSM procedure yields (cases 4 through 8) an estimated source term that agrees fairly well with the real one, whereas the maximum difference (factor 4) was observed in case 9. In this particular case, the atmosphere was very stable and this resulted in a very narrow plume, as shown in Figure 1, causing that a significant number of receptors did not receive any signal at all. Even in this case (9) a difference of a factor of 4 seems quite acceptable, considering that the meteorological conditions were assumed constant throughout the experiments. Even though this method has proven to be a very useful tool for the estimation of the source term, its applicability to a real situation is very limited, and this is mainly due to the scarcity of observation points during the early phase of the accident. Furthermore, even if several observation points were available, they should be near the plume's center line to avoid situations similar to number 9 above, i.e. when the plume is very narrow.

The last column in Table 1 shows the source term resulting from the inversion of a bi-gaussian model and considering only one point, namely the one with the highest concentration. It is worthy to mention

that this procedure also yields significant results and has the advantage of needing very few observation points. Therefore, the reconstruction of the source term during an accident could be possible if one follows a monitoring strategy that focuses mainly on locating the plume's axis. A number of monitoring guidelines are still under development to achieve this goal.

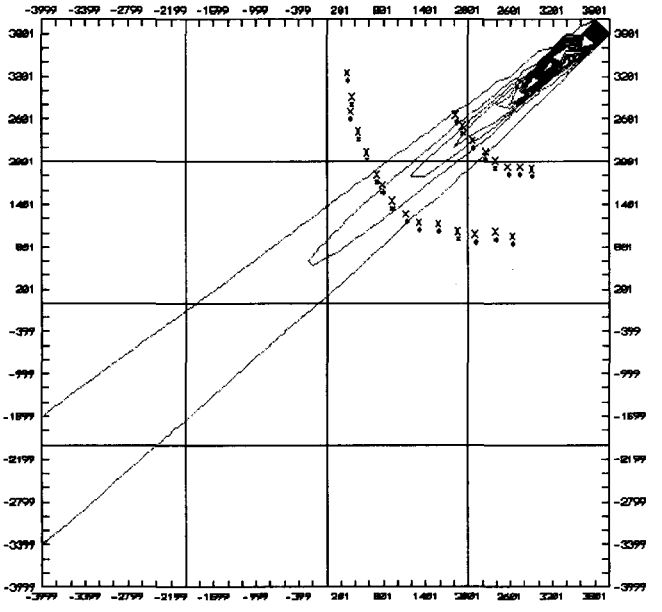

Figure 1. Plot of the resulting plume for case 9. The position of the source in this 41x41 grid has been chosen according to the prevailing wind direction (NE), and the resolution was set to 200m The location of the receptor points is also shown.

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REMOVING FOOD PROTECTION COUNTERMEASURES: THE USE OF LIVE MONITORING TO DERESTRICT MUCH OF THE CHERNOBYL-AFFECTED AREA IN CUMBRIA

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INTRODUCTION

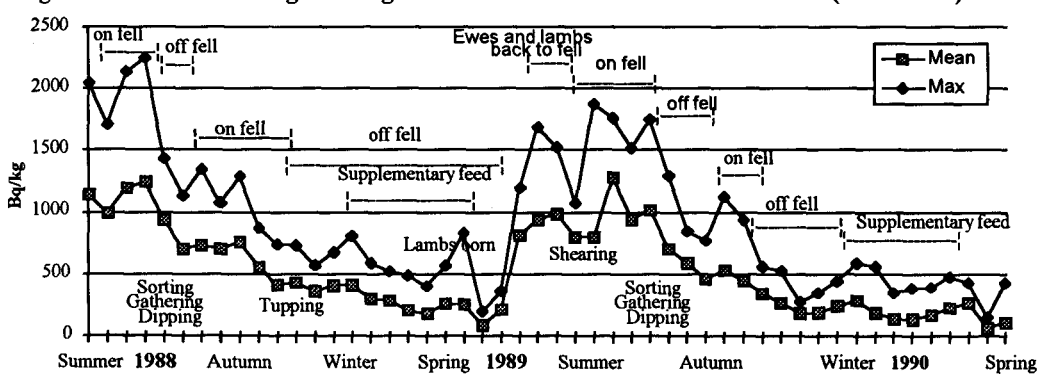
After the passage of the Chernobyl plume across Britain in 1986, some upland areas with peaty soil were significantly contaminated with Cs-137. The main effect on agriculture was the persistence of radiocaesium in sheep grazing these areas. A limit of 1000Bq/kg of total Cs in sheepmeat was introduced to protect consumers, and a system of live-monitoring known as mark-and-release has been applied successfully ever since, ensuring that no animal above this level of contamination could enter the food-chain (1). The level of contamination in the animals has fallen only slowly because of the nature of the soil (2), but over the years it has been possible gradually to reduce the area held under restriction. This paper describes the approach used to determine whether restrictions could be lifted on a farm-by-farm basis in England; similar methods have been used in other parts of the United Kingdom.

METHOD

The essence of all live monitoring is that animals are tested in the field using a hand-held, portable instrument; the calibration curve was established from comparison of live monitoring and accurate determination of activity levels in samples taken from animals after slaughter (3, 4). When animals are destined for the market, in the autumn, farmers in the restricted area are required to present them for monitoring before they leave the area; only those which are below the limit may be moved. A safety margin to bring them within the 97.5% confidence level is applied, because of the natural scatter of the calibration. Failures are then kept on low-lying land where the caesium levels are much lower and within a few weeks can go on to market. The number of animals failing this test in recent years has been minute, less than ten out of tens of thousands monitored each year, partly because farmers generally do not present their animals for testing until they are confident of passing, and partly because of the gradual reduction of radiocaesium in vegetation in the area.

Before removing controls on any farm in the area, it is necessary to be confident that no animal will be contaminated above the limit when it goes to market. A programme of derestriction monitoring has been carried out in the summer of each year since 1989; since 1990 every animal within a selected area was live monitored at a time corresponding to the annual peak of activity. Detailed studies have shown that the peak is reached between June and September each year (Fig 1) (4, 5).

Figure 1. Live monitoring at a single farm in the Cumbrian restricted area (1988-1990)



For this survey, farmers present their animals within 24 hours of bringing them down from the fells, which they can do as part of their summer clipping routine . This approach ensures that the true peak activity in the flesh is measured,as activity falls rapidly once the animals leave the fell, and so if every animal in a flock is below the limit (again with a suitable margin for counting uncertainty) levels will be well below the limit at market time.

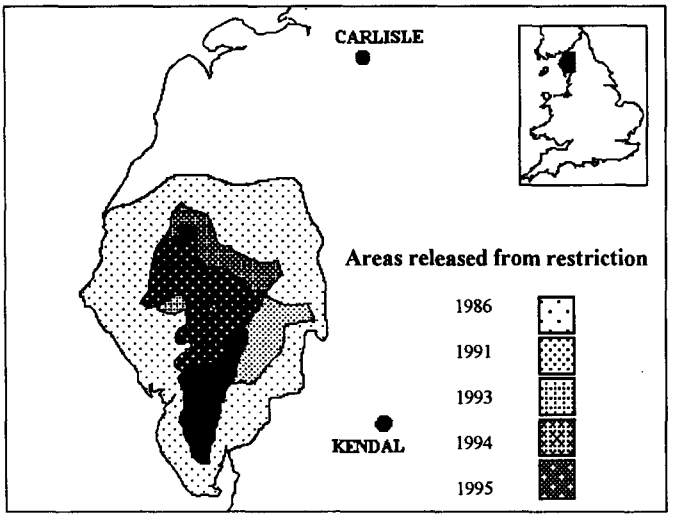
SELECTION OF AREA FOR DERESTRICTION MONITORING

This method is very resource intensive and it would not have been possible to cover the entire restricted area in a single survey. Parts of the area were only selected as candidates for derestriction when there was good reason to believe that many of the holdings tested would pass. Results of each year's mark-and-release monitoring have been an indicator of possible success, but these are mostly a reflection of farming practices. The best indicator has been the soil. Research over several years following the Chernobyl accident showed that within the broad class of organic and peaty soils there are significant differences in caesium retention properties between different soil types (see, eg, 2). Although caesium is recycled and reappears in vegetation each year on many fell soils, the peak level in vegetation falls annually at different rates. These results suggested only deep peat was likely to continue to be a problem in the longer term. Very detailed maps of the soil composition are available (6), and these enabled parts of the restricted area to be selected each year in the expectation that few farms would be likely to show failures.

RESULTS

Figure 2. Sheep grazing areas in Cumbria affected by Chernobyl fallout

In Cumbria, an area of some 197,000 hA, comprising 1670 holdings and containing some 867,000 sheep, was originally restricted in June 1986 on the basis of very cautious assumptions, but1520 holdings with 697,000 sheep were derestricted by September 1986. Fig 2 shows the area covered, and also shows how it has been reduced in recent years following the annual derestriction surveys.



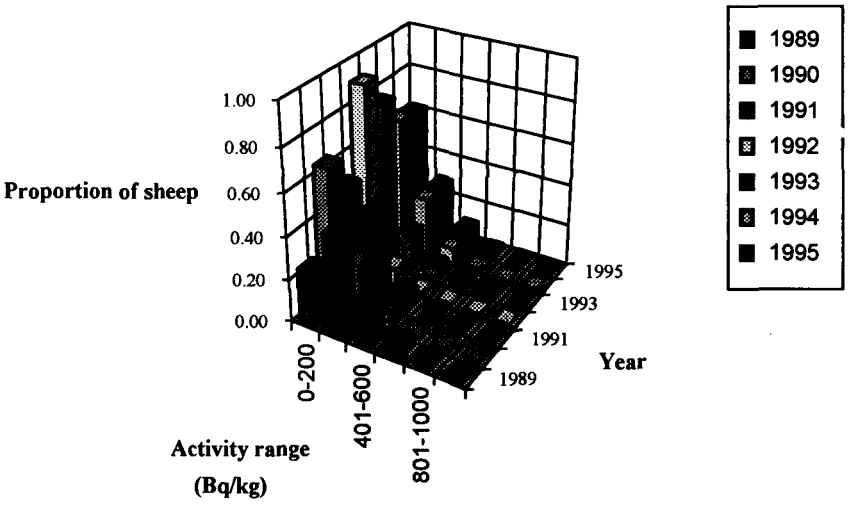
The Table shows how the numbers of holdings under control and the numbers of animals involved have been reduced, up to January 1995. In the summer of 1995 all the remaining area was surveyed; analysis of the latest results is still under way but it is hoped that the number of holdings remaining under control can be reduced to below 20. It is important to recognise that the whole approach is extremely conservative, and a single animal above the limit is sufficient to keep a holding under control. In all cases the vast majority of the animals show very low levels of activity.

Holdings and sheep in Cumbria affected by Government restrictions

Year	Number of Holdings	Number of Sheep
June 1986	1670	867,000
October 1986	150	170,000
December 1991	138	120,000
January 1993	126	110,500
January 1994	112	100,000
January 1995	66	68,819
January 1996	<20	≤ 20,000

Figure 3 shows the distribution of activity measurements for each survey in the last 7 years; although tens of thousands of animals have been below 200 Bq/kg, even at the peak time, the few results above 1,000 Bq/kg have been enough to keep the affected holdings under restriction and require the continuing use of the mark-and-release system.

Figure 3. Activity distributions of sheep in derestriction surveys



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Some aspects on improvement of the organizational basis for public protection of those residing under the conditions of chronic "emergency" irradiation.

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Republic of Belarus

We possess decennial experience of organizing and implementing protective measures for the population in the Republic of Belarus which turned out to be within the sphere of influence of various factors engendered by the Chernobyl NPP accident. It enables us to distinguish some important in essence, from our viewpoint, aspects on radiation protection and safety, whose underestimation might considerably affect adequacy and effectiveness of the entire system of protective measures carried out during different stages of development of a large-scale radiation accident.

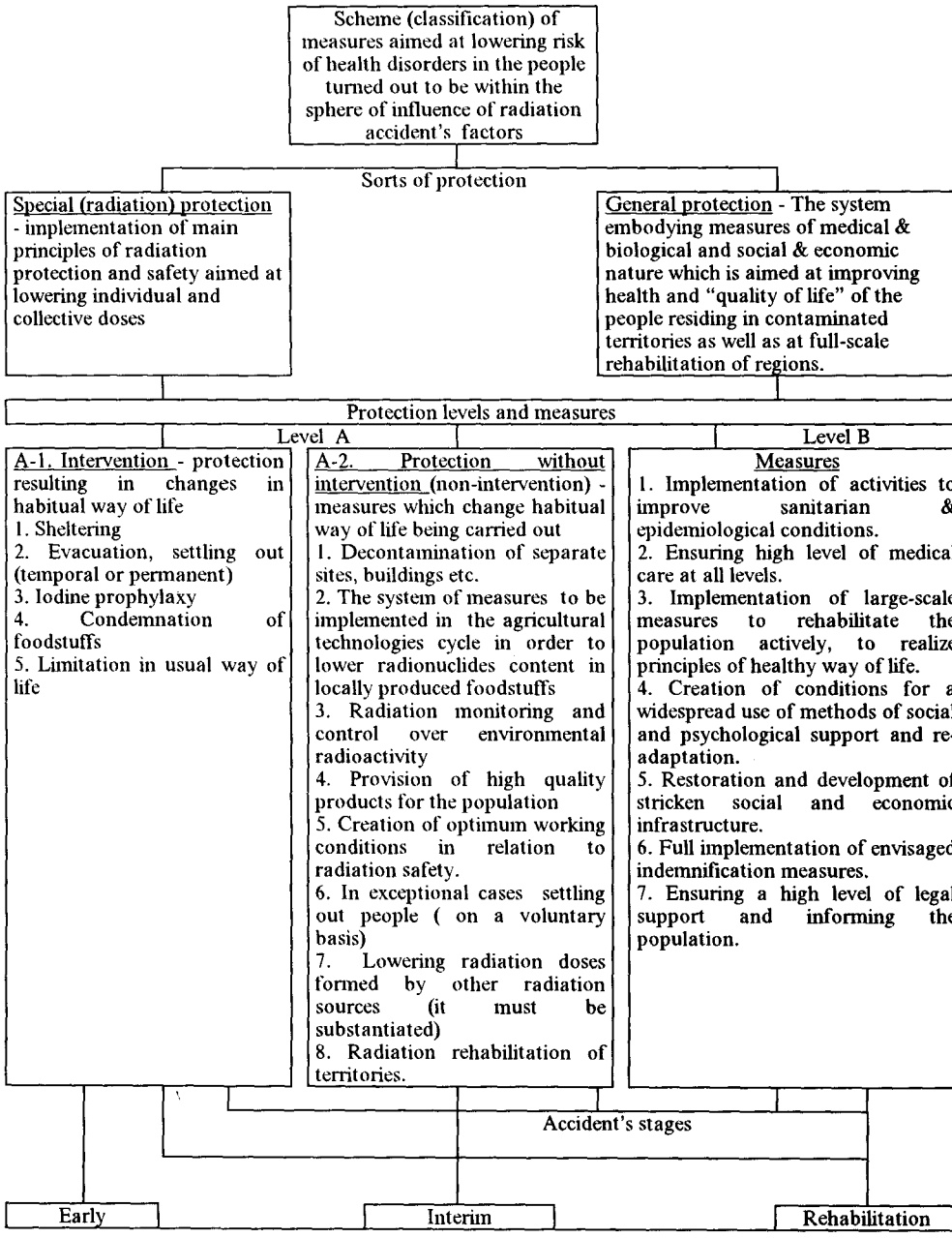
Firstly, definition of separate stages of an accident is based upon methodology which suffers from grave shortcomings. Thus, from the viewpoint of the NCRP, criteria offered to define an accident's stage are not always concrete. A range of reference-points (criteria) for defining a rehabilitation stage is especially broad. There are no numerical characteristics of the stage. It impedes its objectification that, in its turn, does not help to designate conceptual srategical aims and objectives for public protection.

Proceeding from this, we deem that in order to develop methodology of differentiating accident's stages it is expedient to introduce such an additional reference-point (a criterion) as "percentage of a life-long dose formed in people inhabiting radioactively contaminated territories". Doing so and taking into account general appropriateness of dose-forming through "emergency" irradiation, we suggest that one should consider a 50 per cent barrier of a life-long dose to be an upper limit for regarding an accident's stage as a rehabilitation one. It is believed that this approach will help

- to define more exactly time of accident's transition to the rehabilitation stage so that elaboration and implementation of adequate protective measures are underpinned

- to strengthen the system of protective measures since definition of the stage itself can render a positive psychological impact on the population.

Secondly, practice of radiation protection for the population affected as a result of the ChNPP accident convinces one that there is a necessity to draw up a unified international scheme (classification) of protective measures for the people turned out to be within the sphere of influence of an emergency radioactive release. Relying upon international radiation protection principles and experience gained in the republic, Belarus proposes a variant of general systematization (classification) of public protection measures for those affected by radiation accident's factors. Drawing up the given scheme, we proceeded from the fact that from the viewpoint of possible medical and biological consequences a radiation accident can be dangerous not only owing to a radioactive release, but owing to other factors which inevitably arise during any event jeopardizing welfare and health of lots of people.


Scheme (classification) of measures aimed at lowering risk of health disorders in the people turned out to be within the sphere of influence of radiation accident's factors
Sorts of protection
Special (radiation) protection - implementation of main principles of radiation protection and safety aimed at lowering individual and collective doses
General protection - The system embodying measures of medical & biological and social & economic nature which is aimed at improving health and "quality of life" of the people residing in contaminated territories as well as at full-scale rehabilitation of regions.
Protection levels and measures
Level A
Level B
A-1. Intervention - protection resulting in changes in habitual way of life
1. Sheltering
2. Evacuation, settling out (temporal or permanent)
3. Iodine prophylaxy
4. Condemnation of foodstuffs
5. Limitation in usual way of life
A-2. Protection without intervention (non-intervention) - measures which change habitual way of life being carried out
1. Decontamination of separate sites, buildings etc.
2. The system of measures to be implemented in the agricultural technologies cycle in order to lower radionuclides content in locally produced foodstuffs
3. Radiation monitoring and control over environmental radioactivity
4. Provision of high quality products for the population
5. Creation of optimum working conditions in relation to radiation safety.
6. In exceptional cases settling out people (on a voluntary basis)
7. Lowering radiation doses formed by other radiation sources (it must be substantiated)
8. Radiation rehabilitation of territories.
Measures
1. Implementation of activities to improve sanitarian & epidemiological conditions.
2. Ensuring high level of medical care at all levels.
3. Implementation of large-scale measures to rehabilitate the population actively, to realize principles of healthy way of life.
4. Creation of conditions for a widespread use of methods of social and psychological support and re-adaptation.
5. Restoration and development of stricken social and economic infrastructure.
6. Full implementation of envisaged indemnification measures.
7. Ensuring a high level of legal support and informing the population.
Accident's stages
Early
Interim
Rehabilitation

THE ORGANIZATION OF MONITORING SYSTEM IN THE DISTRICTS AROUND THE NPP TO PROVIDE RADIATION COUNTERMEASURES

Ivan Skuratovich,Ivan Matveenko

Committee for hydrometeorology,Minsk,Belarus

There are created the system of prevention and activities during the emergencies in the Republic of Belarus. This system should provide a detection of the beginning of natural calamities, industrial accidents and disasters, a prognostication of their courses and possible damage and a working up of concrete proposals of protective measures realization and consequences liquidation.

Taking into account how great damage was done to the Republic of Belarus following the accidend of only one of the blocks of Chernobyl NPP, in this system the special place is alloted to prevention and activities with accidents on the nuclear units. To detect beginning of the accident on the nuclear unit it is necessary to create the automatized radiological and meteorological monitoring system especially in the regions adjoining the working NPPs. There is not the own NPPs in the territory of Belarus but the NPPs in operation are in the neighbour countries such as Lithuania [Ignalinskaya NPP], Russia [Smolenskaya NPPs] and Ukraine [Chernobylskaya and Rovenskaya NPPs]. All these NPPs are situated near the territory of Belarus.

A creation of that system on all the territory of the Republic takes substantial capital investments and much time. Therefore the works are realized in stages. In the first stage at the expense of means of the EU Comission according to the project worked up by the firm "Consulting Group" [Great Britain] the firm "Horman Systemtechnik" [Germany] mounted 9 automatic points of gamma-level measuring in the region adjoining the Ignalinskaya NPP. [The siting scheme is added]. Futhermore the automatic points measure availability or absence of liquid precipitation and have possibility to transmit information to the centre of data collection and treatment by radio-line. In addition to the available meteo-stations network [the nearest one is at 60 kilometres distance] is mounted the automatic meteorological station in Drysvyati at the distance of 4 kilometres from NPP. The automatic points work in a continuous rate of measuring, and information is transmitted to the Centre once an hour. In the case of gamma-level raising to the fixed threshold is transmitted an extraordinary report, and the measuring results are transmitted every 10 minutes. By the project GAMMA-1 is envisaged a delivery of the mobile laboratory. This laboratory has a possibility to mount in necessary parts e.g. in the contamination zone 4 additional automatic points with autonomous feeding. It allows to receive the more detailed information from the most polluted regions.

The created network of gamma-level and meteorological parameters measuring allows to detect opportunely a beginning of the territory contamination in the case of the accident on the Ignalynskaya NPP. Futhermore using the data of radiation and meteorological monitoring we can prognosticate what parts of the territory will have different contamination levels and where it necessary to carry out the protective arrangements. The possible protective arrangements include:

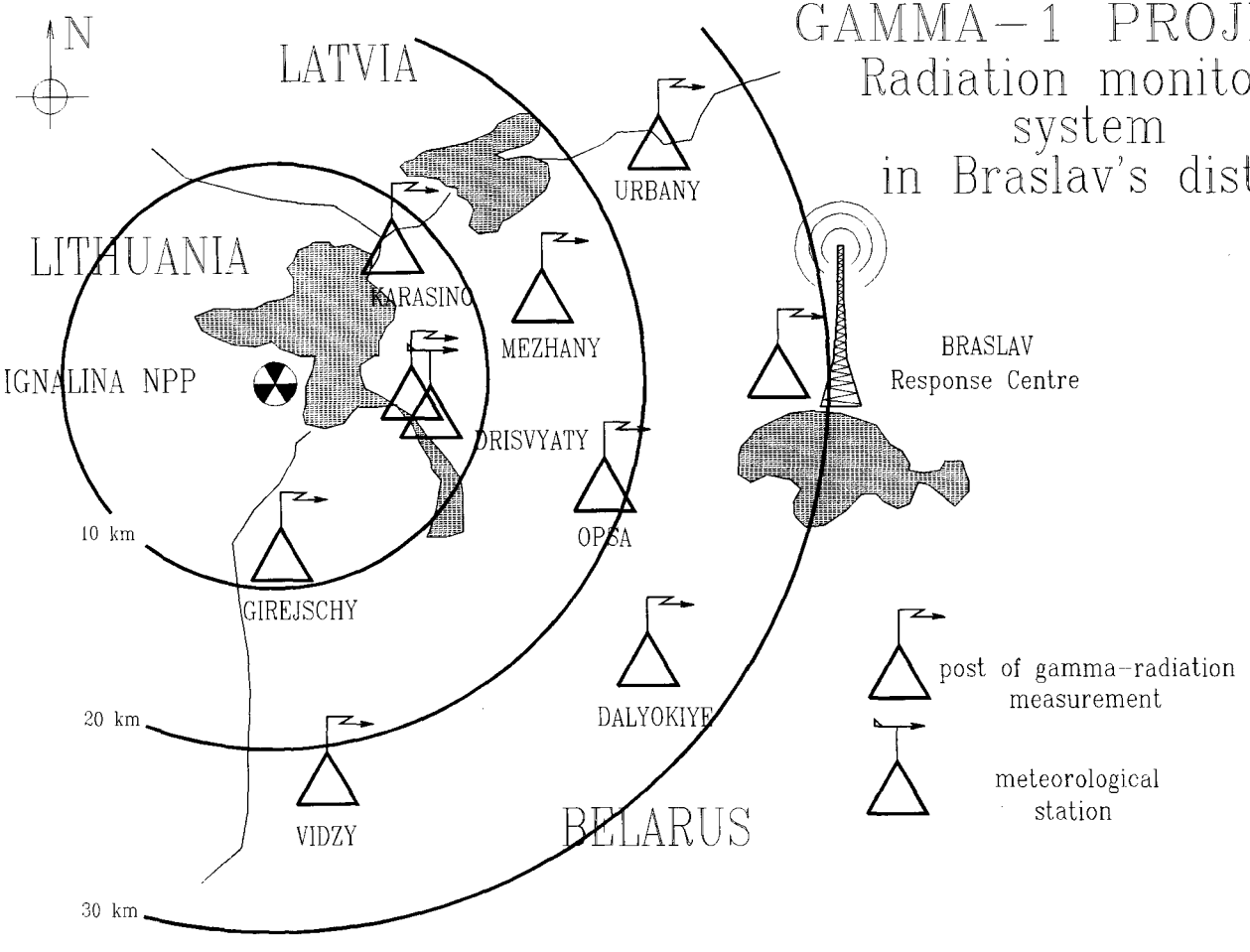
- distibution of iodine tablets;
- sheltering of population and animals;
- limitation of polluted food products consumption;
- desactivation;
- evaquation.

It is very important to determine an economic effectiveness of the realized protective arrangements.

CONCLUSION.

An effectiveness of this system using will depend on amount and quality of the measured meteorological and radiological data. It is necessary to create in futher stages the automatic monitoring network in the other regions adjoining the working NPPs and on all the territory of the Republic.

It is necessary to provide the data exchange between the countries of Eastern and Western Europe. Using the automatized meteorological and radiological monitoring network it is necessary to improve the methods of contamination detecting, monitoring data treatment, contamination levels prognostication and regular studying of operational personnel.


GAMMA–1 PROJECT
Radiation monitoring system in Braslav's district
N
LATVIA
LITHUANIA
BELARUS
IGNALINA NPP
KARASINO
MEZHANY
URBANY
ORISVYATY
OPSA
BRASLAV
Response Centre
GIREJSCHY
DALYOKIYE
VIDZY
10 km
20 km
30 km
post of gamma–radiation measurement
meteorological station

The measurement organization within the radiation monitoring network on the territory of the Republic of Belarus.

Pikalov V.N., Matveenko I.I., Samsonov V.L.

The Committee for Hydrometeorology of the Ministry on Emergencies and Population Protection from the Chernobyl NPP Catastrophe Consequences. Minsk, Belarus

The measurement system of the environmental object radiation contamination is the main component part of the radiation monitoring functioned on the territory of Belarus.High sensitivity of the apparatus of the measuring complex allows to receive trustworthly estimation of the environment radiation contamination.

The radiation measurement's estimation of the environmental object (water,earth,air) carries out by spectrometral, radio-chemical and radiometric methods.

The measurement's results accumulated in the automatized data base and uses for estimation and forecasting of radiation situation on the territory of Belarus.

THE DEVELOPMENT OF THE PROMPT SYSTEM OF GAMMA RADIATION MONITORING IN FR YUGOSLAVIA

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The Institute of Nuclear Science Vinča, Belgrade,Yugoslavia

ABSTRACT

The needs for and characteristics of the Prompt System of Gamma Radiation Monitoring are analysed. The basic structure of the new system, based on the PC computer is presented. The system consists of the central unit and a number of field stations connected via telephone lines. GM counter is used as a detector. The software is written in Visual Basic. The system is capable of performing a variety of tasks: dose rate measurement (analogue and digital), graphical presentation of the results in the given period of time, data base formation and management, data transmission and preparing reports.

The basic characteristics of hardware and software of the system, performance specifications and future plans are presented in this paper.

INTRODUCTION

Besides large benefit especially as energy resource the nuclear energy represents potential danger for humans and the environment. The main danger is radioactive contamination. In normal working conditions contamination is small, controlled and under regulation levels. But in accidental situations such as Chernobyl accident the situation is quite different. Than we can expect large contamination area and population exposure for a long time.

In situations like those the main task is to protect people from internal and external contamination. There are a lot of possible technical, social, economic and political measures. The cost could be enormous. Because of that, all measures have to be optimised. The main condition for this is very early accident detection. This is the reason why prompt monitoring system must exist. It is very important in the case when accident happens abroad and other systems fails (as it has happened in the case of Chernobyl accident). The information of the prompt monitoring system that an accident have happened somewhere is the basis for the response plan activation.

In this paper basic design goals of prompt system monitoring are analysed as well as the some technical characteristics of the system which is under development in Yugoslavia. An example is given as well.

BACKGROUND INFORMATION GOVERNING THE DECISION ON SYSTEM DEVELOPMENT

Strong regulation demands in nuclear energy applications and rapid technological development in the field of electronics and computers enables qualitative improvements in environmental monitoring systems. Global effects of Chernobyl accident have accelerated the development in this field.

Developed western countries, with strong nuclear industry, are already covered with computerised monitoring networks. There are several commercial possible solutions, depending on selected detector system and requested set of information.

At this time the Yugoslav national system for accidental detection is based on several measuring point, spread over the country. The absorbed gamma dose rate is measured manually every hour and, in the case of exceeding the declared threshold level authorised centre alerted.

Besides those discrete measurements on three selected locations, continual gamma dose rate monitoring by pressurised ionisation chambers (Model RSS-112 Reuter Stokes, USA) is established. Additional two ones have already be planed and they are about to be installed.

Contamination levels and the whole response system in Yugoslavia after the Chernobyl accident practically proved the necessity of the existence of automated monitoring network. Finally, in the process of changing the basic Radiation Protection Act (3) this system is introduced as governmental responsibility.

According to the presented scientific and organisational background information it was necessary to choose further course of action. Taking everything in to the consideration including the finance it was decided to develop one own system. Existing monitors will be incorporated into the developed one.

DESIGN DEMANDS AND GOALS

The main construction demands for the prompt monitoring system are analysed here. The fastest and most convenient way for prompt monitoring is based on gamma dose rate measurement at the background level. It determines the lower measuring range limit. It depends on location characteristics. According to the results of environmental monitoring over years, the value of 100 nGy/h is good enough. If detector enables it is recommended to use a lower value. The upper limit must be above local background variation in the worst case. The isodose lines at Chernobyl accident must be taken into account too. The upper limit of 10 mGy/h is appropriate. But if detector enables it is recommended to use even a higher value.

The measuring error depends mostly on detector and electronics, system function and costs. The compromise leads to standard error of ±20% at ^{60}Co gamma ray energy. The system must measure gamma dose rate in the energy range 0.1-2.5 MeV with energy error of ±30%. Statistical error is in connection with detector efficiency and response time. A value of ±20% is accepted.

Prompt system response time depends on detector response time and transmission time. The appropriate value of detector response time is several seconds and even the value of several tenth of seconds is not critical.

The parts of prompt gamma monitoring system must operate in the field condition. They have to be adjusted to that climatic condition.

Besides of satisfaction of all this demands the goal of Yugoslav prompt system monitoring is to be inexpensive, based on modern electronic components, computers and data transmission infrastructure. At the moment the telephone link is the basic link.

The system consists of a central station and several field stations covering state area. Data transfer between them must be automated. The field station have to send last defined series of measurements on the central station request. Only in the case that the result of measurements at any field station indicate an accident or accident possibility, this station must send measurements results permanently until central station stops it.

Data manipulation and presentation must be at the top level. It means that modern inexpensive PC computers and peripherals as well as its software must be used.

SYSTEM CHARACTERISTICS

According to the above mentioned design demands and goals the prompt gamma monitoring system is developed (1,2). It consists for now of a central station and several field units connected as presented on figure 1.

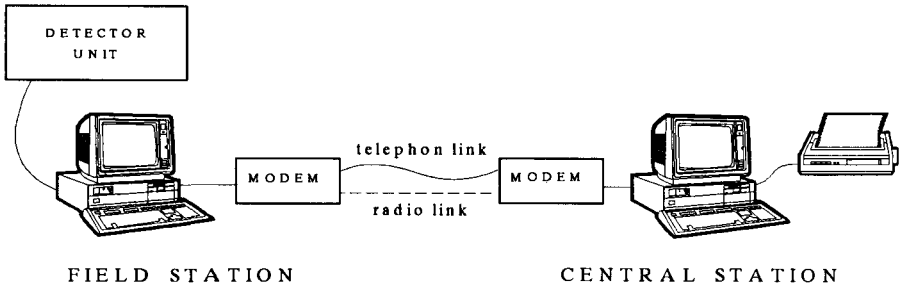

Figure 1 Prompt gamma monitoring system configuration

The detector unit consists of the very sensitive GM counter (ZP1220) with some electronics (high voltage, pulse shaping and forming modules). The pulses are counted in a counter computer card. Ratemeter function is realised by software. By software are realised all other functions as well. One of the important functions is decision making function. Whether the measurement result is a background variation or an accident has occurred ? This function is supported by data base of dose rate measurements at that specific location. This allows the decision function to be self adjusted during the

years and seasonally. The very important software part is communication package. It is based on modern software for PC computers.

Central station have a field station function and an integrating function. All data are here analysed, stored and presented. Decision algorithms are the same as in field station . There is a new software part for area contamination analysing and forecasting. This part is very important in a emergency response plan realisation.

The operation of the prompt system monitoring is very promising on PC computers under Windows™ operating system. So, the program is written in Visual basic™ language. It enables several functions in time: measurement, several types of presentation, data storage, data sending. Besides, data presentation is powerful. Dose rate measurement results are presented in analogue (gauge or bar) and digital form. Analogue form enables easy viewing of full measuring range and alarm levels. Digital form is more accurate. On the PC display a special window is realised for last 24 measurement presentation in a histogram form. It enables daily changes observing. How a PC display looks like during the measurement on field station is presented on figure 2.

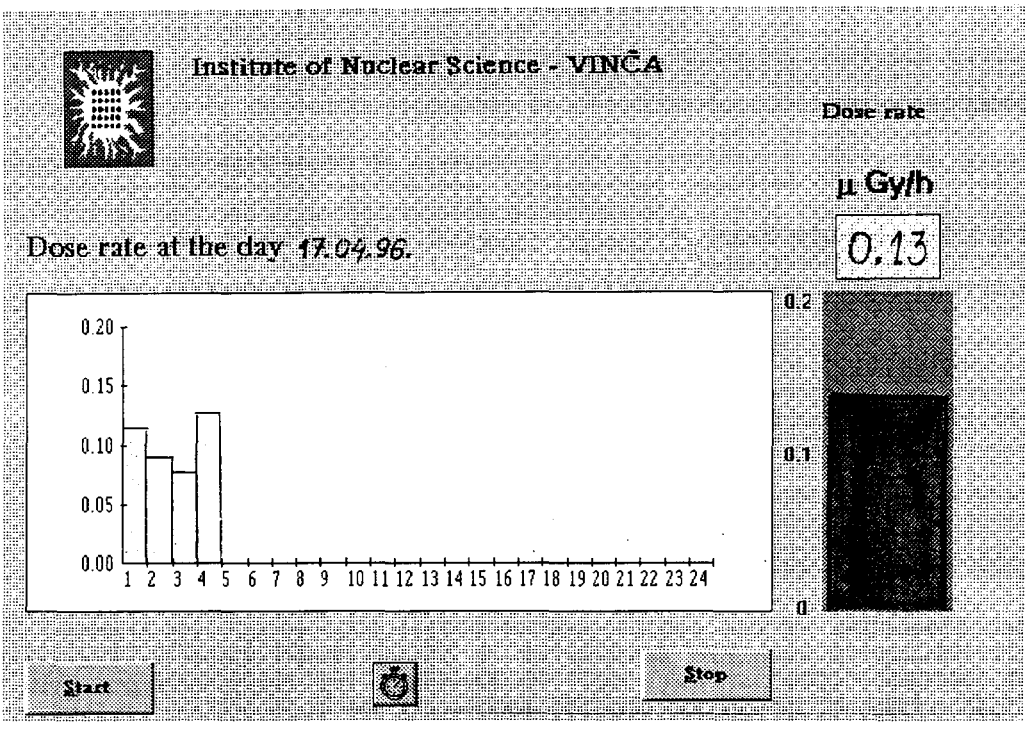

Figure 2. The display of gamma prompt monitoring system during data colection

CONCLUSION

This paper presents new Yugoslav Prompt Gamma Monitoring System for large accident detection. It is based on modern components. Because of this it is cheap, easy for using and with additional capabilities for measurement results presentation, data storing and manipulation and with a appropriate humane-computer interface. It is convenient for small countries.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Abstract No. 20870
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PAPER TITLE Radiation Monitoring of Environmental Contamination around Mayak Atomic Facility and Effective Use of Its Results

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Radioactive contamination in the Urals region resulted from more than 4 decades of the activities of Mayak Industrial Association (Sourth Urals). A specific feature of radiation conditions on the territory around Mayak is the long-term exposure of the residents due to the presence of Sr-90 and Cs-137 among the radionuclides contaminating the environment. Toprovide informational basis for decision-making about nature protection and environmental safety measures on the territory around Mayak facility a comprehensive radioecological monitoring has been organized. Attention is focused on monitoring natural media, sources of industrial contamination, population's health status, etc. The priority task to be solved by radioecological monitoring is to establish a system of collection and analysis of data on radioac - tivecontamination of the environment, setting-up and updating specialized data bases, assessment and prediction of changes in environmental conditions due to impacts of industrial contamination, infor - tional support to long-term and short-term measures aimed at controlling the environmental situation including emergencies. Nature protection activities are aimed at the solution of the triune task: ensuring safe life conditions for the population, restoration and protection of ecosystems, sustained development of the region's economy.

THE NATIONAL RADIOACTIVITY MONITORING NETWORK OF THE NETHERLANDS

Siebren T. van Tuinen[1], J.E.T. Moen[1], J.F. van Sonderen[1], C.P. Deurwaarder[2], A.S. Keverling Buisman[2], C.J. Bartels[1], and J. Dresmé[1]

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INTRODUCTION

In 1988 the Dutch Government decided to integrate the radioactivity monitoring network of the National Institute of Public Health and the Environment (RIVM) managed on behalf of the Ministry of Housing, Spatial Planning and Environment (abbreviated in Dutch: LMR) and a similar network of Ministry of the Interior (BMNI) into the National Radioactivity Monitoring Network (in Dutch: NMR).

After the Chernobyl nuclear power plant accident in 1986, the LMR network was developed within the framework of the National Plan for Nuclear Emergency Planning and Response (EPR) as an early warning and monitoring instrument for future nuclear accidents (1). In March 1990, four years after the Chernobyl reactor accident, the LMR was officially opened. The LMR network consists of 58 stations measuring the natural background γ-radiation level. In 14 of the 58 stations monitors are installed to measure the airborne α/β-radiation level. Because different radioactive nuclides may be released during a nuclear accident the LMR also contains two nuclide-specific monitors which facilitates translation of the measured α-, β- and γ-radiation levels into an effective dose to humans.

The BMNI network is an information system primarily for local management of accidents. This network was designed and built for the Ministry of the Interior by the Netherlands Energy Research Foundation (ECN) in the period 1987-1993. It contains 252 stations measuring the surrounding γ-radiation levels. The network conforms to the management structure of the Ministry of the Interiors Fire Services Department in its hierarchical build-up of four levels: national, provincial, regional and local stations. Measurements are performed at the local level, while the responsibilities for countermeasures during accidents are at the regional level. In February 1993 this network was officially opened.

THE INTEGRATED NETWORK

The integrated network, called the National Radioactivity Monitoring Network, must satisfy the demands of all parties involved. The main objectives of measuring the radiation levels in the Netherlands are to provide:

- early warning against major nuclear accidents
- information on the geographical distribution of radioactive contamination during an accident
- information on the actual radiation doses to the population during an accident
- information on radiation doses to regional, provincial and national authorities
- general information on background levels

The solution was found in maintaining the structures of the existing networks as much as possible and to build an integrating structure, including an overall database at the National Institute of Public Health and the Environment (RIVM) at Bilthoven. In this way the integration is reached by exchange of data, which proved to be much more cost-effective than connecting the different types of monitors and computers using a technical solution.

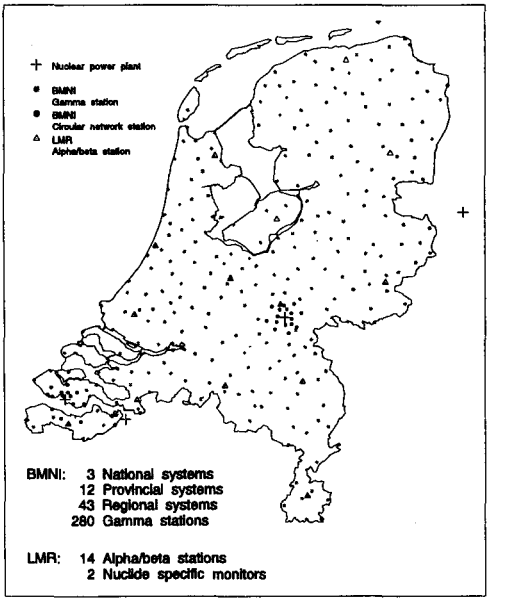

Figure 1. The NMR locations

RADIATION DETECTION

The NMR consists of 280 measuring sites that contain a proportional counter tube (Bitt RS02/1) for the determination of the gamma ambient dose-equivalent rate $H^*(10)$. The distance between sites is typically 15 km. Compared to other European countries, the Netherlands' network, as in Germany and Austria, has a relative high station density (about one station every 150 km^2). A double circular network of monitors has been installed around the two Dutch nuclear power plants: one ring at 5 and one at 10 kilometres from the power plant. These rings immediately provide emission profiles in case of accidents with the nuclear power plants and reduce the necessity for emergency staff to take measurements in highly contaminated areas. Also around the two nearby foreign nuclear power plants some monitors have been installed where the two rings intersect Dutch territory.

An airborne-activity monitor (FAG FHT 59S) determines the aerosol-bound artificial gross-beta activity concentration in air at 14 sites as an indication of inhalation dose. These 14 sites also contain a proportional counter tube to compare artificial gross-beta measurements with gamma radiation levels at the same site. Figure 1 shows all measuring locations of the NMR network.

Two nuclide-specific monitors have been added to the network at RIVM. One, the iodine-monitor, measures the gaseous radioactive iodine concentration in air and the other, the nuclide-specific gamma monitor, identifies the order of contamination of radionuclides by γ-ray spectroscopy.

DATA MANAGEMENT

The special Dutch government emergency telephone network is used for most of the data acquisition. Work stations are set up regionally (sections of provinces, 43 in total at the moment), provincially (12 in total) and nationally (3 in The Hague; a main system, a back-up system and a reserve system in case of malfunctioning of the main system). Acquisition of gamma data takes place once every hour in a hierarchical manner: a regional network calls the sites in its region and collects data, then the provincial work station calls the regional station in its area followed by the national work station, which collects the data from the provincial stations. At the end of a complete upward acquisition cycle, all data are distributed again to the 43 regional work stations. HP9000/340 computer systems, controlled by the regional fire brigades, have been placed in the regional centres. At the National Coordination Centre in The Hague, HP9000/400 computer systems process and store the data. These, as well as the data from the 14 aerosol monitors, are sent to a central database located at RIVM.

NETWORK ALERT

Twenty-four hours a day, every ten minutes, radiation levels are measured at nearly 300 sites in the Netherlands. When the network level is exceeded at one of the sites, the station processor automatically calls the NMR computer at RIVM, which in turn activates a pager carried by specially trained RIVM staff members. The most recent data are sent with the alarm message to the personal computer of the staff member on duty. Ten-minute values of the previous four hours can be examined using NMR presentation software running on personal computers stationed at the homes of RIVM staff. When the alarm level of one of the gamma counters is exceeded, it is followed by an automatic validation procedure: an alarm signal is sent out to the pager (via the NMR computer) only when at least two monitors within an area of 40x40 km^2 have their alarm levels exceeded. The chances of a false alarm due to monitor failure are greatly reduced by this measure.

In 1995 the NMR network performance was tested against the specifications for an adequate alarm response: signalizing an enhanced radioactivity level within 1.5 hours to the central database at RIVM. The tests were performed by simulating a radioactive cloud passing by over the Netherlands. With the help of the regional fire brigades' staff nearly half of the total amount of measuring stations was alarmed during those days. Because of the positive test results the NMR network was declared operational December 1, 1995. It will be officially opened early in 1996.

ACCURACY OF MEASUREMENTS

There are some specific scientific aspects related to the integration of the networks. The reproducibility of the γ-radiation measurements is high (within 3%). As a result, the data can be analyzed to determine and discriminate between the variations in the natural background γ-radiation levels (between 60 and 100 nSv/h) (2). Two γ-radiation measuring stations, one (old station) belonging to the LMR network and one to the BMNI network, are situated at a distance of about 50 metres apart in Bilthoven. The monitors of the LMR are positioned at 3.5 metres height (on top of the measuring station), while the monitors of the BMNI are positioned at 1 metre height (at ground level). Since the monitors are identical, an intercomparison of the measurements is now in progress, which may provide a possibility to separate the different contributions of

cosmic radiation, terrestrial radiation, airborne and deposited radioactivity to the total γ-radiation level. Typical day/night variations in the natural γ-radiation levels were observed for some stations (*Figure 2*). Inspection of the location where the variation was observed showed that for the upper curve (station 01w501 in Kaatsheuvel) the detector was located in the direct vicinity of a shopping centre. During the day a lower γ-radiation level was seen due to the parked cars shielding the emission of γ-radiation from the used building materials in the car park.

For the lower curve (station 22w502 in Heerhugowaard), where an opposite variation in the γ-radiation level was seen, the detector was also located near a car park. However, in this particular case the car park was in front of a block of flats. Assuming the residents to be at home during the night, the γ-radiation level in this period was lower as a result of the shielding by the parked cars. This is confirmed by the measurements in the weekend. The absence of cars in the weekend resulted in a γ-radiation level for station 01w501 without dips in that period. Assuming that the residents in the block of flats also spend some time elsewhere in the weekend, the day/night variations should continue, which was indeed observed.

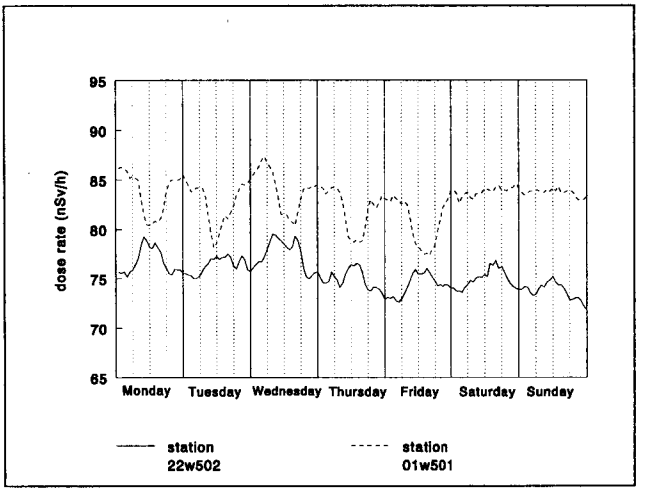

Figure 2. γ-dose rates for a week at two stations in the Netherlands.

These fluctuations and other variations in monitoring levels, with no relationship to an actual accident, can be ascribed to variations in natural environmental conditions. This illustrates that proper scientific judgement should be applied in every activation of alarm levels. RIVM provides this expertise to the EPR organisation.

NETWORK MANAGEMENT

The nuclear emergency response organisation operates on the assumption that nuclear accidents are by definition unpredictable, but do require immediate response. Therefore the network must be permanently maintained. RIVM manages the National Radioactivity Monitoring Network, providing quality assurance, developing and adjusting standard procedures for daily control, providing training of RIVM staff and financial management. ECN is subcontracted to maintain all gamma counters and the work stations. For this purpose a separate maintenance work station is operational at ECN. It is this combined network management that makes the NMR a reliable early warning network for nuclear accidents.

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EMERGENCY MANAGEMENT IN NUCLEAR ACCIDENT SITUATIONS THE DISASTER EXERCISE 1995 'NORTHERN LIGHT'

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INTRODUCTION

Emergency management does not only start after something has happened. Initially, a feasibility study usually assesses the risk for technologically critical processes and applications. Preventive strategies will be employed both in the administrative and technical field to minimize risk. Technical solutions will increase inherent safety or provide monitoring of critical components. Administrative action would result e.g. in restricted access, training programs, or detailed operating protocols. A final stage would be preparation for remedial action and defining the groundwork for emergency management in cooperation with civil defense forces.

Appropriate precautions will be based on hazard potential, which is inherently substantial when dealing with nuclear accidents. Being the last line of defense, the civil or military defense forces will be involved if a major disaster occurs despite all precautions, overpowering on-site crew capabilities. For major disasters requiring even international assistance, the United Nations Department of Humanitarian Affairs has started to conduct disaster preparedness exercises to improve cooperation and communication among the international relief teams and the local authorities. The EXERCISE '95 was organised by the Russian ministry for disaster management simulating a major accident in an atomic power plant located on the Kola peninsula.

SCENARIO

The technological scenario was modelled as a major accident in a pressurized water reactor (VVER-230/213). After depressurization in the vapor generator a general malfunction of the emergency power systems was assumed, resulting in failure of the emergency core cooling system. As a consequence of pressure build-up and additional failure in the main isolating shut-off, radioactivity was released into the environment through a safety valve for approximately 9 minutes. Then auxiliary power can be restored by the reactor personnel, the safety valve is closed, core cooling initiated, and the reactor attains a safe state. A significant portion of the uncovered core is assumed to be released as radioactive vapor into the atmosphere as a supersonic jet from the stack. Initial assumptions rate the magnitude of the total release up to roughly 10% of core activity, with a characteristic distribution regarding noble gases, iodines, and heavier elements, resulting in major contamination of the surrounding areas. The game-weather assumed predominantly easterly winds, causing trans-border effects as the radioactive cloud is blown towards the Finnish border. Thus international assistance was justified by the scale of the accident.

Preliminary action of the Department of Humanitarian Affairs was geared towards setting up of an On-Site Operations Control Centre (OSOCC) to coordinate the international efforts and serve as a head quarter for communication to local authorities. All participating teams were supposed to furnish a liaison officer to facilitate the communication of requests to the team and results back to OSOCC. International observers were monitoring the results achieved by each team on various missions as well as the recommendations of the expert groups.

OBJECTIVES

Within the exercise different objectives were to be pursued, both on the scale of international cooperation and particular to every team, respectively. An overview of the most important objectives is contained in the following list.

- Checking applicability of disaster preparedness and overall readiness to perform specific missions in contaminated areas (e.g. reconnaissance, decontamination).

- Assessing possibilities of international scientific and engineering support for decision making on matters of radiation protection. Determining the extent of the incident and providing the relevant information for the decision making process.
- Investigating the mechanism of international cooperation in case of a nuclear accident with trans-border consequences. Developing of practical strategies to cooperate in an effort to render urgent help in highly contaminated areas.
- Providing an opportunity of practical work for experts and field teams concerning counter measures in nuclear accident situations. Study of practical experience regarding organisation and implementation of emergency measures.

Among other nations, Austria was present with a team of AFDRU (Austrian Forces Disaster Relief Unit), manned by 30 members of Austrian NBC-forces and specialists from Seibersdorf. Approximately 20 tons of equipment were air-lifted to the affected area on the Kola peninsula, comprising of 3 search troops with vehicles, heavy decontamination equipment, as well as a command post and evaluation centre. Associated ancillary equipment included power generators, sanitary facilities, heating equipment and medical supplies.

The composition of the team was carefully selected to sustain prolonged operation in the field. Medical care, decontamination strategies as well as dosimetric coverage of the personnel were considered, to name just a few items. Experts from Seibersdorf assisted with special tasks regarding radiation protection and scenario analysis. In the preparatory phase much attention was paid to the selection of instrumentation and development of new techniques to implement a successful system concept, a cooperative effort between Seibersdorf and the NBC-division.

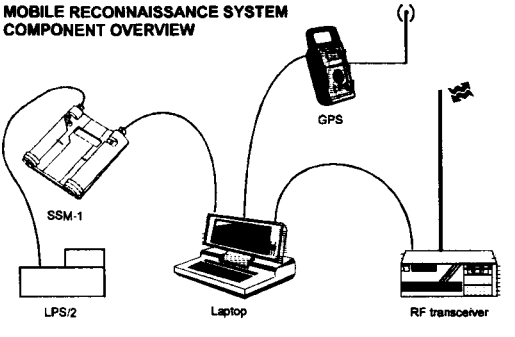
SYSTEM CONCEPT

Seibersdorf was the main contributor in defining the system concept for measuring purposes and situation reporting [1]. A cooperative project was launched together with the NBC-school to define the system requirements, which should allow a self-consistent operation after deployment into affected territory. Basic requirements were considered to be:

- Monitoring of environmental radiation levels at the camp and surrounding areas
- Determination of local weather situation, communication utility for general weather data
- Possibility for measuring of food stuffs or soil samples
- Autonomous mobile reconnaissance system
- Flexible command and evaluation centre
- Scenario analysis and forecasting tool (data interpretation & temporal development)

All of the above functions were implemented to some degree: a satellite telephone and computer hook-up served as communication utility to supplement the local weather information, which was constrained to the camp site; a NaI-based food stuff probe was also used for approximate determination of nuclide composition in soil samples, because a HPGe-detector was not considered feasible for extended field deployment.

The mobile reconnaissance system was a prototype development, based on an Aerial Monitoring System, developed and discussed at Seibersdorf [2]. A laptop computer monitors and profiles ambient radiation levels as determined by a radiation survey meter and high sensitive probe. A satellite based GPS system is used for on-line position detection. The current position is registered on the laptop and all measurements are annotated with positional



and timing coordinates. A RF-transceiver digitally broadcasts position and measurements as well as messages to the headquarter and relays commands to the field team. The headquarter may track up to 3 field teams simultaneously and communicate up to a distance of approximately 40 km. All data are stored internally in case of communication failure.

A scenario analysis tool (MIDAS [3]) was used to correlate the space/time annotated radiation measurement data and interpret them within the framework of the overall situation. Based on current and game-specific weather data, actual reports or projections of the radiation situation could be calculated and for instance heavily affected areas selected for immediate remedial action (evacuation, etc). Hot spots induced by precipitation (rain during passage of the radioactive plume) were modelled based on predefined weather information. These forecasts were used in mission planning for a reconnaissance trip to a heavily affected region on the third day after the accident. Artificial soil samples (provided by the organisers) were then used as corrective inputs to the model calculations, adjusting the assumptions of the source term.

Various inputs simulate the accident (source term, physical properties, local weather). A complex dispersion algorithm tracks individual elements of the release through a 3D wind field. If available, upper air data may be utilized to define wind shear layers aloft. Transfer factors within the model allow calculation such as estimated ground deposition, air-borne iodine, or impact on the food chains. All results are superimposed on a digitized map of the environment, allowing even demographic analysis (population affected, calculation of man-Sv).

RESULTS

The Austrian team and their systems used throughout the EXERCISE '95 were highly acclaimed by the international community. The mission objectives - reconnaissance work, decontamination, medical assistance, decision making support, and expert opinion - could be covered successfully. The scenario analysis - refined with results of simulated measurements - proved to be consistent with Russian model calculations used in preparation of the simulation. The mobile reconnaissance system - and specifically the automatic data acquisition and correlation of measured data with geographical position - significantly contributed to the overall success in forming a clear picture of the large scale situation. This might even be more important in a realistic scenario, where stress and human error can introduce additional problems with data integrity.

CONCLUSION

Emergency management on an international level requires major organisational efforts, in this case furnished to a great degree by the local UN coordination centre. To provide assistance effectively, the international teams have to operate autonomously in the affected territories. This requires not only a high level of training and expertise, but also a sophisticated technical infrastructure. Regarding reconnaissance, results of many different measurements have to be mapped and interpreted to form a comprehensive picture of the overall scenario and its implications. This information might then be used for planning of relief missions to heavily affected areas and in decision support to local authorities. Especially with nuclear accident situations, specific emphasis has to be placed on the future development of the situation. Forecasting capabilities have therefore to be integrated in the decision support systems. Disaster exercises provide a possibility for testing such components under realistic conditions.

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ASSESSMENT OF AVERTED FATAL CANCER CASES DUE TO LIMITATION OF RADIOCAESIUM INTAKE WITH MILK FOR POPULATION OF BELARUS AFTER THE CHERNOBYL ACCIDENT

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After the Chernobyl accident different protective measures for radiation protection of Belarusian population were carried out. As a result, annual reduction of exposure and prevention of significant doses for people of Belarus are registered.

Exposure doses for overwhelming majority of population during all phases of accidental period are conditioned, mainly by the levels of internal doses. For inhabitants of clean and low contaminated territories total doses are formed only by means of internal exposure from consumed foodstuffs, contaminated by radionuclides. For 1.3 million of inhabitants of contaminated territories (with density of radiocaesium contamination more than 37 kBq/sq.m internal doses are higher than external doses in most cases. Only for inhabitants of individual settlements with densities of radiocaesium contamination of territories more than 185 kBq/sq.m, external exposure doses play an important role in total dose formation.

In connection with this, the specific attention during all phases of accidental situation were given to measures, acting to the reduction of internal exposure doses. Internal exposure doses are forming mainly due to radiocaesium intake with ration. Contribution of inhalation component in the total doses of internal exposure is low.

The amount of radiocaesium intake depends on peculiarities of ration structure and caesium transfer by food chain. High specific weight of milk and milk products in daily ration and levels of its radiocaesium contamination are the reasons of main role of milk products in alimentary radioactive intake. As a result of investigation it is known, that for urban and rural population radiocaesium intake with milk can forms 20-24% and 44-54% of total intake with ration correspondingly.

So, establishment of permissible levels for radiocaesium contamination of milk and it revision in the hours of time allows to avert significant part of internal doses.

During intermediate and late phases of accidental situation different permissible levels for radiocaesium contamination of milk were carried out. Since 1986 till 1990 limit for radiocaesium content in milk was 370 Bq/l, since 1991 till 1992 - 185 Bq/l and since 1993 - 111 Bq/l.

Investigation of averted collective doses of internal exposure due to establishment of strict limits in 1990 and 1992 shows the following (table). Averted collective internal dose for urban inhabitants of most contaminated Gomel and Mogilev regions of Belarus forms near 2.2 thousands person-Sv and 1.8 thousands person-Sv correspondingly.

On basis of preliminary estimation it is possible to conclude, that averted collective dose of internal exposure for rural population are similar for those of urban people. But for correct estimation it is necessary to take into account the levels of consumption rate of private milk by rural population. Consideration of this fact can decrease levels of averted internal doses for rural population.

Estimation of averted risks of stochastic effects for urban inhabitants of Gomel and Mogilev regions, that were calculated on the basis of ICRP coefficients, shows that no less than 110 and 88 fatal cancer cases during life and up to 3 cases of genetic disorders among first two generations of descendants of exposed people correspondingly are prevented on a basis of averted doses of internal exposure.

Averted collective dose of internal exposure for urban population of Belarus allows to prevent the probability of occurrence no less than 800 fatal cancer cases and 20 cases of genetic disorders in first two generations.

Data obtained testify that conduction of estimated countermeasure allows to prevent significant part of collective dose of internal exposure and stochastic consequences of accidental exposure for population of Belarus.

Table.

Averted dose of internal exposure and prevented stochastic effects for urban population of Belarus due to establishment of strict limits for radiocaesium contamination of milk.

Region	Averted dose, person-Sv	Prevented fatal cancer cases during life	Prevented cases of genetic disorders in first two generations of descendants of exposed people
Gomel -urban population	2204.6	110.2	2.8
Mogilev -urban population	1758.2	87.9	2.3
Belarus -urban population	16684.0	843.2	21.4

ASSESSMENT OF THE THYROID PROTECTION EFFICIENCY FOR BELARUSIAN CHILDREN AFTER THE CHERNOBYL ACCIDENT

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During the acute phase of the Chernobyl accident different countermeasures directed on the thyroid gland protection were carried out. The main measures are evacuation and administration of stable iodine.

Evacuation of more than 12 thousands of Belarusian people from 30-km zone at May, 4-5, 1986 provided relative protection of thyroid glands. Besides that, prohibition for consumption of local milk and leaf vegetables were carried out. As far as this measure was established in a few days after the accident, effectiveness of it in reduction of radioiodine uptake by the thyroid gland was low.

The main countermeasure for thyroid protection of no-evacuated people was administration of stable iodine.

It is known, that taking stable iodine is a measure for reducing the uptake of inhaled and ingested radioiodine by the thyroid (1). For guarantee of complete thyroid gland protection stable iodine should be administrated before any intake of radioactive iodine. The uptake of I-131 by the thyroid decreases by 90% if stable iodine is administrated at the time of intake. The effectiveness of the measure decreases with delay, but the administration after the radioiodine intake provides 50-90% of protection depends on the time of taking stable iodine (2).

In case of situation after the Chernobyl accident the criterion of necessity for thyroid gland blocking by stable iodine were dose levels of intervention, expressed in projected adsorbed thyroid doses for first week after the accident: 300-2500 mGy (3,4). These levels are higher than recommended international intervention levels (5). But it is also suitable in case of conduction in time.

For protect of thyroid gland few millions of tablets of stable iodine were distributed among Belorusian people depends on the distance from the Chernobyl station. But there are some reasons to suppose that only 20% of distributed tablets were used for protection (6). In addition, distribution of tablets was begun some days after the accident.

So, it is possible to conclude, that countermeasures for thyroid gland protection of Belarusian people were conducted not in time, that was reason of low effectiveness of thyroid protection. In this situation population of some regions of Belarus and, first of all, children, received the significant thyroid doses. As a result, during years after the accident it was registered the increase of the thyroid cancer incidence rate for Belarusian children (7).

Among children till 14 years in year of accident who were exposed in Gomel region of Belarus 217 thyroid cancer cases were registered during 1986-1994. Administration of stable iodine during first hours after the accident allows to prevent approximately no less than 800 and no more than 2 thousands thyroid cancer cases during life among this group of children.

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APPLICATION OF AMMONIUMCYANOFERRATE IN REMEDIAL MEASURES IN AN AREA OF ELEVATED Cs-CONTAMINATION IN MILK

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INTRODUCTION

Due to their relative long half-life and the high transfer factor into foodstuff Cs-isotopes may cause a substantial contribution to the exposure of the population after a fallout from a reactor accident or nuclear weapons' detonations. The reduction of Cs-concentration in foodstuff, therefore, is one of the major goals in proper countermeasures after such a contamination. Ammonium-Ferric-Hexacyano-Ferrate (AFCF), commonly referred to as "Giese-salt", is known to result in a substantial reduction of the Cs-activity concentration in milk and meat when applied with the fodder to the animal. The theoretically achievable reduction in activity concentration under experimental conditions and under scientific surveillance is known and amounts to about 80 - 90 % (1,2,3). Such experiments have been performed up to now on only a few individual animals. The actual reduction, however, which is achievable under realistic conditions is not well known. This reduction depends on such factors as acceptance of application by farmers, knowledge and ability to apply it in the prescribed way, but also on the variability between different animals, etc.

To investigate this question under realistic terms incorporating a large number of animals was enabled by the fact that ammoniumcyanoferrate was applied in an area of elevated ^{137}Cs-activity concentrations in milk in a part of the province of Lower Austria in the first half of 1988 (4). In contrast to the ideal situation of a research farm under controlled and optimized circumstances, the application was performed by several hundred farmers not acquainted to the application before. These had been only trained by members of the local agricultural associations and colleges before application and applied the salt according to their ability and understanding.

Since cyanides are allegedly carcinogenic, prior to application the AFCF was tested for possible long-term effects and in particular for carcinogeneses. The results of these tests shall also be discussed in short.

METHOD

The application of the Giese-salt was performed in an area of about 1000 km^2 involving several thousand cows and some 100 farmers. The region was chosen because of generally higher activity concentrations in milk than elsewhere although samples above EC intervention levels were very rare at the time of intervention. The application lasted for 6 months ensuring that equilibrium conditions were reached (4). The application started only 1.5 years after the Chernobyl event. The reason for this delay compared to ordinary practices after a nuclear fallout was that in the initial phase activity levels in that region were not significantly higher than elsewhere. But, because of the partially semi-natural habitat the long-term decrease was slightly slower than elsewhere in the country. Furthermore, after having realized the higher activity levels in that region, it took some time to get results on the epigenetic and carcinogenic tests providing the basis for approval of the application by the authorities. The application was performed by feeding 2 - 3 g of AFCF per cow each day. It lasted from 1 February to 10 July 1988.

The measurements of the activity concentration in the milk produced were not performed in the milk of individual cows in order to ensure optimum results. Rather the activity concentrations were determined in the milk powder produced in the milk drying plant of that region. By taking milk powder samples each sample averaged about 10000 - 30000 l of milk (5) since the milk is collected in milk collecting tours imploying tanks of about 3000 - 10000 l the contents of which are combined in the drying plant in even larger tanks. Thus, by taking powder samples a good averaging over large feeding areas and a great number of several 1000 individual cows was achieved.

The milk powder samples were measured in 1 l Marinelli-beakers on HPGe-detectors of 20 - 30 % relative efficiency which had been calibrated by intercomparison tests and milk powder standards supplied by the Austrian Research Centre Seibersdorf. Thereby, all measurements, even if they had been from different milk drying plants, are comparable. With a measurement time of 1000 s a detection limit of 5 Bq/kg and an average error of 5 - 7 % with individual samples of typical activity concentrations was achieved.

TESTS ON HARMLESSNESS OF APPLICATION

Ammonium-Ferric-Hexacyano-Ferrate is known to have an extremely low resorption in the body of animals (6). Also practically no transfer into body liquids and particularly milk occurs. Thus no effects in foodstuff should be observed. Despite this low resorption particular attention was paid to possible epigenetic - carcinogenic effects of the substance. Therefore, on behalf of the Austrian Federal Ministry of Health mutagenity tests were performed by the Austrian Research Centre Seibersdorf, Toxicology Departement. They comprised the reverse mutation assay with salmonella typhimurium (AMES) and the micronucleus test (7,8). The micronucleus test was performed on 15 male and 15 female mice in combination with a positive and a negative control group at 3000 mg/kg body weight (7). In the AMES assay AFCF was tested on five different salmonella breeds in two separate tests per breed and at 5 different concentration levels of AFCF with positive and negative controls (8).

AFCF did not show any mutagenity according to AMES nor in the Micronucleus test on NMRI-mice. For health reasons there were thus no obstacles for an application as an additive to fodder in a contaminated area for a limited period of time under restricted conditions (application of 2 - 3 g at most per animal per day)

RESULTS AND DISCUSSION

The weekly average activity concentration in milk powder is given in figure 1 as sum of ^{134}Cs and ^{137}Cs for the period August 1987 (before application) to August 1988 (after application). Obviously, from August to November 1987 the activity concentration decreases only slowly. This is expected due to fixation of caesium in soil and penetration into deeper layers. The effective half-life of this process which was determined for this site in another study over 7 years, amounted to 634 d (9) for ^{137}Cs. This is equivalent to a biological half-life of 672 d. Taking into account that in figure 1 the activity concentration is given as sum of ^{134}Cs and ^{137}Cs, the effective half-life for the sum activity of 540 d is displayed in figure 1. The activity concentration obviously follows this half-life very well during the period August to November.

In December an increase is observed which is mainly caused by feeding of hay which was harvested in summer 1987. The activity concentration, therefore, shows an average value comparable to the hay concentration of that period. This value was also predicted in a study by the Horak et al. (7) in which a large number of grass and hay samples of that region had been measured in summer 1987 and by deploying a transfer factor of 0,004 $d.kg^{-1}$ an average activity concentration of 25.5 Bq/l in milk had been predicted. This activity concentration which should last as long as this hay is fed to the cows (May 1988) is also displayed in fig.1.

With the start of the feeding of AFCF in February the activity concentration in milk rapidly drops to approximately 15 Bq/l followed by a further slow decrease to values around 7 - 9 Bq/l in May and June. The average activity concentration in that period is about 30 % of the value resulting from feeding the hay alone (without application of Giese-salt). Thus, the application of the caesium-binder reduced the activity concentration by about 70 %.

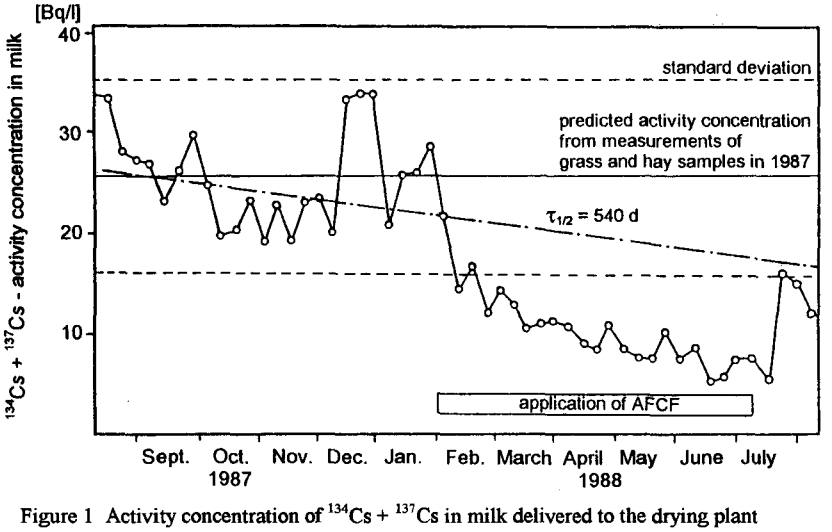

Figure 1 Activity concentration of ^{134}Cs + ^{137}Cs in milk delivered to the drying plant

At the end of the application period (10 July 1988), the activity concentration rises again to about 16 Bq/l. This value would be expected since without application of AFCF and after the winter hay feeding period the activity concentration in milk should be given by the broken-dotted line in fig.1 (resulting from the effective half-life of 540 d as described before). This is well demonstrated by the observed activity levels in figure 1 after mid-July.

Another proof of the reduction effect of activity concentration in milk is given in figure 2 which shows the activity concentration of $^{134}Cs + ^{137}Cs$ in milk in the four major drying plants in Austria in the period January to July 1988. These plants are situated in different provinces in Austria, separated by more than 80 km distance. During the very period all other drying plants show a rather constant value of the activity concentration with plant No.2 and 3 having average milk values of about 10 Bq/l and slowly decreasing to 8 Bq/l on average. Plant No.4 starts off with an average concentration of about 35 Bq/l and stays at this value till about May when the activity levels decrease to about 20 Bq/l. Only in plant No.1 where the AFCF had been applied to the cows, the decrease in activity concentration appears already in February with the beginning of AFCF feeding and stays at this level until June.

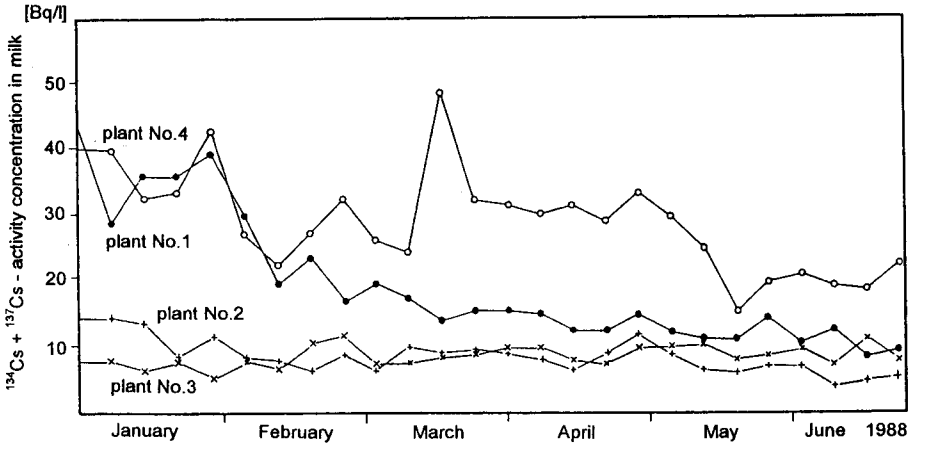

Figure 2 Activity concentration of $^{134}Cs + ^{137}Cs$ in milk in the four major drying plants

CONCLUSIONS

The application of Ammonium-ferric-cyano-ferrate (AFCF), also called "Giese-salt" with reference to its discoverer, may result in a substantial reduction of the activity concentration of ^{134}Cs and ^{137}Cs in milk, even if the drug is applied by farmers not familiar to its application before. Farmers who had been only quickly advised in its application and who had applied the drug for six months, achieved an average reduction in activity concentration by about 70 % during the period of application. This takes into account the variability of Cs-clearance in different individual animals as well as the varying ability of each farmer to cope with the problem.

The decrease in activity concentration is comparable to other efforts to reduce the activity concentration. A reduction by 30 - 80 % had been achieved after the Chernobyl accident in Austria for example by selection of low activity milk in the dairies. The application of AFCF is toxicologically safe and may be satisfactorily performed even by farmers not acquainted with ist application before. It, therefore, is a very valid tool in protecting the public in case of serious ^{137}Cs-contaminations after major nuclear accidents.

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DECONTAMINATION STRATEGIES IN CONTAMINATED SETTLEMENT.

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INTRODUCTION

Six years after the Chernobyl accident, decontamination actions had been completed in many places, the contamination could be considered as fixed, especially on urban surfaces and the social situation was felt to be stabilized. Under those conditions the efficiency of the « classical » decontamination techniques was under question, it was worthwhile to look at new specific techniques. Besides it was necessary to discuss the interest of new decontamination actions in settlements.
The European Union (EU) sponsored a project ECP 4 in order to look at the opportunities for further dose reduction actions in the contaminated territories of the three republics affected by the accident. The objective was to provide a local decision maker, faced with many alternatives for decontamination, with all the elements for determining what to do according to the various objectives he might pursue. The main results are presented here.

DOCUMENTING DECONTAMINATION TECHNIQUES

The project had a very broad objective. It surveyed usual techniques, techniques that had been designed within the framework of ECP4 (under assumption that they can be made effective at a full scale level), techniques that had been developed or examined by other Programmes, and the « do nothing » option. Exhaustivity was searched for. Actions dealing with all the sources and pathways were considered ; decontamination of urban environment (walls, roofs, yards, roads...), soil decontamination (arable soils and pastures) with chemical and physical approaches, decontamination of forests, protection and decontamination of machines and the decontamination of domestic and industrial food. Laboratory and field experiments have shown that it is still possible to decontaminate walls and roofs on which Caesium remained almost 10 years (for example using water under pressure), or fields in which the fertile layer is too thin for ploughing (e.g. turf harvester), and that modification of cheese making processes can achieve significant decontamination factors (e.g. the Phoenix cheese).
Effectiveness of decontamination has been estimated for about 60 techniques. Figures apply to Caesium, several years after an accident. Some typical results on decontamination factors (Df) are presented here (Table 1).

Table 1: Decontamination yields of Techniques applying to urban objects

Technique	Target	D_f	Constraints	Comments
Turning flagstones manually	Flagstones	6	No	-
Set of tools for dismantling	Houses	∞	No	Need to build a new house
Fire hosing	Roads	1.10	No	Water rinsing
Vacuum sweeping	Roads	1.4	No	Dust close to operator
Roof washer	Roof	2	No	Rot. brush, air compressor
Electric drill, steel wool or sand-paper	Roof (iron), wall (painted)	2-2.3	Possibly scaffolding	Grinding
High pressure turbo nozzle OM-22616	Roof, Wall, Asphalt & Concrete surface	1.3-2.2, 1.7-2.2	No	High pressure water hosing 120 bar
Detached polymer paste	Surfaces (smooth)	4-30	T > +5°C	Transports. Manual work
Sandblasting (dry or wet)	Wall	4 (dry) 5 (wet)	Scaffolding preferable	High-pressure with sand, Whole-body protect/air supply
Polymer coatings	Walls (not wooden)	4-5	T=20-30°C	Humidity <80%
Manual electric cutting machine	Wooden wall	5	Residual nail remove	Upper layer mechanically removed (dust)

Front loader / Bulldozer	Soil	28 / 10-100	No	Scraping top soil (10-30cm) Removes fertile soil layer
Shovel	Garden soil	6	Virgin soil	Digging to about 30 cm depth
Turf harvester (small)	Undisturb. grass. soils,	3-20	No or few stones	Removes the 3-5 cm top soil.
Ordinary plough and tractor	Arable soils	9-12(ext.)	Virgin land	Ploughing to 25-45 cm depth
Deep ploughing	Arable soils	2-4(crop)	Virgin land	Plough soil layer (25-35cm)
Skim-and-burial plough	Arable soils	10-20	Virgin land	5 cm topsoil buried at 45 cm
Liming (special trucks for spreading)	Acid arabic land	1.3-3	Soil pH = 4.5-5.5	Requires K addition, Persistent effect during 4-5 y.
Addition of potassium	Arable lands	1.3-3	No	K addition needed
Organic amendment of soil	Arable soils	1.3	-	Yield and quantity increase
Radical improvement of Pasture (draining, cleaning, disking 3 times...)	Pastures	4-16(peat) 4-9 (podzol)	No	Yield increase
Liming and fertilising	Forest pastures	1.5	Manual	Poor soils enriched by Ca, K
Ferrasin bol or Prussian blue	Cow	2-3	3 bolus/3 m.	Where Cs level > 1000 $Bq.l^{-1}$
Clean fodder to animals before slaughter	Cow	2-3 on meat	2 m. before slaughter	Organisation of special animal feed before slaughter
Phytodecontamination	Soils (mixed)	1.1-1.3.y^{-1}	7 procedures	
Ferrasin filters for milk	Milk	ca. 10	Private farm	If milk contamination> 400 $Bq.l^{-1}$

Other techniques concerning the rehabilitation of forests were examined, as well as techniques for food processing (e.g. alteration of the traditional Tvorog cheese production with a Df of 5).
Techniques have been also documented according the needs of optimisation. Components of the costs were depicted in physical terms: direct manpower and overheads, skill requirements for workers, and needs in education for the public, transportation, consumables, loss of value for products, generation of wastes (solid and liquid volumes, activity and toxicity). Prices were necessary in order to assess costs. A short synthesis was made in order to know the prices of manpower (from 70 to 100 ECUs per month), consumables (e.g. 0.15 to 0.3 ECU per liter for gasoline) and products (e.g. 0.25 to 0.4 ECU per liter for milk). Prices are indeed fluctuant at the local scale but this task was necessary to provide default values and standard values when comparing case studies.

IMPLEMENTATION OF CASE STUDIES

Case studies have been implemented in 4 settlements in order to illustrate a methodology for decision, and to look at cost effectiveness of countermeasures under different conditions (Table 2).

Table 2 : Some features of the case studies.

Cases	Reference Soil contamination ($Ci.km^{-2}$)	Present population	% of pre-accident population	Dominant pathway	Previous actions
Zaborie	66	180	20%	External	Extensive
Kirov	30	500	40%	External	Extensive
Savichi	10	160	20%	External	Limited
Millyachi	5	3200	100%	Internal	Numerous

When dealing with the settlements, no cases were found in which decontamination was desired in order to meet one unique target. The global idea is a desire for a return to a normal life style, by suppressions of interdictions in the formally "evacuated" areas (Savichi, Zaborie) or by alleviation of restrictions in villages in the other areas (Kirov, Millyachi). The need to be less dependent on the restrictions to life style is not a quantified criterion but it is a clearly expressed goal.
Individual doses are a matter of concern that was put forward in the four case studies, but not with reference to the same criteria. Average population dose is a criterion in Zaborie, to be compared with 5 mSv per year. Critical groups of workers have been identified in Zaborie, Kirov, Millyachi, whose doses are still to be compared with 5 mSv (e.g. forestworkers, cowboys,...). Reference to 1 mSv for the average population is also quoted (e.g. Belarus "Passports" on settlements).
It has been said that population stayed or came back in evacuated territories. Direct observations also shown that prohibited land was used either by the people from the settlement or by neighbours.
No objectives were assessed in terms of collective dose. Nevertheless, the gain in collective dose was considered as a good criterion to assess the efficiency of a countermeasure. Cost effectiveness was measured by the « cost of averted Man Sievert ». It was admitted that the most cost effective measure should be applied first in the strategy.
In Zaborie, application of countermeasures was limited by the fact that extensive decontamination took place so that options like ploughing were pointless. However, calculated doses being only slightly higher than 5 mSv (while measured doses are lower), many countermeasures would allow to go below this figure ; cleaning yards, education on mushroom consumption, but combination of agricultural countermeasures. Looking at cost efficiency, actions dealing with yards (Front loader or Bulldozer) or educational programmes yield to a cost of averted Man Sievert below 2000 ECUs, and they would be used first in applying a strategy. The most effective action was the cleaning of the yards surrounding the houses, provided that a local solution for waste disposal is accepted, as was done in the neighbouring settlement of Yalovka. Should generic costs be applied it would be 50 times less effective. Education on mushroom was priced on the basis of one visit per year to each family. Liming, in spite of a poor global impact, has a good cost-efficiency ratio and deserves to be included in the strategy. Both would go along better with a rehabilitation programme. Other options still have interest. Some options yield very high costs for averted Man Sievert, others are limited in their scale of application, because areas fitted for them are limited (e.g. Turf harvester).
Situation in Savichi is such that the effectiveness of countermeasures is usually lower, simply because the contamination levels are lower. Most countermeasures yielded cost of avoided ManSievert higher than 8000 ECUs. Turf harvester and ploughing achieved a cost lower than 1000 ECUs (actually less than 200 ECUs), because there was still room for such countermeasures, so that improving meadows and pastures can be proposed. Cleaning yards remains the most cost effective of the measures dealing with housing. In parallel with the cost benefit analysis, classification of Savichi as a prohibited settlement was reconsidered.
In Millyachi, the importance of internal dose and the high transfer coefficients increase the respective efficiency of agricultural countermeasures. Education for the mushroom consumption remains highly efficient. Many agricultural measures have cost efficiency around 1000 ECU per averted ManSievert, but cleaning yards still remains effective.
Almost in all cases there are actions with a reasonably low cost for averted dose. Should ManSievert be valuated to 5000 ECUs or even 2000, there are indications for application of countermeasures. Besides, it remains true that actions on non disturbed land are among the most efficient and that they are limited in the present case studies.

CONCLUSION

The necessity to undertake new decontamination actions or other dose reduction actions almost one decade after the Chernobyl accident raised serious doubts when the programme was launched. Today, decontamination of all the contaminated territories is still out of reach, but the studies that were undertaken have shown that there are clear indications for action in practical cases. It is believed that this feeling is shared by the local authorities that were involved in the case studies.

Acknowledgements

This study is published under the title Strategies of Decontamination, along with the other E.U Collaboration Projects. The participating teams were IPSN (France) , CIEMAT Spain), RISØ (Denmark), RPII (Ireland), AEA Technology (U.K.), AUA (Greece), RECOM (Russia), IBP (Russia), BIIM (Russia) BIRH(Russia), UICBGE (Ukraine) , IGMO (Ukraine), UIAR (Ukraine), UIG (Ukraine), MinChernobyl (Ukraine), INR (Ukraine), IPEP (Belarus) , IREP (Belarus), BIAR (Belarus),CSCB (Belarus). BSU (Belarus).

MANAGEMENT OF CONTAMINATED AGRICULTURAL ENVIRONMENTS FOLLOWING A MAJOR NUCLEAR ACCIDENT. AN OVERVIEW ON POSSIBLE SHORT-TERM AND LONG-TERM COUNTERMEASURES

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INTRODUCTION

Accidental release of radioelements from nuclear facilities may give rise to severe consequences for the agricultural environment. The deleterious effects will, in addition to the nuclides released and their deposition density, depend on the sensitivity of the agricultural land with respect to climate and soil types and hence the type of agriculture practiced. Another crucial point for the consequences, at least during the first year after the accident, is the point of time in the year when the fallout occurs. In the winter period the deposition mainly will take place on the bare or snow-covered ground sur - face, while a fallout during the vegetation period to a varying extent will be intercepted and retained by growing crops. Depending on the stage of development of the crop and the translocation rate of the single nuclides food and feed can be highly contaminated (the concept of "seasonality", *1*).

CRITICAL NUCLIDES AND EXPOSURE PATHWAYS. DOSE CALCULATION MODEL

With respect to the volatility of nuclides of biological importance, which can be released at a large reactor accident, their abundance will decrease in the following order: iodine isotopes > caesium isotopes > strontium isotopes > transuranic elements. While the radioiodine isotopes ^{132}I and ^{131}I in practice have decayed after one and eight weeks respectively, the effects of radiocaesium (^{134}Cs and ^{137}Cs) and of ^{90}Sr will last for years or decades. Radioiodine and radiocaesium exert both external and internal exposure, while ^{90}Sr (and the relatively short-lived ^{89}Sr) and transuranic elements mainly will be of importance for the internal exposure.

Fallout nuclides in the agricultural environment thus will give rise to external exposure of farmers and others living in the contaminated area and to internal exposure of people eating food with origin from the area. The following general dose calculation model may be used for assessment of the dose contribution from a certain exposure pathway *(2)*:

$$H = C * U * D * P$$

where H means the dose rate to a certain organ or individual, C means the deposition density of a certain nuclide (in *Bq per* m^2) or the concentration of this nuclide in a certain crop product or food stuff (in *Bq per kg*) and U is the utilization rate of this food item or, in the case of external exposure the time of sojourn in a certain environment. D is the dose conversion factor for the nuclide in question (*Sv per Bq*) and P is a dose reduction factor taking into account that the external radiation partly is shielded by soil material, building structures vehicles etc. (as compared with the deposition on a smooth ground surface). In the case of internal exposure P may be considered as a dose reduction factor due to various mitigating actions, which can be undertaken in order to reduce the transfer of the nuclide to a certain crop product or food stuff (as compared with the case where no counter measures have been taken). For calculation of the total dose commitment the contributions have to be summarized for various organs in the body and for all nuclides and exposure pathways integrated over the time and taking a number of natural dose-reducing processes into consideration.

Immediately after a release from a nuclear accident external and internal exposure from the passing plume, and external exposure from nuclides deposited on the ground surface, buildings, vegetation etc. potentially will give doses high enough to cause deterministic effects. After some days decay of the short-lived nuclides the exposure will give rise to stochastic effects only. The external exposure from the deposited nuclides (mainly ^{131}I and $^{134+137}Cs$) will be a dominating exposure pathway, independent of the season when the deposition occurs.

Direct contamination of growing crops will give rise to a number of internal exposure pathways as the nuclides are transferred through the food chains to man. Such a critical exposure pathway is the transfer of radioiodine, radiocaesium and radiostrontium through the chain: pasture grass - grazing cows - milk - man. The effects of a direct contamination will last as long as the crop products are available for feed and food, usually up to one year after the deposition, unless the crop products are condemned.

During the following years the internal exposure will be due to the soil-plant transfer of the long-lived nuclides (indirect contamination). This pathway will usually give much lower nuclide concentrations than after a direct contamination of the crops.

BASES AND STRATEGIES FOR DECISIONS ON COUNTERMEASURES

A number of dispersion models have been calculated for possible reactor accidents, taking various source terms and weather parameters etc. into consideration. The Chernobyl accident in 1986 showed, however, that the dispersion pattern of a radioactive plume can be much more complicated, with respect to variation in deposition density and range of the plume, than can be foreseen in more idealistic models. This fact, in addition to the variation in the radioecological sensitivity with respect to soils and crops, also means that the need for countermeasures in the agricultural environment will vary considerably even between adjacent single farms.

Preparedness planning for the agricultural sector in the case of large reactor accidents includes a number of steps. Besides the basic dose criteria, knowledge of the agricultural environment with respect to soils and crops, population and livestock density etc. will be of importance for predictions of the consequences. Transfer factors for the various nuclides and steps in the food chain, obtained in experimental or environmental studies, will also be a necessary tool. Early warning systems, locally for the environment around single nuclear facilities and on a national or international basis, will be valuable in order to prevent exposure leading to deterministic health effects, but also in such a way that countermeasures can be taken in advance before the passage of the plume. A well established organization for monitoring, sampling and nuclide specific activity measurements will be of utmost importance as a basis for fast decisions on suitable countermeasures. Open and continuing information from the authorities to farmers and others living in the contaminated area will also be important, as any for psychological reasons.

As soon as data on deposition density and nuclide concentrations are available these should be used in prediction models for calculation of the expected nuclide concentrations in crop products, milk, meat etc., as to estimate whether these will be acceptable for human consumption or should be discarded. A continuos following up of the monitoring will be necessary during the vegetation period. All countermeasures should be based on the dose criteria given by ICRP and the cost of an action should be reasonable in relation to the value of the saved dose commitment (*3*).

COUNTERMEASURES FOR THE AGRICULTURAL ENVIRONMENT

A number of possible countermeasures for various situations after an accidental release are discussed and evaluated below. The fallout is assumed to take place just before the first harvest of grass for hay or silage.

A. The acute situation (the first week after the accident).

If the early warning time permits and the trajectory of the plume can be foreseen *evacuation* of persons not involved in the livestock management should be undertaken to prevent exposure from the passing plume and deposited activity. Grazing cows should be installed. Harvest of uncontaminated forage, e.g. by using modern techniques for enwrapped silage, would be an effective countermeasure in such a situation. For persons staying on the farm *sheltering* or *indoor sojourn* during and after the passage of the plume should be recommended as far as possible. *Intake of stable iodine* before or up to 30 min. after the passage of the plume would be an effective mean as to reduce the thyroid dose. After the passage of the plume all field operations should be postponed and the use of food and feed should be restricted until the fallout has been monitored with respect to external exposure and nuclide contents in the crops. (*3, 4*).

B. The intermediate situation (from one week up to one year after the accident)

The restrictions for field operations and for use of crop products should remain until the external exposure has become acceptable and the deposition density and the nuclide concentrations in animal and crop products can be estimated. If the concentrations of critical nuclides in the crop products will not be judged as acceptable as food or feed, *removal and discarding of growing crops* may be a way to reduce the deposition density of the land. A dense crop may initially intercept up to 50% or more of a deposition, why fairly concentrated wastes are obtained. After removal of such a highly con - taminated grass crop the new growth, however, may be acceptable with respect to the nuclide concentration (compare C below). *Removal of contaminated soil* etc. around the farm buildings may be effective for reduction of the external exposure.

C. The long-term situation (one to ten years after the accident)

During this period of time the internal exposure mainly will be due to the soil- plant transfer of radiocaesium and ^{90}Sr. The transfer factors for ^{90}Sr have been shown to be up to one order of magnitude higher than for radiocaesium and for both elements the transfer usually will decrease in the order peat soils > sandy soils > clay soils. The possible strategies for reducing the long-term effects refer to the following categories (*4*): *a. removal of the contaminated soil surface layer; b. deep placement of the contaminated layer; c. application of agrochemicals to the soil* (or application of feed additives) and *d. changing the line of production* (*4* and refs. cited here)

Scraping of the contaminated surface soil for removal and deposition can be a rather efficient method for decontaminating agricultural land, although the costs will be high. For handling the large amounts of soil (350-600 tons per ha) special equipment is needed and the work has to be carried out under ideal conditions. Therefore this method will hardly be realistic for large scale operations. On grassland the use of flail-type forage choppers with equipment for removal of the crop may be a method for the removal of a contaminated crop and the upper part of the contaminated sod

By ploughing the contaminated land the fallout nuclides will be distributed in a 20-30 cm deep soil layer and thus be less available to the crops as compared with a distribution in the uppermost few cm. Ploughing up contaminated grassland, wherever this will be possible, followed by the establish- ment of a new grassland will almost always reduce the nuclide transfer to a factor 5-10 or even more. By deep ploughing or by use of specially designed double layer ploughs a deeper placement of the contaminants can be obtained than after conventional ploughing. However, such ploughs are rare and have a low capacity. All ploughing should be performed under ideal conditions as to be a successful mitigating action.

Application of potassium fertilizers usually will depress the transfer of radiocaesium (by a factor of 2-3 or more at annual applications of 100-200 kg K per ha) and liming acid soils will reduce the transfer of ^{90}Sr and often also of radiocaesium. Generally a well-balanced fertilization, aiming at a positive potassium balance and an optimal pH of the soil, will give a sustainable low transfer of most radioelements.

By changing the line of production of the farm, e.g. from growing crops where vegetative parts are utilized (grassland crops) to cereal or seed crops, a reduced nuclide transfer into the food chain will be obtained. Usually a combination of the various measures will be necessary for the best outcome of the problems.

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PAPER TITLE: CAN AGRICULTURAL COUNTERMEASURES LEAD TO AN INCREASE IN POPULATION DOSE ?

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MAJOR SCIENTIFIC TOPIC NUMBER: 4.5 ... (see page 7) ; 8.5

Practical application of Control Level (CL) distinguishing the production as "pure" (contamination level below CL) and "dirty" (above CL) requires a special analysis of strategies in the field of countermeasures (CMs). It has been shown, that implementation of CM under conditions of inhomogeniety of radioactive contamination (of areas, products) can give non-trivial results, e.g., an increase in collective dose due to differential uses of pure and dirty produce. Based on the analysis of contamination structure before and after CM, it has been shown that, whereas radionuclide removal with dirty produce is decreased, radionuclide content in pure production as a result of CM can be increased, even in unchanging gross production level. Besides, average contamination of pure produce after CM also may be increased. It has been, that CMs such as radical amelioration of pastures and ferrocyne treatment can result in 20-50 fold increase of radionuclide content in pure milk. Therefore, in spite of a general diminishing of radionuclide content in products obtained, population dose can be increased after CM as a result of consumption of pure and processed dirty production. CMs in agriculture give increased percentage of pure production. In this case, it is only based on complex analysis of radiological and economical parameters that effectiveness of introducing the CM can be estimated (e.g., using cost-benefit analysis; the cost-effectiveness analysis is not applicable in this case). A similar approach used for estimation of changes in the structure of contamination of products in the course of time (due to radioactive decay and self-cleaning) also results in a conclusion of possible increase of collective dose during the subsequent years.

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PAPER TITLE Chronic irradiation from incorporated radiocaesium could stimulate its uptake by plants

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ABSTRACT (See instructions overleaf)

We have investigated the influence of incorporated by roots radiocaesium on its uptake processes. 3-day pea seedlings were growing for week on the radiocaesium chloride solutions with specific activity from 0.2 to 519 kBq/l. Stable caesium chloride was added in these solutions in concentration of 1 μM/l to achive the equal chemical conditions for various variants. The all amount of radiocaesium from solutions was absorbed by plants over the week but its distribution within the roots was different for various accummulated doses. Apex radioactivity was equal to 14.9 % from radioactivity of whole root for dose of 0.004 Gy and 1.9 % for 2.475 Gy. That is in the later case roots faster absorbed caesium and redistributed it from root to stem under influence of chronic irradiation. There was an another evidence of it. The root apeces after previous chronical irradiation have accumulated radiocaesium in 4 times greater during 2 hours than control ones. So we could supposed greater level of radiocaesium accumulation by plants in conditions of chronic irradiation on the radionuclide contaminated places. This is especially important for developing of the efficient countermeasures for radiation protection in the environment.

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Mini-Presentation

PAPER TITLE Optimal systems of countermeasures and universal algorythm of decontamination of radionuclide's contaminated soils

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MAJOR SCIENTIFIC TOPIC NUMBER 4.5 (see page 7)

ABSTRACT (See instructions overleaf)

"OPTIMAL SYSTEMS OF COUNTERMEASURES S AND UNIVERSAL ALGORYTHM OF DECONTAMINATION OF RADIONUCLIDE,S CONTAMINATED SOILS" Y.Kutlakhmedov,A.Jouve,N.Zezina,A.Micheev (ICBGI NAS of Ukraine,IPSN CEA,France)(Abstract)

1.We provided of comparative analysis of efficacy methods and means decontamination of soil which were contaminated of radionuclide after accident on ChNPP.It was calculated Cd(coefficient of decontamination)of different countermeasures on criteria of decreasing individual and/or collective doses for population.

2.All possible means and methods decontamination can be divide on the two groups:the first - for ploughed soils(after accident);the second - for unploughed soils and territories.

3.It was created universal algorythm for election of means and methods decontamination of soil for different conditions.

The optimal choice of means and methods of decontamination depends from next circumstances: a) Ploughed or unplouged the soils; b)The characters of distribution of radionuclides on profile of soils ; c) The aims of decontamination (agricultural usage of territories,usage of territories for recreation ,The decreasing of individual and/or collective doses for population); d) The relation of benefit - cost for concrete countermeasures.

USE OF SAPROPELL SEDIMENTS AS A COUNTERMEASURE FOR CONTAMINATED SOILS

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ABSTRACT

The objective of this study is the characterization of a set of sediments in terms of radiocaesium specific sorption properties and the development of laboratory tests in order to provide reliable tools for predicting the amendment effectiveness of sapropell sediments in radiocontaminated freshwater sediments. Results for radiocaesium and radiostrontium partitioning between freshwater and sapropell sediments and values of adsorption and desorption distribution coefficient (K_d) for these sediments have been obtained, using two experimental protocols. Results show that: for radiocaesium, significant fractions are irreversibly retained in the freshwater sediments and not involved in the partition process between the two systems; for radiostrontium, the systems are completely reversible. It can be concluded that in the area of terrestrial radioecology the efficiency of the use of sappropell sediments as soil amendment will be confined: for radiocaesium, to soils of very low radiocaesium interception potential; for radiostrontium, for soils of very low cation exchange capacity.

INTRODUCTION

After the Chernobyl accident and the contamination of large areas with radiocaesium and radiostrontium a broad of countermeasures have been tried in order to put a brake on the soil-plant transfer of these radionuclides. In the recent past, such measures were most frequently focussed on liming and mineral fertilizing combinations mainly to radiocaesium. It has been found that the effect of liming and or potassium fertilization can reduce the radiocaesium level in the crop by a factor of 3 to 4 (1). It is generally accepted that these measures operate at the plant level and do not improve the condition of the soil. A number of additional countermeasures which do operate at a soil chemical level was tested in the field. Such measures are based on the addition of a range of adsorbent materials (zeolites and others) to the soils. Sapropell sediments which are characterized by very specific sorption properties for radiocaesium and radiostrontium, have been used in Chernobyl area as a countermeasure for radiocontaminated soils (2). Sapropell sediments are originate from deposits at the bottom of natural lakes in the CIS countries, particulary in Belarus and Ukraine. They are composed of indigested vegetal and animal residues decomposed under anaerobic conditions. The main purpose of this study is the characterization of a set of sapropell sediments in terms of radiocaesium specific sorption properties and the development of chemical tests in freshwater sediments in order to provide reliable tools for predicting the effectiveness of sapropell sediments in soil amendments. Laboratory experiments have been performed with freshwater sediments and a sapropell sediment (used as adsorbent because its high selectivity for caesium ions). Since the mechanisms of radiostrontium and radiocaesium sorption in freshwater sediments are identical with those in soils the conclusions and experimental protocols of this study are also applicable in the area of terrestrial radioecology.

1. PHYSICO-CHEMICAL CHARACTERIZATION OF SEDIMENTS

1.1. Materials and methods

A set of five sapropell sediment samples from Turichanskoje (TL1, TL2), Orechovskoje (OL1, OL2) and Snovskoje Lakes (SL1) (Belarus) and three freshwater sediments T2 (Tejo river, Portugal), A (Tejo estuary, Portugal) and KR-4 (Kiev reservoir, Ukraine) have been characterized in terms of cation exchange capacity (CEC), using the silver thiourea procedure (3), organic matter (OM) content (weight loss at 750°C), pH values (solid/liquid = 1/10 and 10^{-2} M KCl) and specific radiocaesium interception potentials, i.e. $[K_d m_K]$ and $[K_d m_{NH4}]$ values (4).

1.2. Results and discussion

Table 1 summarizes all the sediment parameters determined. It is seen that, CEC values cover a range of 7.5 to 45.0 meq.100g^{-1}. Organic matter contents cover a range of about 10 to 20%. Normalizing CEC values with respect to O.M. content leads to values in the range of 0.8 to 2.7 meq.g^{-1} showing that exchange capacity is to a major associated with organic matter. The $[K_d m_K]$, $[K_d m_{NH4}]$ and ($K_c(NH_4/K)$) values cover a range typical for soils and sediments (4) although the values obtained for samples OL1, OL2 and SL1 are quite low and similar to what is found for podzols and peat soils (5).

Sediments	pH	O.M. (%)	CEC meq.100g^{-1}	[$K_d m_K$] meq.g^{-1}	[$K_d m_{NH4}$] meq.g^{-1}	$K_c(NH_4/K)$
TL1	7.8	15.6	13.5	1.05	0.16	6.5
TL2	7.9	17.0	13.8	1.13	0.16	7.1
OL1	5.6	14.3	23.0	0.28	0.08	3.7
OL2	6.2	9.1	16.5	0.29	0.06	4.7
SL1	7.7	18.5	25.6	0.23	0.04	5.8
T2	5.9	5.1	7.5	0.60	0.18	3.2
A	7.6	13.4	27.9	4.55	0.82	5.6
KR-4	6.0	16.5	45.0	3.90	-	-

Table 1- Some relevant parameters for sapropell and freshwater sediments.

2. PARTITIONING OF ^{137}Cs and ^{85}Sr BETWEEN FRESHWATER AND SAPROPELL SEDIMENTS

2.1. Materials and methods

Partitioning of ^{137}Cs between one sapropell sediment (TL1) (used as adsorbent) and the freshwater sediments T2, A and KR-4 has been studied according to the following procedure:

Sediment samples (1g, enclosed in a dialyse membrane with 5ml 0.005N $CaCl_2$) and sapropell sediment (1g, also in a membrane with the same solution) were equilibrated with 200ml of a mixed 0.005N $CaCl_2$ and 10^{-4} N KCl solution for 4 days (one-time equilibration). This procedure leaves both systems essentially in the homoionic bivalent ion form. After equilibration, two protocols were followed:

2.1.1. Protocol (a)

The sapropell sediment sample (TL1) was set aside (to be used in a desorption test) and the freshwater sediment was contaminated with ^{137}Cs by equilibration with 50ml of a mixed 0.005N $CaCl_2$ and 10^{-4} N KCl solution labelled with ^{137}Cs (2 days, end-over-end shaking). The values of the distribution coefficient K_d were obtained by monitoring the activities of equilibrium solution and the sediments. The sapropell sediment (in the membrane) was then reintroduced and the systems were equilibrated (end-over-end, shaking). Countings of solution, freshwater and sapropell sediments, were made after an equilibrium time of 3 and 10 days. With these measurements it is possible to obtain the partitioning of ^{137}Cs between freshwater and sapropell sediments, the K_d (adsorption) for freshwater sediment and sapropell sediment (in the desorption process from the freshwater sediment), and the K_d (desorption) for the freshwater sediment.

2.1.2. Protocol (b)

This protocol is exactly identical to (a), except for the fact that the freshwater sediment sample was first removed and the ^{137}Cs was first introduced into the sapropell sediment. This procedure generates K_d (adsorption) for the sapropell sediment and freshwater sediment (in the desorption process from the sapropell sediment) and K_d (desorption) for the sapropell sediment. Protocols (a) and (b) differ merely in the sequence of the addition of reagents, and the entire procedure is a common final state experiment.

The same common final state procedure was used to ^{85}Sr, except that the composition of the liquid phase used for preequilibration of freshwater and sapropell sediments was only 0.005N $CaCl_2$.

2. 2. Results and discussion

Table 2 summarizes all data for 3 and 10 days of equilibration times, in terms of the ratio of ^{137}Cs contents and adsorption distribution coefficients for freshwater and sapropell sediments, for protocol (a) and (b). It is seen that the ratio of ^{137}Cs levels on sapropell (TL1) and freshwater sediments differ significantly depending on the fact if the sediment or sapropell are first labelled (protocol (a) or (b)), showing an irreversibility pattern of radiocaesium in the sediments. It appears furthermore that the ability of sapropell to desorb radiocaesium from freshwater sediments is quite limited, due to the very low interception potential to radiocaesium, and decreases with increasing [$K_d m_K$] values of the sediments (Table 1). Applying the argument developed for calculating the levels of radiocaesium fixation by sediments using clinoptilolite as adsorbent (6), ^{137}Cs levels of about 50-65% (T2), 45-65% (A) and 45-70% (KR-4) were obtained, for equilibration times of 3 and 10 days respectively.

Table 3 shows the results obtained for ^{85}Sr using protocols (a) and (b) and for equilibration times of 3 and 9 days. It is apparent that the partitioning of ^{85}Sr between the sapropell and freshwater sediments is identical in protocols (a) and (b), i.e. the system is completely reversible. Moreover the ^{85}Sr in TL1/^{85}Sr in sed. ratios are in excellent good agreement with predictions based on the sorption distribution coefficient values for the systems. The statistical averages of distribution coefficient ratio (sapropell/sediment) are: 1.2 ± 0.1 (T2), 0.46 ± 0.02 (A)

and 0.34 ± 0.02 (KR-4). These values confirm the reversible nature of the process. Evidently the use of sapropell to reduce radiostrontium availability in sediments, based on ion exchange selectivity, should be restricted to sediments of very low cation exchange capacity.

	Protocol (a)		Protocol (b)	
Sediments	^{137}Cs in TL1/ ^{137}Cs in sed.	Kd sed. (ads.) $ml.g^{-1}$	^{137}Cs in TL1/ ^{137}Cs in sed.	Kd TL1 (ads.) $ml.g^{-1}$
		three days		
T2	0.26 ± 0.01	6920 ± 1350	3.1 ± 0.1	9497 ± 188
A	0.10	12950 ± 927	1.5 ± 0.1	5131 ± 432
KR-4	0.09 ± 0.01	19420 ± 1219	2.1 ± 0.2	8357 ± 550
		ten days		
T2	0.43 ± 0.01	6920 ± 1350	1.2 ± 0.1	9497 ± 188
A	0.19	12950 ± 927	0.47	5131 ± 432
KR-4	0.21 ± 0.02	19420 ± 1219	0.54 ± 0.03	8357 ± 550

Table 2 - Radiocaesium partitioning between freshwater and sapropell (TL1) sediments for equilibration times of 3 and 10 days.

	Protocol (a)		Protocol (b)	
Sediments	^{85}Sr in TL1/ ^{85}Sr in sed.	Kd sed. (ads.) $ml.g^{-1}$	^{85}Sr in TL1/ ^{85}Sr in sed.	Kd TL1 (ads.) $ml.g^{-1}$
		three days		
T2	1.10 ± 0.02	21.1 ± 0.7	1.17 ± 0.03	24.3 ± 0.3
A	0.40 ± 0.01	51.5 ± 0.7	0.43	23.6 ± 0.4
KR-4	0.30 ± 0.01	64.0 ± 0.3	0.30 ± 0.01	22.1 ± 0.4
		nine days		
T2	-	21.1 ± 0.7	1.16 ± 0.02	24.3 ± 0.3
A	0.40 ± 0.01	51.5 ± 0.7	0.41	23.6 ± 0.4
KR-4	0.29	64.0 ± 0.3	0.30 ± 0.01	22.1 ± 0.4

Table 3 - Radiostrontium partitioning between freshwater and sapropell (TL1) sediments for equilibration times of 3 and 9 days.

CONCLUSIONS

It can be concluded that for radiocaesium, significant fractions are irreversibly retained in the sediments and, accordingly, desorption yields are lower. In the case of radiostrontium, desorption yields from sediments are consistent with complete reversibility. In the area of terrestrial radioecology "practical" results of the use of sapropell sediments as soil amendments can be expected: for radiostrontium, only for soils of very low cation exchange capacity and by using high capacity and high Sr-selectivity adsorbents; for radiocaesium, successful application of such amendments will be limited to soils of very low radiocaesium interception potential.

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ENVIRONMENTAL REMEDIATION OF SOILS WITH HETEROGENOUS RADIOACTIVE CONTAMINATION IN CANADA CHARACTERIZATION AND VERIFICATION METHODS

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INTRODUCTION

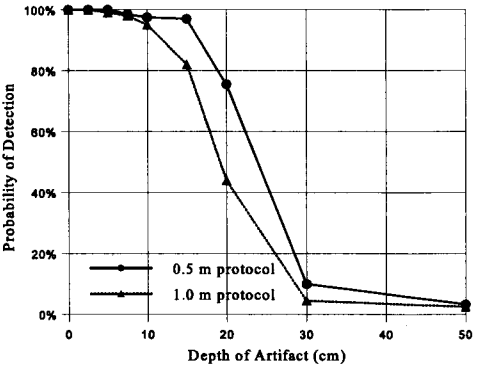
The distribution of contaminants in soil varies from site to site depending on the transport processes that have occurred. Natural transport processes tend to produce a continuous or homogeneous distribution. Human activities, such as earth moving, often result in a discontinuous or heterogeneous distribution presenting special problems for characterization and cleanup. The volume of suspect soil can be orders of magnitude larger than the volume of the discrete particles and pockets of contaminated soil that are present. It is desirable to characterize soil with discontinuous contamination in sufficient detail to allow cleanup of only the contaminated material and to determine with a reasonable certainty, that the material remaining following the cleanup does not require any further action.

In Canada, the Low-Level Radioactive Waste Management Office (LLRWMO) manages historic radioactive wastes that are a federal responsibility. The LLRWMO has developed and refined methods to characterize and verify the success of cleanups of soil with heterogenous uranium mineral and radium contamination. A combination of surface and sub-surface measurements are collected and statistically analyzed to determine the cleanup limits. The LLRWMO has been successful in cleaning up industrial and residential properties affected by heterogeneous contamination. Tens of thousands of cubic metres of uncontaminated soil, from 45 hectares of industrial land in Fort McMurray, Alberta and hundreds of private residential properties in Scarborough, Ontario have been freed from further actions or restrictions. This paper presents the methods the LLRWMO has developed and used for characterizing soil with heterogenous uranium ore and radium contamination, and for verifying the success of environmental cleanups. Examples from LLRWMO projects are described.

METHODOLOGIES

Surface readings are used as much as possible because of the economy of data gathering. The primary method used for surface characterization is the Large Area Gamma Survey System (LAGS) system that has been developed by the LLRWMO (1,2). Gamma radiation data are collected continuously along specified lines (generally one-half meter or one meter lines) of pre-established blocks and read along with location data into a computer. At the completion of each survey block, the collected data are analyzed and the locations of discrete anomalies in the data are identified for followup investigation. This is accomplished by a combination of manual detailed surveys and computer analysis. The computer analysis, called discriminant analysis, is a statistical technique whereby an empirical relationship between the quantitative variables and the classification of an object are established from a calibration set (2). Several different statistical tests are performed on the data, each of which results in a value indicating the probability of the presence of a discrete radioactive source. Several methods of identifying larger areas of possible radioactive contamination have been developed, each relying on the interpretation of a type of site map. The detection limit for a discrete source is de-

Figure 1: Probability of Detection by Depth
(37 kBq cRa-226 Artifact)



pendent on the depth of the source. Figure 1 shows the probability of identifying a 37 kBq eRa-226 source as a function of depth. These probabilities were determined by empirically verified computer simulation assuming a typical background, and using the most sensitive system settings with acceptable false positive rates. It is clear that discrete sources of this activity deeper than 30 cm are not likely to be seen by the LAGS survey.

In areas where the depth of human disturbance is greater than 30 cm, boreholes or test pits/trenches are required to obtain an indication of the presence of contamination. The frequency of sub-surface investigation levels depends on the circumstances at the site, including the original source, the method of dispersion, and the size of the area. For areas of up to several hectares, where earth moving activities have increased the affected area 10-fold, a 20 m grid spacing for subsurface locations has been used. For sites with shallow disturbance (to about 3 meters depth), test pits are the preferred approach unless restoration of landscaped areas prevents their use. Test pits allow a much larger area for inspection and sampling than boreholes. Such shallow test pits can be excavated, inspected for depth of disturbance, the disturbed area sampled and backfilled, cost-effectively. In areas where a review of surface and subsurface data suggests the presence of contamination, carefully placed test trenches can be used to better characterize the area. The trenches can be installed in lifts, each of which can undergo a surface LAGS survey, and each trench can then be inspected and sampled.

APPLICATIONS

During an investigation of an historic uranium transportation network from Port Radium, Northwest Territories to Fort McMurray, Alberta, a number of sites were identified as having been contaminated by spillage of uranium ore and concentrates. This transportation network served mines operating from the 1930's to the 1950's, and carried ore along a 2,200 km water transportation system, through the barge to railcar transfer station in Fort McMurray, Alberta, and on to Port Hope, Ontario for refining. The original spillage occurred at material handling locations such as portages along the route and at warehouses. However, movement of materials on these industrial sites, such as by grading and tracking by vehicles has distributed discrete fragments of uranium ore over a much larger area.

Preliminary investigation in Fort McMurray determined that some 45 hectares of industrial land potentially contained some amount of uranium ore contamination. The object was to establish the area requiring cleanup and to verify both that the cleanup was successful in the areas where cleanup was conducted and that no cleanup was required elsewhere on the properties. As part of the characterization of suspect properties in Fort McMurray, a representative 10% of the surface areas was surveyed using the LAGS system. Part of this LAGS analysis was conducted in trenches that were excavated in lifts that extended to the depth of disturbance. Data were analyzed for small discrete pieces of uranium ore and areas identified by this method were investigated with an intensive hand-held instrument scan. Identified sources were removed and characterized for uranium content to determine whether the cleanup criterion for the project was exceeded. This permitted characterization of the detailed nature of these contaminating particles (volume, mass, composition) and their spatial density (number of contaminated particles per 100 m^2). Following cleanup, a comprehensive verification program was carried out. The program consisted of 100% surface LAGS surveys, both in areas which had been excavated as well as areas where no excavation had taken place. In areas where no excavation had taken place, test pits were excavated on a 20 m grid, the depth of disturbance limit identified, and a sample representating the disturbed material taken. Trenches were excavated in areas where the analyses of surface and test pit data indicated the likelihood of additional buried contaminated material. In excavated areas, soil samples were taken and portable gamma spectrometer readings were made.

As of 1995, approximately 30 hectares of industrial land in Fort McMurray had undergone characterization, cleanup and verification activities. Following successful completion of the verification program, no restrictions on land use were required. Approximately 31,000 m^3 of contaminated soil has been removed from the properties. The area requiring cleanup was 5.3 hectares. The cost of verification in Fort McMurray is approximately $13,000 per hectare or approximately $12/$m^3$ of cleaned up material, and is roughly equal for verifying that a cleanup is successful or verifying that no cleanup is required. The cost of cleanup, restoration and management of waste for this project is approximately $80/$m^3$. Since about 80% of the 30 hectares affected did not require cleanup, the savings compared to cleaning up the entire area potentially affected by distributed contamination may be $10 to $12 million. Table 1 contains a summary of this information.

In the fall of 1980, mildly contaminated soil was discovered in the residential community of Malvern in Scarborough, Ontario. The contaminant was Ra-226 from a radium recovery operation that took place in the 1940s on a farm that subsequently was developed in the 1970s. The contaminated materials included bulk soil contamination as well as discrete sources of radium-contaminated waste. Soil grading and spreading operations during the construction of the residential lots appeared to have spread these small artifacts beyond the limits of the original farm site. In the spring of 1990, a second finding of radium contamination was discovered within Malvern at a site located approximately one kilometre north of the old farm. Investigations of the new site indicated the primary source of the radioactivity to be radium impregnated pieces of plastic tubing similar to tubing found at the 1980 site. During the summer of 1990 a cleanup was conducted and approximately 20,000 pieces of radium contaminated tubing were identified and recovered from 2,500 m^3 of soil.

The discovery of the second site of radium contamination prompted concern that additional contaminated waste may have been discarded at other locations within the Malvern community. The objective of the survey program was to ensure that no additional sites, similar to the two already found, existed in Malvern. From 1992 until 1995 gamma radiation surveys were conducted at more than 1,000 residential properties and 14 schools in the Malvern area covering about 5% of the total area of Malvern. The surveys consisted of LAGS surface surveys with followup surveys for all alarm conditions. No additional sources of contamination were encountered. This extended survey program, which cost about $875,000, provided sound data to release suspicion from several thousand properties in Malvern from possible future restrictions. Summary data are presented in Table 1.

In 1995, a cleanup of some 16,600 m^3 of contaminated soil was conducted at 65 properties in Malvern. As a result of the extended surveys, 25 of these properties, all within a 0.75 km radius of the original source of contamination had been added to the original list. Following this cleanup a verification program was conducted to ensure that the cleanup criteria had been met. The verification program consisted of a LAGS survey over the entire property, soil samples and portable gamma spectrometer readings inside the excavations and perimeter boreholes logged by portable gamma spectrometry around the excavations.

Table 1: Summary Data For Fort McMurray and Malvern Radiological Surveys

	Fort McMurray	Malvern
Area Potentially Affected With Distributed Contamination	30 ha	1,400 ha
Area Surveyed	27 ha	65 ha
Area Freed From Further Action	21.7 ha	1, 397
Area Found Contaminated (Requiring Cleanup)	5.3 ha	3.3 ha
Cost of Survey	$375,000	$875,000

CONCLUSIONS

Techniques for characterizing distributed sources of radioactive contamination have been developed and applied by the LLRWMO. Unnecessary cleanup work has been avoided and restrictions due to possible radioactive contamination are not required on some 1,400 hectares including thousands of residential properties in the community of Malvern in Scarborough, Ontario, and 30 hectares of industrial land in Fort McMurray, Alberta.

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MODELLING EXTERNAL RADIATION DOSES IN CONTAMINATED URBAN AREAS: IMPLICATIONS FOR DEVELOPMENT OF DECONTAMINATION STRATEGIES

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ABSTRACT

The Chernobyl accident in 1986 highlighted the need for contingency strategies for identification and mitigation of the potential long-term consequences of a radioactively contaminated (essentially with ^{137}Cs) urban environment.

To satisfy this need, the PC model URGENT has been developed. The model predicts, as a function of time, the dose rate in urban environments of various population densities. Input parameters for the model, together with associated uncertainties, were derived mostly from in situ measurements following the Chernobyl accident.

The model shows that in the case of dry deposited fallout, indoor surfaces can make a significant contribution to the total radiation dose. This is addressed in terms of 'location factors' which describe potential fractional dose inside the buildings.

In principle, URGENT can be used to describe any decontamination procedure and assess its effectiveness at any time after deposition. Worked examples of how the calculations can be exploited in the development of decontamination strategies are given.

INTRODUCTION

The PC model URGENT for calculation of the time-dependence over longer periods of the dose-rate contributions from different ^{137}Cs contaminated surfaces in the urban environment is essentially based on a set of coupled first order differential equations. However, the time-dependence can not be adequately described by this system for all radiocaesium migration processes. For instance, the weathering processes for roofs, roads, pavements and walls have been modelled by two component functions, as it was clear from the in situ measurement data recorded after the Chernobyl accident, that different fractions of the radiocaesium initially deposited to these surfaces were weathered off with different half-lives (1).

Where possible, the dynamic model has been based on measured data from investigations reported by other workers as well as measurements made by the Contamination Physics Group at Risø. The calculations of the resulting gamma dose rates are currently made using the dose conversion factors presented by Meckbach et al. (2).

The structure of the model, where the various compartments represent different fractions of the deposited radiocaesium on the surfaces in different states (loosely held, fixed, penetrated, etc.) allows a simulation of practically any dose reduction countermeasure by removal of a part of the contents of one or several compartments.

RESULTS/DISCUSSION

In the following is given an example of an application of the model. Consider a scenario, where a dry deposition of ^{137}Cs has occurred in the early spring (when deciduous trees were in leaf) to a highly populated area in the city, consisting of two-storey row-houses, paved areas, roads, and also some green areas with a few trees.

The relative initial distribution pattern of radiocaesium on different surfaces can be assumed not to differ significantly from the typically recorded values for urban dry deposition (3). Based on a limited local Danish poll, the assumption has been made that the average person spends 85 % of the time indoors and 15 % of the time outdoors. The general validity of this assumption has been indicated by the results of a recent state-wide survey of the activity patterns of Californians (4). It is assumed that

the time spent indoors is equally distributed between the two residential floors of the building. However, in detailed strategy formation, the dose rates to people on the individual floors of a multi-storey building should also be considered, as for instance the contamination on the roofs of these buildings may give a large contribution to the dose rate to people living on the top floor, but at the same time give a negligible contribution to the inhabitants of the ground level flats.

Using the above assumptions the URGENT model can be used to estimate the dose rates or accumulated doses to an average person in the considered type of environment at different times following contamination.

Through the years following the Chernobyl accident numerous clean-up procedures have been investigated (5), and the most promising of these were recently investigated in semi-large scale in the Novo Bobovitsi settlement in Russia. Although a strategy certainly may not include decontamination operations for all types of surface, it was found from the field investigations that for environments such as that under consideration, the most cost-effective means of decontamination of the individual surfaces were probably to use a specially developed roof-washer (5) for roofs, high pressure water treatment for walls, cutting of trees and bushes, sweeping of roads and a 'triple' manual digging procedure (5) for grassed areas and bare soil, whereby the top layer containing most of the contamination is buried under shielding layers of soil. These procedures have been simulated by the model.

Table 1 shows the calculated percentage dose reduction achievable by these methods if they are initiated 6 months or 10 years after the deposition took place. Figures are given for both the reduction of the total accumulated life-time dose over 70 years by application of a method and for the reduction in dose-rate at the particular time when the method is applied. Also given is an estimate of the costs of the procedures.

Table 1. Costs and benefits of application of different clean-up methods in an urban row house environment. Percent reduction of accumulated doses over 70 years and of immediate dose rate reduction are given, assuming that clean-up is initiated 6 months or 10 years after deposition.

Surface:	Roofs	Walls	Roads	Trees	Soil
% 70-y dose red. (6 months after)	2.7	1.6	1.1	4.5	63.7
% dose rate red. (6 months after)	6.0	1.3	3.5	28.2	42.1
% 70-y dose red. (10 years after)	0.3	0.4	0.1	0.2	22.5
% dose rate red. (10 years after)	1.2	1.5	0.5	3.9	68.0
Costs (ECU/m^2)	2	1.7	0.1	7	0.5

From Table 1 it is clear that a cleaning of the areas of soil would give the greatest effect, both after 6 months and after 10 years. As the natural reduction of the dose rate from soil areas is rather limited compared to the effects of weathering on most of the other surfaces in the environment, the dose rate contribution from the soil will become relatively larger with time. As the cost of triple digging is relatively small, clean-up of grassed areas would be given first priority in a clean-up strategy for this scenario.

Even if 10 years go by before the garden is dug, it is still possible to reduce the total accumulated dose by almost one-fourth by digging the garden. However, it is clear that in urban centres with smaller garden areas, the other surfaces will be much more dominant. It is therefore important to tailor a strategy for use in a specific type of area. It should be mentioned that recent investigations in Russia have shown that the dose contribution from roofs after 9 years (and certainly earlier) may in some cases be much greater.

Calculations, in which URGENT output for dose rates inside and outside different buildings due to an outdoor deposition has been compared with a semi-empirical indoor deposition/dose model, have shown that in some dry deposition scenarios the indoor deposition may contribute greatly to the average dose-rate. The semi-empirical deposition model was based on the following equation:

$$D_i / D_o = (V_d / V_{dg})\, f\, \lambda_r / (\lambda_r + \lambda_d),$$

where D_i is the average deposited contaminant concentration on indoor surfaces, D_o is the deposited contaminant concentration on a smooth, cut lawn, V_{dg} is the typical deposition velocity to a cut grassed

surface, $V_d = \lambda_d$ v/A (where v is the indoor volume, A is the indoor surface area and λ_d is the rate coefficient of deposition, in other words: the fraction termed λ_d of aerosols in the building deposited per unit time). λ_r is the rate coefficient of ventilation (the fraction termed λ_r of air exchanged per unit time), and f is the filtering factor (the fraction of aerosols in air entering the building which is not retained in cracks and fissures of the building structure). Parameters for ^{137}Cs, which are believed to be realistic estimates, were found from practical investigations (3,6): $V_{dg} = 4.3\ 10^{-4}$ m/s, $\lambda_d = 0.8\ h^{-1}$, $\lambda_r = 0.4\ h^{-1}$, f = 1.0.

It was further assumed that the average indoor contamination level decreases to 70 % in 10 years. The effective dose rates were calculated at 1m above the floor of a room with a ground area of 4m by 4m and a height of 3m. The dose contributions from scattered radiation and deposition on indoor surfaces of neighbouring rooms were not included in the calculations, which were made for three different types of environment, where the essential difference is the shielding provided from outdoor contamination. The results in Table 2 are presented as location factors (defined as the ratio of the effective dose rate received indoors to that received outdoors following a uniform deposit of radiopollutants).

Table 2. Calculated location factors 10 years after a dry deposition of ^{137}Cs to different housing areas, assuming that no indoor deposition occurs and assuming that an indoor deposition does occur.

Location factor (t=10 years)	Without indoor deposition	With indoor deposition
Low shielding building	0.51	0.59
Medium shielding building	0.091	0.14
High shielding building	0.019	0.083

As can be seen from Table 2, the influence of indoor deposition on the location factor in dry deposition scenarios can be great. Indeed, in areas of buildings with a good shielding effect, this contribution may dominate. Similar calculations of location factors at other times after deposition have shown that the relationship between ^{137}Cs location factors with and without indoor deposition does not appear to change significantly with time.

CONCLUSIONS

The role of computer modelling in development of decontamination strategies has been demonstrated by an example of dry deposition to a row house area. It was found that the garden areas contributed most to the dose to people living in this type of area. Even after 10 years it is still possible to reduce the total accumulated dose significantly. A comparison between the URGENT model results and a semi-empirical indoor deposition/dose model has indicated that indoor deposition may in some cases contribute greatly to the average dose.

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THE GERMAN CATALOGUE OF COUNTERMEASURES - A SUITABLE AND NECESSARY TOOL IN EMERGENCY SITUATIONS

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1. Introduction

The "overview of countermeasures to reduce the exposure to radiation after events of considerable radiological impact" (here briefly: Catalogue of Countermeasures) is a useful manual for selecting protective measures on the basis of simple calculations. The first version of the Catalogue of Countermeasures was duly submitted in the summer of 1992. A new and more synaptical version of the Catalogue of Countermeasures is currently in preparation.

2. Valid range of the Catalogue of Countermeasures

2.1 Target group

Users of the Catalogue of Countermeasures will be those who need to evaluate the information from an event of considerable radiological impact and possibly resulting consequences. These are experts from competent governmental and state fields of authority and persons belonging to the respective advisory and supporting panels.

2.2 Limits of the Catalogue of Countermeasures

In consequence of the manual-like character of the Catalogue, its universality is considerably limited compared to computer programs.

- For pre-calculation, it is necessary to establish specific model parameters. Under certain circumstances, such model parameters can be changed, if this should result in a better estimation of their actual value.

- By a computer program such essential quantities as contamination and radiation exposure can be determined by one process for all involved sites and points of time. By manual, however, these can be determined for always only one site and one point of time. Therefore, on hand of measurement results it is essential to first gain an overview of the sites and points of time for which estimates are to be prepared.

- A particularly important restriction applies to the reference dose value. In the Catalogue of Countermeasures, the dose that corresponds with the reference dose is assumed to be fully exhausted by one exposure pathway. In practice, however, it must be assumed that next to one pathway there may also be others that play a more or less important role. Accordingly, the derived reference value that corresponds

with the respective reference dose value is generally too high. Nevertheless, this approach seems to be justified by the fact that the derived reference value is calculated by basing it on the lower reference dose value of the ICRP bandwidth concept. In addition, the proportions of the various exposure pathways contributing to the total exposure by one single nuclide may be determined by prepared nomograms.

3. Content of the Catalogue of Countermeasures

On the basis of evaluations taken from literature, measures were compiled in the Catalogue of Countermeasures which are discussed in reference to avoiding or reducing the radiation exposure after events of considerable radiological impact. The necessity for this compilation arose from the consequences of the Chernobyl event.

When using the Catalogue it must principally be presumed that not all of the measures will apply to all situations. Still, an attempt was made to provide a comprehensive overview as a basis for possible argumentation if, for example, a specific measure should not be initiated on account of its low effectiveness.

The main criterion for initiating and executing a protective measure is the radiation dose expected to be received from each of the considered pathways (external radiation, internal radiation after inhalation or ingestion). Since radiation is generally not directly measurable, for practical purposes it is not the radiation dose itself but the derived and directly measurable quantities that are used for decision-making.
To calculate derived reference values, models must be used which e.g. include the circumstances of the release due to the condition of the nuclear installation, the dispersion of radioactive substances in the atmosphere, the radioecology as well as the incorporation-related metabolism of the radioactive substance.

The following derived reference values are used:

- time-integrated air concentration in Bq h/m^3,
- surface and skin contamination in Bq/m^2,
- specific activity in Bq/kg or Bq/l.

On the basis of the derived reference values, the Catalogue of Countermeasures is an aid in deciding on the initiation of measures by indicating the type of effectiveness, the effectivity and the possible problems that may occur from the application of countermeasures.

4. Structure of the revised Catalogue of Countermeasures

The revised Catalogue of Countermeasures will consist of two parts. The first part, in form of models and tables, will allow for shortterm decisions on initiating precautionary measures on the basis of available data and includes:

Chapter 1 Introduction

A condensed description of essential general data on the purpose and use of the catalogue.

Chapter 2 Orientation tables and models

The orientation models in this chapter are serving as guide for the use of orientation tables, including criteria for the selection of required measures. On the basis of actually available information, e.g. measured or prognosed data on the time-integrated air concentration of specific nuclides or tracer nuclides, the measurement table of relevance for these data can be determined from this chapter and identified on hand of a model. For each countermeasure, the table refers to relevant additional information and data contained in other chapters of the catalogue.

Chapter 3 Graphic aids

In this chapter, all measures are graphically compiled. This - in addition to chapter 2, although differently demonstrated - is a rapid means of countermeasure orientation on hand of barred diagrams demonstrating the areas where countermeasures need to be applied.

Chapters 4 to 6 compilation of commentaries on all countermeasures

Based on each area where countermeasures apply, the compilation is divided into respective focal points:

- catastrophe protection measures (chapter 4)
- precautionary radiological protection measures (chapter 5)
- measures in the agricultural area (chapter 6)

The text and tables summarize important information on countermeasures, particularly emphasizing on pre-conditions, feasibility, effectiveness, advantages and disadvantages.

The second part of the catalogue contains supplemental information that may be of use in working with the catalogue. This includes, among others, data on nuclear power plants in Europe and tables in reference to sites and inventories. The second part of the Catalogue of Countermeasures contains also a summary of theoretical principles together with the most important equations on which the catalogue is based. This section should provide the background for a more specific familiarization with the catalogue.

ROLE OF DIFFERENT POTASSIUM CONCENTRATIONS ON ACCUMULATION AND EXCRETION OF RADIOCAESIUM IN A FRESHWATER FISH

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ABSTRACT

The objective of this study is the role of different K^+ concentrations in water on radiocaesium accumulation and on the biological half-life, in a freshwater fish, in order of its possible use as a countermeasure.

At the temperature of 12°±2°C the accumulation kinetics at K^+ concentrations of 3.5 and 35 ppm are quite similar and CF's at steady state are one order of magnitude lower than with a K^+ concentration of 0.35 ppm. Therefore K^+ concentration in that range seems not to affect radiocaesium accumulation.

The retention experiment lasted for 130 to 260 days and showed increasing radiocaesium excretion rates at increasing K^+ concentrations.

It is suggested, at least for this temperature, that potassium might be used as a countermeasure in lakes of low potassium content.

INTRODUCTION

The transfer of radiocaesium into and outwards the freshwater biota, is affected by several environmental parameters. Out of the most importants are temperature and water chemical composition, mainly the K^+ concentration, as both elements are chemically analogues and K^+ is the non isotopic carrier of caesium.

It is known that in nature, the radiocaesium bioaccumulation shows a large variability, being higher in oligotrophic lakes with low potassium concentrations and lower in freshwater with high potassium content (1); but the influence of different K^+ concentrations in water on radiocaesium excretion is not very well known.

In different lakes it is possible to say that radiocaesium accumulation by fish is directly proportional to the radiocaesium content and inversely proportional to the potassium content in water (2), (3). Other authors obtained significant correlation between ^{137}Cs in fish and K^+ in water (3), (4). More recently, after the Chernobyl accident, very high ^{137}Cs concentrations in fishes were found, mainly in lakes in Northern Europe, with large differences from lake to lake (5), (6). However, it must be stressed that in these cases, the bioaccumulation occurs not only through the water, but mainly through the food webs. On the other hand, some authors stated the above relationship as being inverse, although not statistically significant (7); but they were mainly analysing temporal trends at several latitudes, dealing with a large variability of data.

The balances of radiocaesium and potassium in water have already been studied in laboratory by several authors and an inverse relation was quoted (8), (9), (10).

Concerning radiocaesium biological half-life in freshwater fishes, several values are reported in the literature (11), (12), (13). However, the influence of soluble K^+ on the ^{137}Cs excretion rate, only recently was studied (10), (14).

The objective or this work, was to study the effect of different Potassium concentrations in water (0.35, 3.5 and 35 ppm), on the radiocaesium biological half-life for a freshwater fish species (*Chondrostoma polylepis polylepis*), at the temperature of 12° C.

MATERIALS AND METHODS

All the experiments were performed in 5 litters aquaria, using an artificial freshwater, without water filtration system, but with aeration, artificial light during 8 hours a day, except for weekends (continuos lighting) and the temperature was kept at 12°±2°C. Artificial medium was prepared, in order to get a basic cationic composition similar to the Tejo River water at the site of Fratel dam. Ca^{2+}, Mg^{2+}, Na^+ and K^+ concentrations were, respectively, 36, 11, 25 and 3.3 mg l^{-1}. Only K^+ concentration was changed and the values used were 0.35, 3.5 and 35 mg l^{-1}. ^{134}Cs was in the chloride form, in a 0.1 M solution of hydrochloric acid, with a concentration of 2 μg Cs^+ml^{-1}. Water was changed once a week and the contaminated faecal pellets were daily separated by screening, to prevent their ingestion by fish.

Small specimens of the cyprinid fish *Chondrostoma polylepis polylepis*, aged about 1 year, with a mean weight of 1.8 g, were used and fed 5 days a week, with milled soft parts of bivalves containing 0.9 mg (K^+) g^{-1}, each meal representing about 5% of the total fish weigth. Each group was previously acclimatised to the artificial medium, for 2 weeks, before contamination with ^{134}Cs. Growth rates were evaluated separately for the three groups of fishes and within them, for uptake and elimination period; growth rates for the whole period were low, varying from 0.0011 to 0.0018 day^{-1}. During the uptake phase, fishes were fed in separated aquaria with uncontaminated water, to avoid any contact of food with radioactive caesium.

Radioactivity measurements were made on pre-weighed anaesthetised fishes. ^{134}Cs concentration in the liquid phase during the uptake, was measured in water samples filtered through membranes (0.45 μm). The measuring equipment was based in a well-type NaI(Tl) detector, associated with a multi-channel analyser.

RESULTS AND DISCUSSION

Concentration factor was computed considering the mean value of water radioactivity for all the uptake period. Elimination of the radioisotope was treated by multicompartmental analysis (15), (16).

^{134}Cs concentration in fish, during 4 weeks of direct uptake, has increased in all three groups without reaching a steady state. All the uptake kinetic curves were fitted according to the treatment described in (15), (16), where fish growth, bioelimination, and physical decay of ^{134}Cs are taken into account.

The higher the K^+ concentration in water, the lower the radioactive contamination of fishes **Fig.1**. The evaluated CFs at steady state for K^+ concentrations of 3.5 and 35 ppm are of the same order of magnitude; uptake curves are almost overlapped, as it happens at 20°C (10). As K^+ concentration in water of about 2 ppm seems to be critical (7), it is understandable that 3.5 or 35 ppm has no significant influence on the radiocaesium uptake.

Biological half-life means the time needed for a retention compartment to loose 50% of its radioactive content. Two compartments were evaluated, **Fig.2**, with the following Tb's: at 0.35ppm K^+, Tb_1=21 days and Tb_2=472 days; at 3.5ppm K^+, Tb_1=24 days and Tb_2=254 days; at 35ppm K^+, Tb_1=21 days and Tb_2=136 days.

The elimination of radiocaesium by fish seems to be affected by the external potassium concentration in water. The longer biological half-life (Tb_2) decreases as the K^+ concentration increases.

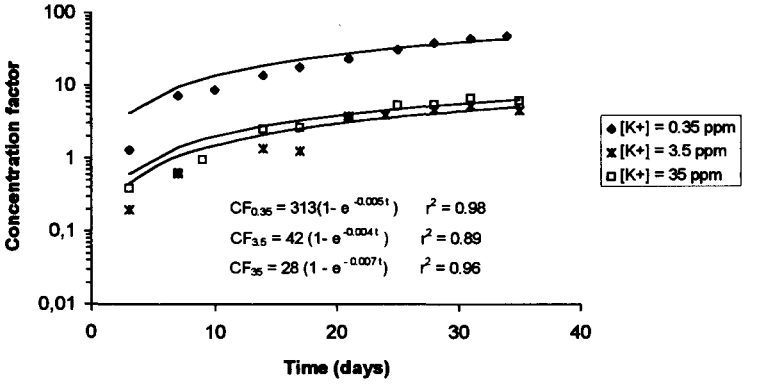

Fig. 1 ^{134}Cs uptake by a freshwater fish from water with different K^+ concentrations.
(CF is the ratio Bq g^{-1} (fish) / Bq ml^{-1} (water))

In (10) the conclusion seemed to be different, however, it needs to be enhanced that it concerned a short-term experiment (about 50 days), while the present one lasted more than 250 days. Retention analysis made at 50 days showed that, in this period, the retention curves exhibited parallel profiles and Tb_1 actually would not significantly differ, but Tb_2 would be very different. A question raises from this data, whether a long-term experiment influences the goodness of the results, although the growth rate will be lower.

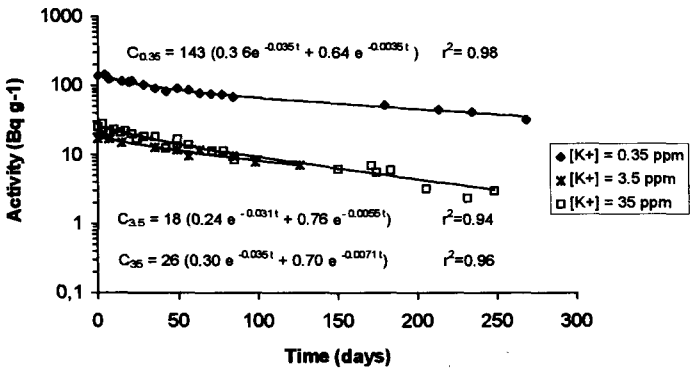

Fig. 2 ^{134}Cs retention by a freshwater fish at different K^+ concentrations in water, at 12°C.
(C is the radioactivity of fishes)

CONCLUSIONS

At 12°±2°C, a long-term retention experiment (more than 4 months), with *Chondrostoma polylepis polylepis*, revealed that at higher K^+ concentrations in water the elimination is faster, i. e., the excretion rate increases with K^+ concentration, therefore the biological half-life of the longer component is shorter.

Relating the interpretation of uptake and loss data, it seems that in water bodies of low K^+ concentration (below 2 ppm), the increase of K^+ concentration might increase the radiocaesium excretion.

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SIDE-EFFECTS OF APPLICATION OF MANURE FROM AFCF TREATED ANIMALS

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INTRODUCTION

AFCF [Ammonium-ferric-hexacyano-ferrate] is a very effective caesium binder. Mixed with the animal feed, presented in the form of salt licks or introduced into the rumen as slow release *boli*, this compound is an efficient countermeasure to limit the gastro-intestinal uptake of radiocaesium by farm animals and wild ruminants (1-3). Less than 1 % of the ingested AFCF is excreted in urine or secreted in milk (2), suggesting that it crossed the gastro-intestinal tract unabsorbed and is finally excreted in faeces together with the caesium bound in the gut. This means that AFCF from treated animals returns directly to pastures while animals are grazing or that it can be spread on fields fertilised with animal manure. Although no toxicological problems have been observed on animals given hexacyanoferrates in the recommended doses, the fate of this molecule in the environment after excretion is not well documented. Except for limited data obtained in Norway and in the CIS (4), practically no information is available regarding its action on the availability of Cs present in the soil, nor concerning potential side-effects of its possible degradation to cyanides and other materials with a concomitant release of bound Cs over long periods of time (2, 3).

MATERIAL AND METHODS

^{134}CsCl and ^{137}CsCl sources were purchased from Amersham and AFCF (Industrial Giese Salt, containing 60-65% AFCF and 35-40% NH_4Cl) was obtained from Riedel-deHaen.

Blank and ^{137}Cs contaminated sheep faeces, with and without AFCF, were successively obtained from the same animal housed in a metabolism cage which was fed 150 g beet pulp per day and mixed grass hay *ad libitum* (fig. 1).

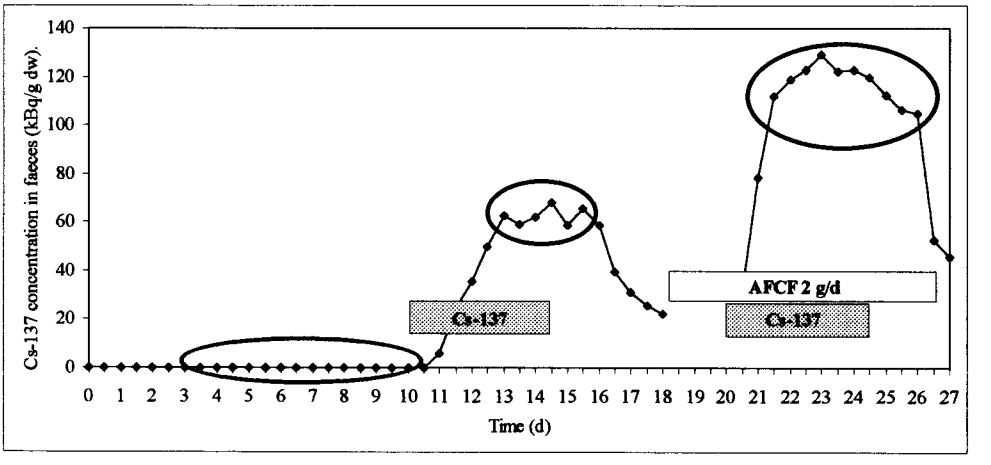

Figure 1: Blank, ^{137}Cs and ^{137}Cs-AFCF faeces production schema. The samples surrounded by an ellipse were pooled to give the different faeces types.

After a period of adaptation to the diet (4 days), Blanca faeces was collected for one week. From the 11th day onwards, the sheep was orally dosed 100 MBq ^{137}Cs.d^{-1} every morning for 5 days, then the

contamination was suspended during 4 days and resumed at a rate of 60 MBq^{137}Cs.d^{-1} for another 5 days' period. Giese salt administration (2 g.d^{-1} given *per os* in the morning, 15 minutes after the administration of Cs) was started two days before the second Cs contamination phase, and carried on until two days after. Both morning and night faeces production was collected. After γ-counting (minaxi, auto-gamma 5000 series, Packard Instrument), faeces with a high and rather constant contamination level was gathered, mixed and ground. Cs-AFCF faeces was diluted with Blank faeces to reach the same radioactivity level as in Cs faeces (i.e. 63 kBq.g^{-1} dw). The AFCF-level of the contaminated Cs-AFCF faeces was estimated from the amount dosed and the faeces dry weight production, assuming that an equilibrium between AFCF ingestion and excretion had been reached when the Cs-AFCF faeces collection was started. After dilution with Blank faeces the concentration was about 0.10 g AFCF.g^{-1} dw faeces. These faeces were mixed with blank and ^{134}Cs contaminated soil. Since 10 g faeces were applied (3.3 t.ha^{-1}), this resulted in a ^{137}Cs contamination level of 630 kBq per container, mixed in the top 0.4 cm layer. The AFCF amount applied with Cs-AFCF faeces was equivalent to 1.0 g AFCF.m^{-2}.

The transfer to ryegrass was studied in greenhouse. Rectangular shaped darkened containers (20x15x11 cm^3) were filled with 3200 g of dry soil (Orthic Podzol). 2 g of Italian ryegrass (*Lolium multiflorum*) was spread evenly over the soil. The seeds were covered with a 0.4 cm moist soil layer homogeneously contaminated with ^{134}Cs (1.0 Mbq per container), mixed with different additives, according to treatment applied. Four treatments were considered:

1. no additives other than ^{134}Cs in the top soil layer,
2. a soil amendment with ^{137}Cs contaminated sheep faeces (3.3 t dw.ha^{-1}) from an AFCF treated animal,
3. a soil amendment with ^{137}Cs contaminated sheep faeces at the same rate but from an untreated animal,
4. the application of an AFCF water solution (30 m^3.ha^{-1}) at a rate of 10 kg Giese salt per ha.

All treatments were set up in triplicate. Pots were watered every two days. Ryegrass was harvested every two or three weeks, depending on growth. Dry weight was determined and activity measured by γ–counting. The ^{137}Cs and ^{134}Cs transfer factors are defined as (Bq.g^{-1} dry plant material)/(Bq.g^{-1} dry soil). After every harvest, plants were given $N_{13}P_{13}K_{21}$-fertiliser (30 ml.dm^{-2} of a solution containing 17 g fertiliser.L^{-1}). In total 19 cuts were harvested over 365 days.

The data were analysed using "Statistica®" as statistical software.

RESULTS AND DISCUSSION

Compared to the control, no statistically significant effect on the biomass production was observed due to the addition of Giese salt (10 kg.ha^{-1}) or animal manure (3.3 t dw.ha^{-1}) (data not shown). The analysis of the variance (two-way anova) underlines a highly significant effect of the time (with an increased yield in spring and early summer), but no difference between the four treatments. The absence of toxic effects on plant growth due to the AFCF applied with faeces is not surprising considering that no effect (visual toxicity symptoms or biomass production decrease) of AFCF has been observed on the same soil type (sandy podzol), after application of up to 1 t AFCF.ha^{-1} in a water solution (5). Such a high dose would only be achieved after more than 270 y on pastures supporting a grazing pressure of 5 cows.ha^{-1}, in so far as the cows are constantly dosed 2 g.d^{-1}, the farm manure produced in the winter period is spread on the grazed pastures and the AFCF molecule is not degraded and its application is thus cumulative.

The availability for plant uptake of radiocaesium excreted with faeces (^{137}Cs) was significantly lower, by a factor 5, in Cs-AFCF-faeces compared to that in Cs-faeces from untreated animal. Moreover, the AFCF brought to the soil (10 kg.ha^{-1}) with Cs-AFCF-faeces (3.3 t dw.ha^{-1}) decreased in the same proportion the bio-availability of ^{134}Cs present in the soil surface layer than that of ^{137}Cs in manure (fig. 2). A comparable effect was obtained by the application at the same rate of AFCF in water.

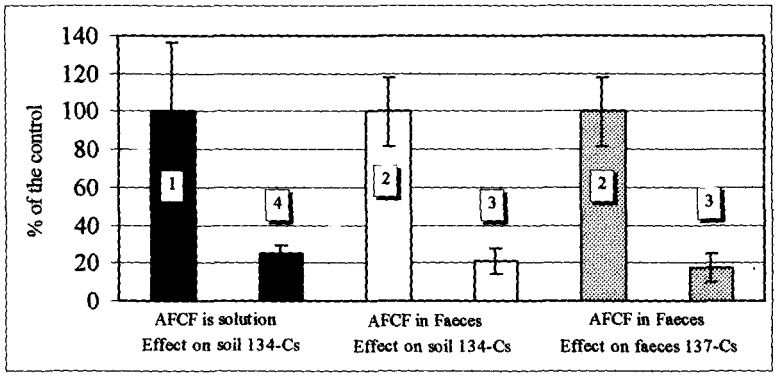

Figure 2: Effect of AFCF applied at a rate of 10 kg.ha^{-1} on the transfer to plants of caesium from faeces or already present in the soil top layer. Average over one year observation (19 successive cuts) for the treatments and their respective control (± mean standard deviation between 4 replicates)

From the comparison of tretaments 1 and 3, regarding the soil ^{134}Cs transfer to plants, we notice that the addition of organic matter can also contribute to decrease the availability for plant uptake of the ^{134}Cs in soil by increasing the soil exchange capacity. In our case, the application of 3.3 t.ha^{-1} of animal manure to a soil containing some 4 % organic matter induces a 20 % reduction of the transfer to plants compared to the non amended control.

CONCLUSIONS

From the results presented here, some conclusions can already be drawn :

- The addition of organic matter (dry faeces) reduces the soil caesium transfer to plants.
- AFCF in faeces from sheep treated with a 2 g.d^{-1} dose limits the availability for plants of the caesium excreted in the faeces.
- The presence of AFCF in the faeces acts also on the availability of the already present caesium in soil, and reduces its bio-availability for plant uptake. This effect lasts for at least one year after the application.
- Taking into account the absence of toxic effect of AFCF addition on plant growth when it is applied on sandy soils as a water solution in concentrations of up to 1 t.ha^{-1}, it is improbable that applying on field or pastures, manure from AFCF treated animals in appropriate doses would negatively affect the biomass production.

However, some attention should be given to the influence of AFCF on the Cs mobility in soil and the risk linked with a potential contamination of the (under)ground waters.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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MAJOR SCIENTIFIC TOPIC NUMBER 4.6 (see page 7)

ABSTRACT (See instructions overleaf)

The selection of a near surface disposal site requires a multi-attribute analysis. Some of these attributes deal with the radiological impact.
The first part of the paper describes the most important radiological protection attributes: individual and collective doses, for workers as well as for the general public, for normal evolution scenarios and for accidental situations during the operational and post-closure phases. The post-closure phase includes also an institutional control phase. For potential exposures, one has to consider also the probability of occurrence of the accident.
The second part of the paper identifies the site specific parameters and the relevant scenarios, i.e. scenarios leading to a radiological impact of more than 10 μSv/year in terms of individual dose and more than 1 man.Sv in terms of collective dose or scenarios with a probability of occurrence of more than 10(-8) per year.
The third part of the paper evaluates the relative importance of these site specific parameters intervening in the relevant scenarios.

ANALYTICAL DETERMINATION OF ALPHA EMITTERTS IN WASTE SAMPLES CONTAINING A LARGER NUMBER OF RADIONUCLIDES

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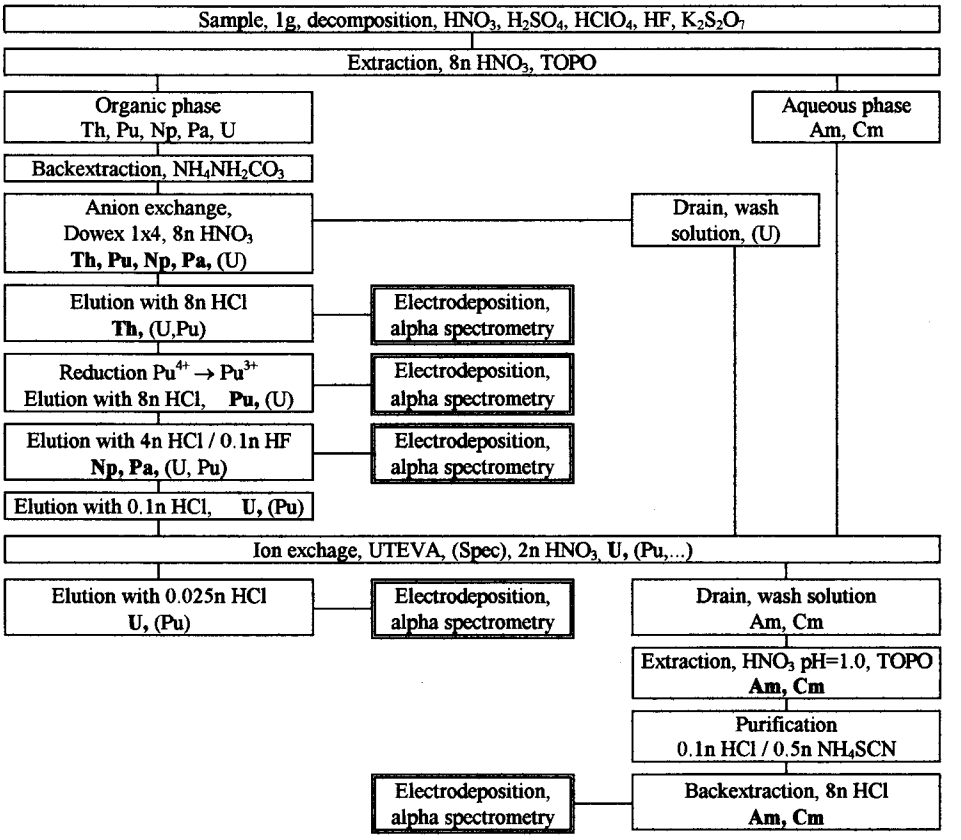
INTRODUCTION

For the disposal of wastes containing alpha emitting nuclides it is necessary to determine the activity of about 20 nuclides of the elements thorium, protactinium, neptunium, uranium, plutonium, americium and curium.
In general, the activity of alpha emitting nuclides is determined by the measurement of samples to be prepared such that the emission of alpha particles takes place largely without loss of energy. However, the separation of the nuclides from sample matrix, purification and preparation of the samples is always connected with losses. Therefore, tracers like U-232, Pu-236, Pu-242, Am-243, etc. which normally are not expected to be present in the sample have to be added in order to determine the yield.
Because of the requirement to determine these isotopes which are used as tracers for yield determination and the determination of very low concentrations in the presence of much higher concentration of other nuclides it was necessary to develop a new analytical procedure.

METHODS

The procedure is schematically presented in Table 1.

Table 1. Schematic survey of the analytical procedure



The idea to perform this complex determination was as follows: The separation of a nuclide to prepare a counting source for alpha spectrometry needs not to be quantitative, if it is possible to detect the residual portions of this nuclide in the other counting sources together with the other nuclides.

Therefore after separation of an element in order to prepare a counting source the following separations have to be performed in such a manner that elements not yet quantitatively separated by previous steps can be detected.

The procedure requires the preparation of up to five counting sources, which give a good energy resolution for alpha spectrometry. Then each alpha spectrum has to be inspected for those nuclides which were not quantitatively separated in previous steps. The activity for each nuclide then results as the sum of activities obtained from the alpha spectra of all relevant samples (see Tab. 2.).

RESULTS

In order to demonstrate the effectiveness of the procedure described above the elements Th, Pa, Np, Pu and U were simultaneously deposited, according to the common radiochemical procedure. Pu-236 was added as tracer. Fig. 1. shows the resulting alpha spectrum. The energy resolution is bad due to the high area weights. The nuclides of interest cannot be estimated with the required detection limit.

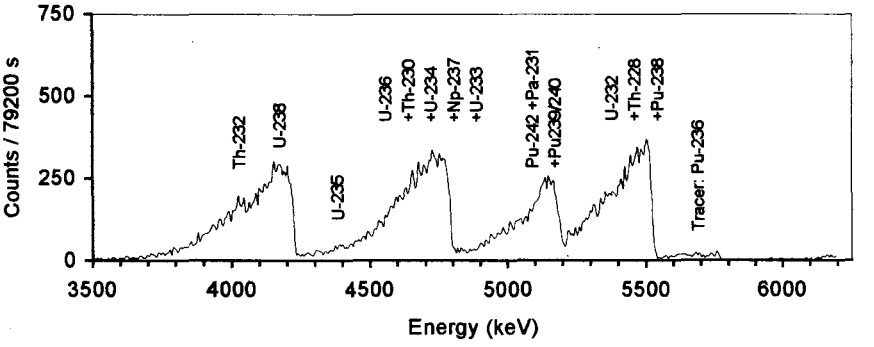


Fig 1. Alpha spectrum of the simultaneously deposited elements Th, Pa, Np, Pu and U. The counting source was prepared according to the common procedure.

In contrast, Fig. 2. shows the alpha spectrum of the uranium fraction obtained by the method described in Tab. 1. All uranium isotopes can now be detected and estimated with required accuracy. As to be expected the counting source contains a portion of plutonium isotopes which are not quantitatively separated in previous steps.

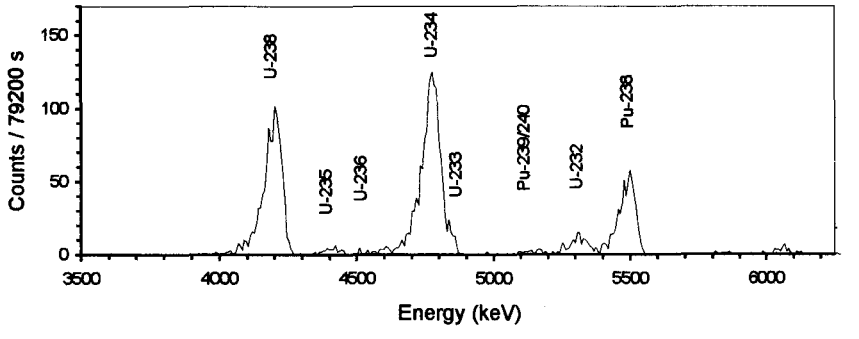


Fig. 2. Alpha spectrum of the counting source prepared from the uranium fraction

Some elements, especially uranium and plutonium appear in nearly all fractions. The activity concentration for the respective nuclide results as the sum of the activities obtained from alpha spectra of all relevant samples. Table 2. shows the nuclide composition of counting sources expressed as the fraction of total activity.

Table 2. Nuclide composition of the counting sources obtained by the method briefly described in Tab. 1.

Counting source	Nuclide composition of the respective sample expressed as the fraction of the total activity						
	Th	Pa	Np	U	Pu	Am	Cm
Th	**96**			6	3		
Pa, Np		**< LD**	**< LD**	**10**	1		
Pu				3	**83**		
U	2			**81**	13		
Am, Cm	2					**100**	**100**
LD: Limit of detection							

In Fig. 3. the resulting spectrum of the sum of the alpha spectra of the Th-fraction, Pa-Np-fraction, Pu-fraction and U-fraction is presented. The resulting spectrum presented in Fig. 3 and the spectrum presented in Fig. 1 represent the same nuclide composition so that they can be compared. It is pointed out that splitting of the counting sources results in a substantial better resolution of the spectra and a lowering of the detection limits.

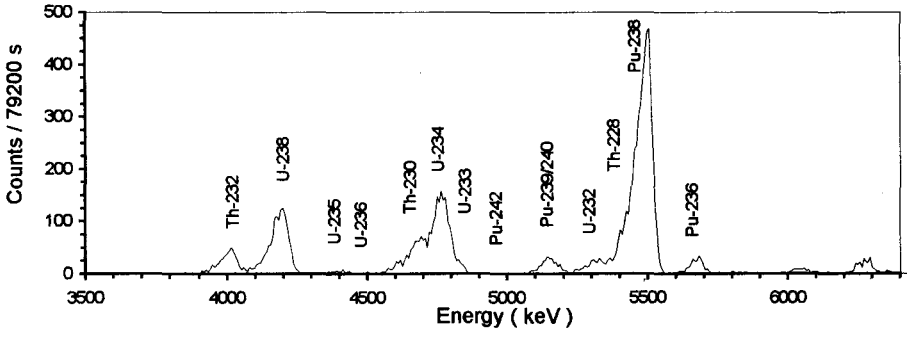


Fig. 3. Sum of the alpha spectra of the Th-fraction, Pa-Np-fraction, Pu-fraction and U-fraction

EDF's "DRA" SOFTWARE FOR THE MANAGEMENT OF RADIOACTIVE WASTE

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I - FOREWORD:

In January 1991, the quantities of radioactive waste produced by EDF's nuclear power plants (nearly 13 000 containers per year sent for storage) led to develop a computerised management system for radioactive waste produced by the operation of nuclear units. The development of this computerised system met the restrictions laid down by the ANDRA (Agence National pour la gestion des Déchets RAdioactifs - National agency for the management of radioactive waste), which is responsible for the storage of such waste and which laid down a validation procedure for the characteristics of packages before they are dispatched to the storage centre.

II - OBJECTIVES:

The initial objective was to standardise management procedures throughout all nuclear sites by combining the framework of regulations for the transport and storage of waste with the need to integrate Quality Assurance rules into the software through the use of *"qualification"* which control access to the various functions and provide for an internal check (operator/inspector).

As later developments were included, the following additional objectives were reached:

- automation of radioactivity calculations,
- provision of a powerful statistical tool which provides for the making of comparisons between sites,
- transport planning aid, etc.

III - FUNCTIONS:

The following functions are provided:

- the management of empty containers (metal drums and vessels, concrete containers),
- the calculation of the radioactivity of containers through the use of mathematical tables, which associate:
  - . the dose rate measured with the activity of each radionuclide detected,
  - . the activities of the radionuclides which cannot be measured with the activity of the "key" radionuclides,
- the management of local storage areas,
- the determination of container characteristics (some of which are generated automatically by the application),
- the real time dialogue between EDF and ANDRA sites through the transmission of data files to ANDRA in order to obtain a "Valid" status for the container in return,
- the calculation of the criteria for the Transport of Dangerous Goods Regulations,
- the making up dispatch batches from the stock of "Valid" containers,

- the inspection of containers using bar codes (reception/dispatch),
- the printing of transport documents,
- the retrieval of data concerning radioactivity, stocks, etc.

Additional functions are also available:

• *for operators:*
  - calculation of radioactive decay to optimise transport resources and costs,
  - a library of reference documents for each site,
  - pre-formatted statistical printouts.

• *for central EDF departments:*
  - monitoring of volumes of products and stocks to establish performance indicators (WANO) and for the planning of transport,
  - adding reference documents to the library.

• *for the National Administrator:*
  - updating of tables,
  - specific transactions concerning data bases (change of status) and the interpretation of ANDRA return codes, etc.

IV - RESOURCES:

IV.1 HARDWARE/ARCHITECTURE

Users provided with local PC type workstations which are connected to a host computer (IBM 3090) via a special TRANSPAC 19200 baud links. The IBM 3090 host computer is connected to the ANDRA host computer via a TCP-IP link operating in a UNIX environment. This system provides for on-going communications between the EDF installation which produces the waste and ANDRA.
The software has a CRADLE type architecture combined with the COBOL programming language.
The various container production phases are managed by status.

IV.2 MODULES

The functions described above form part of 6 main modules:
- qualifications,
- radiological spectra,
- bar code readers,
- container characteristics,
- transmission,
- dispatch.

All these modules are specific to each user site which has its own data bases (access controlled by a "site user code"); the tables and operations in the data bases can be updated by the National Administrator in the event of the failure of the application.

The *"Qualification" module* is used to establish the access hierarchy for the various functions according to the authorisation criteria adopted for the user (Operator, Inspector).

The *"Bar Code reader" module* enables container identification numbers to be input using an optical bar code reader (reception and pre-dispatch inspections).

The *"Radiological spectra" module* provides for the input and storage of the radioactivity characteristics of the containers.

The *"Container characteristics" module* is used to establish the regulation statement for storage.

The *"Transmission" module* is used to transmit previous data to ANDRA in the form of computer files.

The *"Dispatch" module* is used for the preparation of the containers to be sent for storage, according to their shape and size and their level of radioactivity.

V - EXPERIENCE FEEDBACK:

By the end of 1995, the software will be used to manage more than 150 000 containers (under construction and dispatched).
Each container record comprising 80 fields, the disk space was approximately 250 Mb.
Availability of the tool is very satisfactory as is the reliability of the transmission system.
The initial objective has been achieved since the automation of the software has led to significant improvements to the following:

- the container descriptions required by ANDRA for the long term storage of data,
- the internal management within EDF.

The next version of the software will involve its modification for the new processing modes for certain types of radioactive waste (fusion, incineration) and require interfaces to be set up with other contractors.

THE FRENCH CENTRALISED LOW LEVEL RADWASTE TREATMENT CENTER NAMED CENTRACO

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SUMMARY

Socodei, a subsidiary company of EdF and Cogema is commissioned to design, finance, build and operate two low level radwaste treatment facilities : a contaminated scrap metal melting unit, and a solid and liquid waste incinerator.

These units frame a low level radwaste treatment centre named Centraco, located near Marcoule in the south of France, and will receive in 1998 waste coming from dismantling, maintenance and operating works of French and foreign nuclear sites.

The decision to create this centre is due to the low density and large variety of low level radwaste which take a volume out of proportion with their activity, specially in the surface storage centre. Up to now, all low level radwaste were sent and stored with no treatment optimisation in surface storage centres.

Socodei proposes in one single site, to optimise low level radwaste management and reduce the volume of ultimate waste to be stored : in a ratio of one to ten by casting ingots coming from melting contaminated scrap metals ; in a ratio of one to twenty by encapsulating earth ashes and ashes resulting from incineration of solid and liquid waste.

This is a centralised treatment centre and that's why Centraco is a new waste management system. Getting together all means in one place reduces costs, avoids mismanagement and risk increase, and allows consistency in safety, environmental impact, transport and personnel radioprotection.

BACKGROUND AND GENERAL FEATURES

Today, low level radwaste generated on nuclear sites are compacted and shipped in metallic casks to the burial French landfill site operated by ANDRA (Centre de Stockage de l'Aube). Waste are coming from nuclear plants (EDF), reprocessing factories (COGEMA) and from labs (CEA) and hospitals.

EDF and COGEMA have ascertained needs to limit low level radwaste volume and environmental impact of waste coming from their sites. Moreover, in July 1992, the French government enacted a law which assigned industrials to enhance value of all their waste in order to get reusable materials from these so that ultimate waste alone are landfilled. SOCODEI has been founded to adress this issue.

CENTRACO holds two low level radwaste treatment units on a 100 000 square meters area : First is the incinerator designed for 3 500 TPY of combustible solid waste and 1 500 TPY of liquid ; Second is the melting furnace designed for 4 500 TPY of scrap metals.

Low level radwaste management is carried out using one of these two process lines, incineration and melting. All waste generated by the process lines - slag, refractory, baghouse filters, earth ashes and fly ashes,... - are encapsulated in order to be sent to the surface storage center.

All packages belong and are allocated to customers in proportion to their inlet processed weight.

WASTE ACCEPTANCE CRITERIA

Low level radwaste containers or packages are accepted if data sheets specifying nature and level of radioactivity, chemical and physical nature are in agreement with Centraco's acceptance criteria. Waste are then oriented to the optimal process and are stored on site to be processed by campaigns.

The main waste acceptance criteria are :

- Alpha activity level has to be less than 370 Bq/g per package and beta-gamma activity level has to be less than 20 000 Bq/g per package (metal melting & incineration),
- Incoming packages can be drums, metallic boxes or containers less than forty feet length. Pieces have to be less than twelve meters long and fifteen tons weight (metal melting),
- Plastic films and wood wedges for loading are accepted but explosive, inflammable and toxic stuffs are prohibited. All metals are accepted except mercury (metal melting),
- Incoming solid waste packages can be closed cardboard drums or metallic drums (incineration),
- Incoming liquid waste packages can be drums or tanks (incineration),
- Metals, aerosols, batteries and explosive and mercurial stuffs are prohibited (incineration).

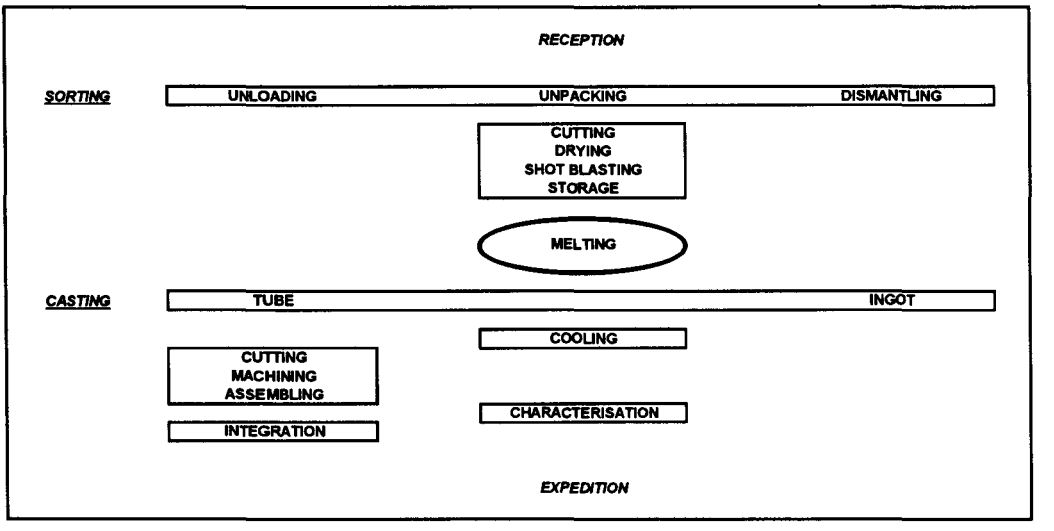
SCRAP METAL MELTING PROCESS

Scrap metals coming in twenty feet containers from the nuclear sites are unloaded and sorted out according to the nature of radioactivity and metal quality - carbon steel, stainless steel and non-ferrous metal. Scraps are first dried out to remove water, then shotblasted to remove painting if necessary.

The high efficiency induction furnace runs at 1 600°C and produces on top of the molten bath, slag with high content of radioactivity scrapped out in specific casks. The molten bath is poured out afterwards in casting casks previously preheated at 850°C.

Molten metal is put either in casts to produce ingots, or in a centrifuge to manufacture tubes. These cylinders are welded with non contaminated disks in order to manufacture shields for casks.

These casks are used to encapsulate and store ultimate waste of high level activity such as spent IER (ion exchange resins) or incinerator ashes.



INCINERATION PROCESS

The twenty feet containers coming in from the nuclear sites are loaded with closed cardboard drums of solid low level radwaste. Drums are unloaded and sorted out according to their heat content and nature of radioactivity, and are sent to the incinerator buffer storage.

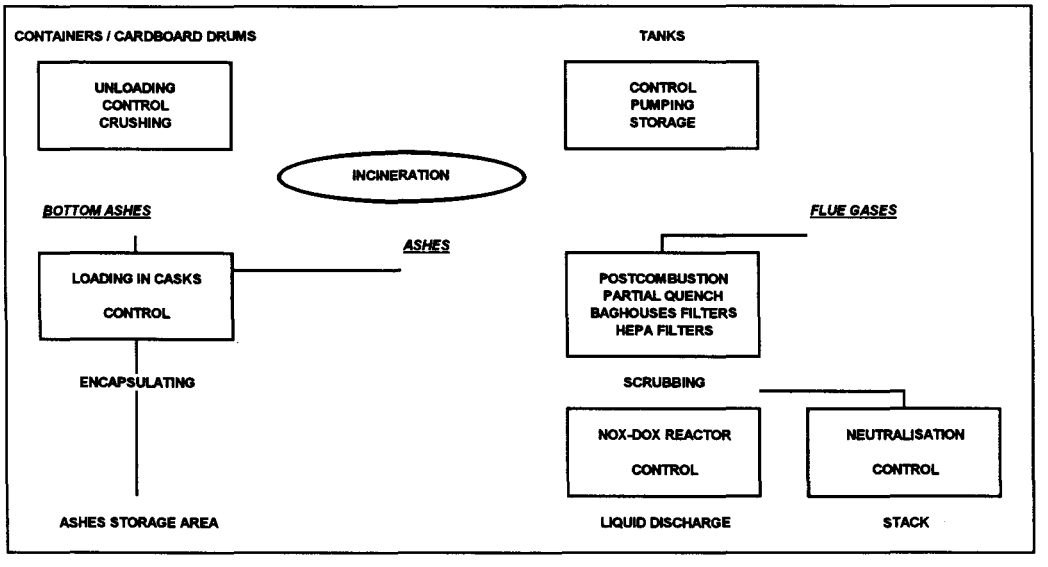
Liquid waste are coming in various containers and are sucked out towards storage tanks specifically devoted to different kinds of liquid. Drums are fed directly without opening through an air lock at the top of the furnace primary chamber at a temperature of 900 / 1 000°C. Liquids are pumped directly out the storage tanks to the corresponding injectors.

While burning, solid waste is moved forward to the end of the chamber by two air-cooled screws, and the resulting bottom ashe is collected by gravity in special high containment casks. Liquid waste is injected through air atomising nozzles in the same chamber.

Support fuel oil is used when the waste heat content is too low. Cooling water is injected on the opposite when the waste heat content is too high. In a secondary combustion chamber, a final combustion is achieved at 1 200°C, in compliance with regulations for incineration.

Flue gases are then cooled down in a partial quench tower to less than 200°C before filtration in series in a bank of baghouse filters and a bank of HEPA filters. Fly ashes are collected in confined casks at the bottom of baghouses filters.

The flue gas treatment consists of two scrubbing towers to remove halogen acids, sulfur dioxide and of a catalytic reactor to remove nitrogen oxides, dioxin and furan. The treated flue gases are discharged through the stack of the site after on-line regulatory control.



COMMENTS

Centraco was asked to be a model plant, and as example has used the ALARA approach right from the early stage of the design. This means that all working stations are optimised to limit the annual dosimetry of the personnel, including both operation and maintenance people.

Centraco will be the link between nuclear sites and surface storage center. The start-up of the melting facility is expected in September 1997, the incinerator will be in operation in May 1998.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE CHARACTERIZATION OF HANFORD HLW BASED ON CORE SAMPLE ANALYSIS

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MAJOR SCIENTIFIC TOPIC NUMBER 4.6... (see page 7)

ABSTRACT (See instructions overleaf)

A preliminary radiological analysis was performed on core samples from the Hanford Reservation high level waste (HLW) tanks. The purpose of the study was to ascertain the nuclide composition of the HLW, and then determine reference dose rates for determination of shielding requirements for transportation casks. Core samples representative of both single-shell tank (SST) and double-shell tank (DST) waste were analyzed for radionuclide content. The main contributors to the total activity were found to be Sr-90 and Cs-137, for both SST and DST waste; smaller amounts of Tc-99, Pu-239, Pu-240, and Am-241 were also identified. Other transuranics, as well as fission products, were present in negligible activities (< 1.0 mCi).

Radiation doses were calculated for a characteristic 10 ml sample size. In terms of the radiological dose equivalent, the alpha and beta decay of the majority of the constituent nuclides could be neglected. No significant neutron-emitting nuclides were identified in the samples, so it was concluded that beta and gamma radiation would be of greatest concern for an unshielded source, specifically gamma/beta decay of Cs-137 and beta decay of Sr-90. Analytical expressions for gamma and beta dose from an unshielded source were derived. Maximum calculated ranges of beta particles in air and tissue were 220 cm and 0.28 cm, respectively, and it was concluded that although the beta dose rate is greatest in magnitude, realistic shielding by the packaging material would reduce it to negligible values. Specific recommendations for gamma shielding, however, were determined for transportation of Hanford HLW core samples.

EVALUATION OF SECONDARY STREAMS IN MIXED WASTE TREATMENT

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INTRODUCTION

The United States Department of Energy (DOE) and its predecessors have generated wastes containing radioactive and hazardous chemical components (mixed wastes) for over 50 years. Facilities and processes generating these wastes as well as the regulations governing their management have changed. Now, DOE has 49 sites where mixed waste streams exist. The Federal Facility Compliance Act of 1992 (1) required DOE to prepare and obtain regulatory approval of plans for treating these mixed waste streams. Each of the involved DOE sites submitted its respective plan to regulators in April 1995 (2). Most of the individual plans were approved by the respective regulatory agencies in October 1995. In many cases, mixed waste treatment that was already being carried out and survived the alternative selection process is being used now to treat selected mixed waste streams. For other waste streams at sites throughout the DOE complex, treatment methods and schedules are subject to negotiation as the realities of ever decreasing budgets begin to drive the available options.

Secondary wastes generated by individual waste treatment systems are also mixed wastes that require treatment in the appropriate treatment system. At large DOE sites, secondary waste streams will be a major influence in optimizing design for primary treatment. Understanding these impacts is important not only for system design, but also for assurances that radiation releases and subsequent radiation exposures will be carefully controlled. Secondary wastes can greatly affect treatment system capacity needs, the types of treatment required, and the health physics program requirements.

MIXED WASTE TREATMENT

Mixed wastes subject to treatment are those wastes that meet one of the following criteria:

- Low-level mixed waste - contains hazardous constituents and radioactivity and is not classified as high-level waste, transuranic waste, spent nuclear fuel, or tailings or waste produced by the extraction or concentration of uranium or thorium from any ore processed for source material.
- Mixed transuranic waste - contains hazardous constituents and radioactive materials contaminated with greater than 100 nanocuries per gram of alpha-emitting radionuclides with atomic number greater than 92 and half-lives greater than 20 years.
- Mixed high-level waste - contains hazardous constituents as well as highly radioactive material with fission products, traces of uranium and plutonium, and other transuranic elements that result from the initial stage of chemical processing of spent nuclear fuel.

A review of some individual site treatment plans indicated that the compounding effects of secondary wastes from these primary treatment processes were not fully understood. The generation of secondary waste was one of several treatment evaluation criteria under the general category, *treatment effectiveness*, and was required to be considered in alternative assessments (3). The importance of this step is realized once the treatment of numerous waste streams begins and secondary waste from one or more treatment modules becomes competitive with primary waste streams requiring the same treatment. We will restrict our discussion to treatment of low-level mixed wastes. Selection of waste treatment technology depends on the physical form and the hazardous constituent or characteristic. With the exception of volatile species such as ^{137}Cs, radiological constituents are generally not considered important for selection of treatment technology. There is a U.S. EPA regulatory requirement to select the best available demonstrated technology for treatment based on the hazardous constituents in the waste.

The design of treatment modules identified during development of the treatment plans requires a thorough characterization of each waste stream. That is, each waste stream must be evaluated completely for physical, chemical, and radiological parameters regardless of its volume (mixed waste volumes range from a few liters to thousands of cubic meters). One obvious benefit from this evaluation is the determination of just how much of the current and projected waste is simply hazardous or low-level radioactive, and how much is truly mixed. Potentially hazardous wastes that do not exceed the hazardous waste treatment standard concentrations listed for wastes prohibited from land disposal may be shipped to and disposed of in existing permitted facilities. Likewise, low-level

radioactive waste that is free of hazardous constituents or characteristics may be eligible for disposal in existing facilities licensed by the U.S. Nuclear Regulatory Commission or a host state.

TREATMENT CAPACITIES

Throughout the DOE complex, a number of permitted systems currently exist for the treatment of hazardous and toxic waste and mixed waste as well. Some of these systems represent the preferred treatment technology for mixed wastes currently stored or being generated at sites other than the host site. Consequently, where the waste streams from these other sites meet the waste acceptance criteria for existing facilities, agreements have been completed or are under negotiation for the acceptance of waste from these other sites. DOE encouraged this practice to the extent it was feasible (3), given the potential constraints of economics and acceptance by the cognizant regulatory authority. Working off the backlog of waste materials at the host site and from off-site locations then becomes a matter of coordination and scheduling.

Mixed waste to be treated on-site (whether generated on-site or from other sites) is subject to milestones included in the site specific compliance order issued by the regulatory authority with jurisdiction for that site. In establishing and then meeting these milestones, consideration must be given to the total capacity needs for any specific treatment technology (primary and secondary streams. During the review of individual site treatment plans, it was apparent that commercially available mobile or transportable treatment modules (4) would be used where possible. For a site with multiple mixed waste streams over a wide volumetric range, rigor must be applied to calculations of the total capacity of a given treatment module. This is especially true if waste stream treatment is to be carried out in parallel.

As an example, one DOE site has a total of 14 individual waste streams in the aqueous liquids/slurries waste category, but the total volume of this category is about 100 m^3, or 86,000 kg. Waste streams ranged from nickel stripping solution, blue print solution, firing range wash water, waste acid and bases, and decontamination wastewater. The preferred treatment technology for this category was physical/chemical treatment followed by recycling of treated water or discharge through a permitted National Pollutant Discharge Elimination System (NPDES) discharge point, and in either case, solidification of residuals. Due to the variety of streams in this waste category, the aqueous treatment module would require, at a minimum, the following capabilities: demusification, oil/water separation, chromium reduction, precipitation (as hydroxide or sulfide), chemical oxidation, alkaline chlorination, sedimentation, filtration, ion exchange, and carbon adsorption. Development of the detailed design or preparation of performance specifications for the aqueous treatment module requires consideration of secondary wastes generated by other treatment modules that will require subsequent treatment in the aqueous treatment module. At the example site above, secondary wastes from the combined treatment of: recoverable metals, inorganic media without arsenic, soils contaminated with chromium, cleanup and spill response residue, mercury-contaminated debris, combustible debris, inorganic metal debris, and batteries, requires 15 times the treatment capacity of waste streams in the aqueous liquids/ slurries alone.

TYPES OF SECONDARY WASTE

Secondary wastes from the primary treatment of mixed waste is a function of the total volume of waste in any given stream. A description of the variety of secondary wastes generated and the respective treatment for each is presented in Table 1. Although these example waste streams are from a single DOE site, a similar situation would be expected at any other site where the same preferred treatment technology is utilized. An example of the impact may be seen here where the required capacity for treatment of existing aqueous wastes is small compared to secondary streams from soil washing, from soft and hard debris washing, from the crusher/shredder, and from deactivation.

RADIOLOGICAL CONSIDERATIONS

From a radiological protection standpoint, special consideration must be given to the handling and processing of secondary wastes. In many instances, for example, where washing/leaching is used as the primary treatment of hazardous constituents, radioactive species may actually be concentrated in the process and thus introduce potential exposure situations that were not encountered in the ordinary health physics program prior to initiation of mixed waste treatment. Expected situations could require the use of shielding material around ion exchange columns where gamma-ray emitters (i.e., ^{137}Cs) have been stripped from aqueous solutions, and around sludge settling basins where gamma-ray emitters were precipitated during primary treatment. Concentrations of ^{3}H, ^{14}C, ^{99}Tc, ^{90}Sr, etc. in secondary streams are very likely to be significantly higher than in the original stream prior to treatment. Thus, special requirements will exist for the sampling and analysis of these secondary streams. Radionuclides mentioned above are beta emitters and are virtually impossible to characterize using in-situ techniques. The laboratory analysis

of secondary stream samples will require significant resources and will require close coordination to ensure that treatment milestones are met.

Managing these secondary waste stream products will present at least some potentially new radiation exposure situations with resulting challenges for the radiation protection staff. Some considerations that will need to be made include the adequacy of training programs for members of the waste management teams and for the radiological control staff, the adequacy of protocols for sampling secondary wastes, the adequacy of and need for additional types of personnel protective equipment (gloves, shoe covers, clothing, respirators, etc.), and the adequacy of portable and installed instruments for monitoring radiation fields in work areas and the environment.

Table 1 Secondary Wastes from the Treatment of Mixed Waste

WASTE STREAM	PREFERRED TECHNOLOGY	SECONDARY WASTES	TREATMENT MODULE FOR SECONDARY WASTE
Aqueous Liquids/Slurries	physical/chemical	sludges oil/grease	stabilize/solidify incineration
Organic Liquids	incineration	ash and bags scrubber sludge	stabilize/solidify stabilize/solidify
Spent Organic Carbon	regeneration	condensed water/organics returned ashes/bags	incineration stabilize/solidify
Recoverable metals	washing/solidification	washwater washwater sludge waste metal oil/grease	aqueous treatment stabilize/solidify stabilize/solidify incineration
(1) Inorganic media w/o arsenic (2) Debris with mercury (3) Batteries	(1) stabilization (2) Acid leach/chemical precipitation (3) recycle	washwater washwater sludge	aqueous treatment stabilize/solidify
Organic Sludge/solids	incineration	ash/bags	stabilize/solidify
Soils with VOCs	incineration/stabilization	scrubber water ash/bags scrubber water sludge	aqueous treatment stabilize/solidify stabilize/solidify
(1)Soils with chromium (2)Spill response residue	(1)wash/chromium reduction/ stabilize (2) characterize/segregate/ treat	(1&2)processed waste (1&2)washwater (1&2)washwater sludge (2)oil/grease	stabilize/solidify aqueous treatment stabilize/solidify incineration
Combustible debris	physical/chemical extraction/stabilization	washwater washwater sludge	aqueous treatment stabilize/solidify
Inorganic metal debris	stabilization	separated water separated oil and grease	aqueous treatment incineration
Lab Packs	incineration/stabilization	washwater from crusher crushed waste streams washwater sludge ash/bags	aqueous treatment stabilize/solidify stabilize/solidify stabilize/solidify

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RADIOACTIVE CONTAMINATION OF RECYCLED METALS

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INTRODUCTION

Radioactive sources commingled with metal scrap have become a major problem for the metals recycling industry worldwide (1). Worldwide there have been 38 confirmed reports of radioactive sources accidentally smelted with recycled metal. In some instances, contaminated metal products were subsequently distributed. The metal mills, their products and byproducts from the metal making process such as slags, drosses and dusts from furnaces can become contaminated. In the U.S., imported ferrous metal products such as reinforcement bars, pipe, flanges, table legs and fencing components have been found contaminated with ^{60}Co. U.S. steel mills have unintentionally smelted radioactive sources on 16 occasions. The resulting cost for decontamination, waste disposal and temporary closure of the steel mill is typically USD 10,000,000 and has been as much as USD 23,000,000. Other metal recycling industries that have been affected by this problem include aluminum, copper, zinc, gold, lead and vanadium.

Exposures to radiation from radioactive sources mixed with metal scrap can be significant, even injurious. In Taiwan, steel reinforcement bars that were contaminated with ^{60}Co were used in building construction and may have resulted in doses of 1 Sv or more to individual members of the public (1). Estimates of whole body external doses received by workers and residents in Cuidad Juarez, Mexico, as a result of a ^{60}Co teletherapy unit disposed as metal scrap in 1983 ranged from 0.13 Gy to 5.5 Gy (1). More recently, in Estonia, a ^{137}Cs source that was mixed with metal scrap was detected by an Estonian scrap metal company and transferred to a waste disposal facility. It was subsequently stolen from the facility. One person who handled the source and stored it in his home died as a result of exposure to radiation and another household member suffered radiation injuries (1).

In the U.S. in 1994, an unshielded 14 GBq ^{137}Cs source was found buried at a scrap metal processing site. It is not known when or how this source entered the plant. It is not known what doses workers at the site may have received. The potential for significant doses from such an unshielded source is, however, clearly evident.

The most prevalent preventative measure that the metal recycling industry has undertaken is radiation surveillance of incoming metal scrap shipments to detect radioactive sources. While such surveillance has resulted in detection of radiation sources that were mixed with metal scrap, even the best surveillance systems cannot provide 100% protection because of the shielding provided by source containers and by surrounding metal scrap. The industry in the U.S. believes that the U.S. Nuclear Regulatory Commission (NRC) and the States should take additional regulatory measures to assure that licensed sources are properly controlled by licensees and, when transferred or disposed of, are done so in an approved manner and, thus, are prevented from entering the scrap metal stream (1).

THE PROBLEM

In the U.S. there are approximately 7,000 persons specifically licensed by the NRC to use radioactive materials. Another 15,000 specific licenses have been issued by 29 States under agreements with the NRC. Most of these licensees are subject to routine inspections by the NRC or the States at intervals up to 5 years between inspections. In the U.S., persons may also possess and use certain devices containing radioactive materials under a "general license". Such devices must meet more stringent standards for design so that they may be used safely by persons without special training in radiation safety. There are about 95,000 general licensees in the U.S. Most general licensees are not subject to routine inspections. Thus, there is a significantly large number of licensees and radioactive sources that are under minimal routine regulatory scrutiny.

One of the most common uses for radioactive sources is in industrial devices for "the purpose of detecting, measuring, gauging, or controlling thickness, density, level, interface location, radiation, leakage, or qualitative or quantitative chemical composition, or for producing light or an ionized atmosphere" (2). Approximately 1,500,000 of these devices have been distributed in the U.S. to general licensees but not all contain radioisotopes in forms and quantities that might concern the metal scrap recycling industry. There are about 86,000 nuclear gauges in the U.S. which contain radioisotopes in forms and quantities that could create radiation and contamination hazards if mixed with metal scrap, e.g., MBq and larger quantities of ^{137}Cs. If warning labels on these devices and gauges become obliterated or knowledgeable personnel at facilities that use them leave employment, the devices may become lost or inadvertently disposed of as ordinary metal scrap. About 200 licensed sources are lost or stolen each year in the U.S. (1). Some have subsequently appeared in metal scrap destined for recycling. In 1991, the NRC proposed a rule that would have required general licensees to inventory their devices periodically and report their inventories to the NRC (3). Resource constraints prevented the NRC from finalizing this proposed rule. Furthermore, resource constraints are tightening: For the fiscal years 1993 through 1996, Congress' appropriations to the NRC fell from USD 540,000,000 to 474,000,000, a 12% reduction. The NRC personnel ceiling in this period fell by 5%. Most State radiation control programs are under similar constraints.

FINDING SOLUTIONS

Given these constraints, the NRC decided to seek alternatives to the 1991 proposed rule and in 1995 formed a joint State - NRC Working Group to review the regulatory program for radioactive devices and develop recommendations for changes, as appropriate. The Working Group conducted two public meetings in 1995 and a public workshop in 1996. A deliberate effort was made by the Working Group to inform and involve persons who could be potentially affected by regulatory changes, e.g., the metal recycling industry, vendors and user of radioactive devices, consultants and service companies, the insurance industry and the public. The Working Group's findings and recommendations will be finalized in May, 1995.

Among the issues that the Working Group is considering are:

- Device design - Currently, design requirements for generally licensed devices are more stringent than for other licensed devices. A different licensing method might require different design criteria.

- Retrofit - Should new requirements be applied to existing devices or be limited to newly manufactured and distributed devices?

- Options for disposal - Disposal options are limited by availability of access to waste disposal sites and costs.

- Device identification - Are there ways of enhancing the ability to identify devices that are improperly transferred or disposed?

- Radiation exposure savings - What savings might result from improved regulatory controls?

- Cost and fees - What options should be considered to pay for the cost of implementing additional regulatory controls?

- NRC and State compatibility - To be effective on a national basis, regulatory changes must be adopted by the States which have agreements with the NRC as well as by the NRC.

Lastly, the Working Group is considering methods to determine the effectiveness of regulatory programs to ensure adequate control and accountability of licensed sources.

INTERNATIONAL IMPLICATIONS

The U.S. has a well developed regulatory infrastructure that dates to the 1950's. Nonetheless, there is concern that the present U.S. regulatory regime does not provide adequate oversight of users of radioactive devices insofar as ensuring that such devices are properly controlled and transferred or disposed of and thus do not pose a radiation exposure hazard to workers, the public and the metal recycling industry. Other national regulatory agencies may also need to review their capabilities in this highly important area. It is important to recognize that recycled metal scrap is an internationally traded commodity and that it may contain unwanted radioactive materials. When recycled metal scrap that contains a radioactive source is smelted, contaminated metal products and by-products can result which could become sources of radiation exposure. These, too, may be introduced into international commerce. Therefore, the problem of radioactive sources commingled with recycled metal scrap requires attention worldwide.

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Planning for unconditional release measurement

Dr. Bodo Krebs, RADOS Technology GmbH

1. Introduction

The term "unconditional release measurement" is taken to mean the release of residual substances or refuse from the realm of nuclear approval regulations. Here, it must not just be ascertained through measurement that the residual substances are free from artificial activity with regard to the official approval values, but also that all measurements are documented and available so that they can be verified quickly and logically. Often, due to the low amounts of residual substances that are released, manual measurements are carried out which only provide an adequate level of safety if the rules regarding these measurements are properly followed. Increasingly, different nuclide inventories in the residual substances must be considered. This requires special procedures in data processing.

That the unconditional release technique is becoming more and more important is at least partially linked to the fact that the dumping costs for residual substances have, as expected, reached levels which make every other legal possibility of disposal economical. Here, the gross gamma method represents a very economical method.

2. The gross gamma method

The gross gamma method is based on an approximate 4π measurement of gamma activities. The better the background, which may be present at elevated levels in nuclear power plants, can be reduced in the interior of the chamber, the lower the measurement times are for the actual measurement. With systems without 4π shielding, the background can lead to material of high density and large volume being unable to be measured according to the limits of DIN 25482 for unconditional release. In particular, changing fields, such as for example contaminated residual substances which are moved in the vicinity of the system, produce erroneous results.

The advantage of the gross gamma measurement is in the very short measurement time(about 10 seconds). Here, precise knowledge of the activity inventory as the nuclide vector is essential, because it is used computationally to find the total activity. Also with unconditional measurement systems, which measure spectroscopically, the laboratory determination of the nuclide vector is essential, because all radiation sources are to be measured, including those which cannot penetrate the packaging material (e.g. Fe-55 and many more). For residual material containing ^{40}K, differentiation may be possible with correspondingly extended measurement times. The extra amount of time required for this however does not lead in most cases to a positive cost/benefit calculation, because contaminated material cannot be assumed.

Unconditional release measurement systems with a lid can also handle long-format material, e.g. fluorescent tubes. An enclosed measurement is then carried out. The length that can be measured is only determined by the chamber volume.

The system calibration is carried out with different types of measurement material, i.e. different refuse substances are combined to form one type of measurement material and a calibration standard is set up for this material. Different interpretation of influences can lead to a conservative or progressive determination of the activities, but normally over-estimation of the activity inventory occurs in both cases. In practice about 10 types of measurement material are used.

3. Large unconditional measurement systems or manual measurements

The question of whether the procurement of a comprehensive unconditional release measurement system is economical, cannot be answered solely based on the quantities of residual substances that are to be measured. Certainly, the answer to the question would be affirmative for quantities over a few 10,000 Mg/a, such as arise with demolition. However, here the fact is ignored that even large systems have their limits with such large throughputs. Unconditional release measurements can only have a supplementary role here.

Anyway, the question becomes more interesting with regard to lower quantities, such as for example a few 10 Mg/a which would arise in normal power plants, because the cost/benefit calculation is significantly less favourably slanted towards an unconditional release measurement system. Apart from the procurement costs, the servicing and actual operational costs must also be taken into account. If this calculation is carried out over a few years, the procurement costs play a lesser role. The servicing and repair costs are clearly below a few 10k DM/a and can therefore be largely regarded as negligible. The personnel costs therefore remain the main cause of expenditure. Here, two fundamentally different methods can be applied. Firstly, charges of residual substances can be measured with high priority and without delay by using the help of external personnel. This method is needed for modification work or partial demolition, where either no deposition area is available or the costs for the unconditional release measurement arise simultaneously. Secondly, the measurements can be taken using one's own personnel in periods of less workload, so that internal storage capacity is relieved. With this method the personnel costs should not be assigned completely to the unconditional release measurement, because they represent part of the fixed costs. Here in present circumstances, the procurement of a system is worth it even with a few 10 Mg, whereas a positive cost/benefit calculation can be assumed for the first method from quantities above 60 Mg with 70% unconditional release.

Substantial personnel costs, which are often not taken into account, also occur with manual measurements. The advantage of manual measurement is in the clearly higher sorting rates of contaminated residual substances. The disadvantage is in the difficult later management of result documentation and the very awkward and time-consuming execution of the method. In addition, with the throughput of larger quantities, a reduction in the measurement accuracy is to be expected. At first sight, this presents no problem until a serious incident arises.

4. Concept of the unconditional release measurement

A chamber with a volume of at least 300 to 400 litres is used for the unconditional release measurement. A drum with residual substances taken from the charge to be measured is used as a standard for calibration. If slightly different measurement charges are to be measured at a later point in time, it is recommended that a specially developed calibration drum is used, which is able to accommodate radiation sources with any iron equivalent. With this method and with appropriate preliminary measurements, almost all refuse drums and refuse densities can be produced. The calibration is arranged to be as progressive as the approval authorities or the regulations allow. The necessary safety margins should already be included in the calibration. Additional safety margins should be defined by the radiation protection personnel or by the operator applying clearly defined rules. This avoids a measurement result being later interpreted as being just under the permissible limits.

In principle however, all unnecessary safety margins which are only required for the operators' peace of mind should be avoided so that the successive limits are not brought far below the activities occurring in nature.

The calibration procedure is described and documented.

The charges of measurement material are measured drum by drum. The drums which lie above the defined limits are sorted separately. Depending on the personnel capacity and time available, the drums which cannot be released unconditionally can be analysed more closely.Graphical representation of the measurement results enable a preliminary estimation. If possible, individual parts with high activity are singled out after the drums have been opened. Here, a manual measurement is again required. The drum should be measured again immediately after this treatment. Using this method, rates of about 80 to 90 % for sorting out can be achieved. It is not recommended that samples are taken of the charge of measurement material.

The most important data from the unconditional release measurement should be fixed to the drum in the form of an adhesive label. All the required data and results should be documented drum by drum. This can take place by a log printout or by saving data files in a data base or, better still, using both methods. For uncontrolled release projects complete and understandable documentation should be maintained to serve as documentation for the authorities.

With present-day PC technology these are requirements that can be easily fulfilled, even far more than is necessary.

5. Conclusion

In the course of forthcoming demolition projects in the FRG important knowledge about the process engineering involved in coping with large quantities of refuse will be obtained. In particular it will be seen that very large single amounts can be measured for unconditional release.

The available devices must be able to be quickly adapted to forthcoming requirements and to special individual needs. Now that it has been shown that the total gamma method represents a successful concept, the measurement technology can be refined such that even more interpretations are possible with regard to the activity inventory.

Enlargened chamber volumes with improved detector quality will increasingly be a feature of future monitor designs.

The documentation of the results will be more and more matched to the requirements of the individual user and will accompany and support the approval procedure for an unconditional release project more than ever before.

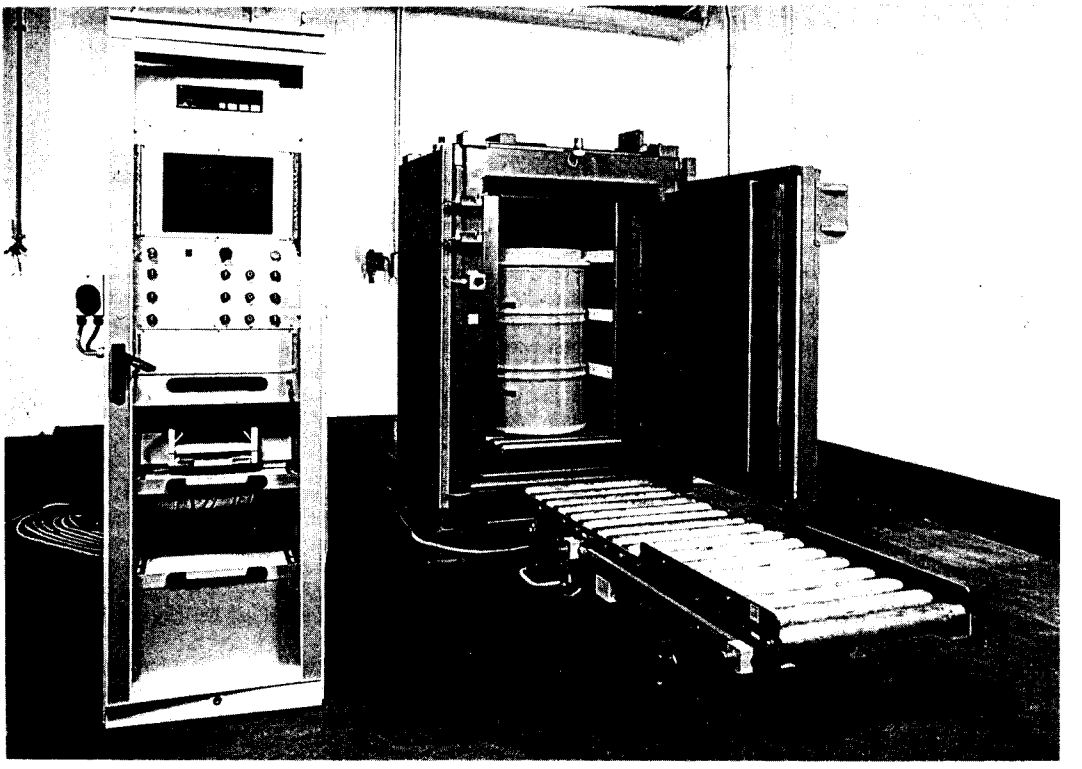

Fig. 1)
Example of an unconditional release measurement system: The RADOS Release Measurement Station H13640 for 200 litre containers. A total of ten separate detectors ensure adequate segmentation. Special design methods in the construction of the frame minimise the effect of the background. The cover and door guarantee optimum charging of the system and the capability of measuring longer objects. With the presence of strong changing fields, a supplementary external detector can be included to take these fields into account.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Abstract No.
Receipt
Author
Acceptance
Mini-Presentation

PAPER TITLE Monitoring of Moving Objects for Radioactivity

AUTHOR(S) NAME(S) Dr. B. Krebs

SUBMITTING AUTHOR

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Quick measurements of objects which are large in size if compared to the detector system, can only be carried out, if the objects are moved past the measurement system. This procedure is to be applied for objects such as vehicles of all kind, waste or recylclates on conveyor belts, or even people.

Especially for large and heavy objects - we like to mention trucks and trains - systems are required, that combine high detector sensitivity with high operation safety. The measurement system must be exact and provide a decision within half a second. Object recognition and measurement start have to be automatic in order not to endanger a smooth operation.

No standardized concepts, for judging the measuement results have been worked out yet. The criteria may be based on dose rate or activity or a combination of both. In many applications these values may have only little to do with each other.

The use of a metal box, in which a defined activity is positioned and with walls that allow to read off a defined dose rate at the outside, has proven to be an optimum for judging the measurement sensitivity of different systems. In practise this procedure rather presents a function test.

WASTE ARISING FROM DISMANTLING OPERATIONS : MELTING CONTAMINATED SCRAP IRON

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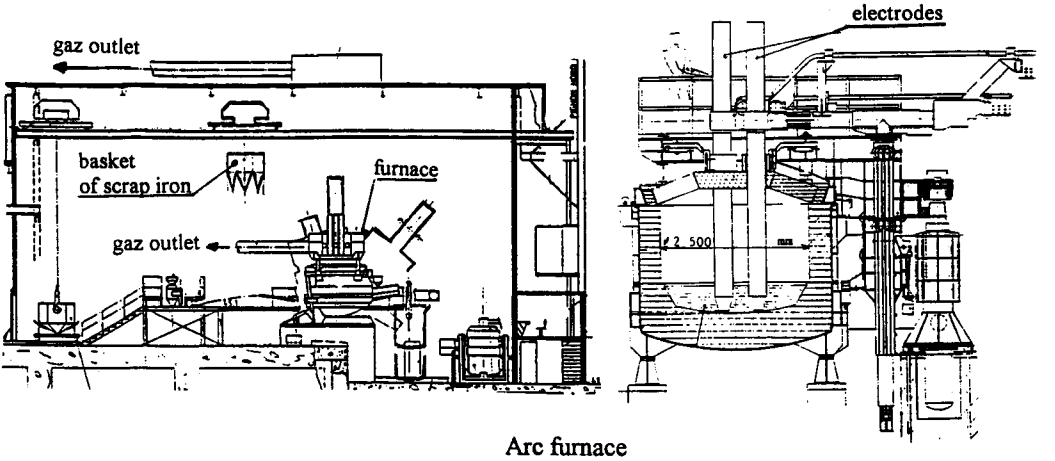
INTRODUCTION

The « Institut de Protection et de Sûreté Nucléaire » (IPSN) assessed the safety of melting operations for contaminated scrap iron which are carried out in an arc furnace with a capacity of 15 tonnes, located at Marcoule. The « Unité de Démantèlement des Installations Nucléaires » (Unit for Nuclear Installations Dismantling) of the CEA « Direction du Cycle du Combustible » (Fuel Cycle Direction), is responsible for operating this furnace.

To day, more than 5,000 tonnes of radioactive iron have been melted, of which 4,000 tonnes resulted from the dismantling of the coolant systems (CO_2) and the auxiliary equipment of the G2 and G3 reactors. Initially, the iron was cast into 25 kg ingots and into 4 tonne blocks which were placed in interim storage in the G2 and G3 facilities. Then, the scrap iron was made directly into either biological shielding or into cast iron containers for packaging highly radioactive waste.

The radioactive spectrum of scrap iron from dismantling of the G2 and G3 reactors was composed of ^{60}Co (95%) and ^{137}Cs.



Arc furnace

DESIGN OF THE FOUNDRY : LIMITING THE RISKS

The maximum allowable level of radioactivity for materials to be melted was set so that the permissible limits regarding radiological conditions at work for personnel, gaseous discharges into the environment and removal of waste, could be respected.

IPSN took particular care to verify that the dose rates in areas where personnel would be permanently present always remained less than 2.5 μGy/h and that the maximum atmospheric contamination level in the melting room always remained less than 80 times the Derived Air Concentration Limit relating to the radiological elements present.

The arc furnace has a cup which enables very large pieces of scrap iron to be melted. This reduces the amount of cutting, dismantling and handling during preparation of the scrap iron in the foundry. This helps to reduce the overall doses and also helps to reduce the risk of dispersal of radioactive materials during these operations.

This is due to the fact that sections of the pipe-work from the coolant systems of the G2 and G3 reactors, which were 1,200 and 1,600 mm in diameter, could be put directly into the furnace cup which has an inside diameter of 2,500 millimetres. It was not necessary to cut these pipes into small pieces, as would have been the case with an induction furnace which has a much smaller capacity.

To limit the ionising radiation exposures of the workers in the foundry, the majority of the operations are carried out remotely from a control room. The presence of operators in the melting room next to the furnace is limited to rake out slag and to control temperature, i.e. to four workers for 20 minutes per heat.

In order to prevent dispersal of radioactive materials in the foundry rooms, the room containing the furnace is maintained at lower pressure by means of a ventilation system. This ventilation system is also designed to capture the gases and aerosols produced by melting, as close to their emission as possible.

These gases and aerosols (400 kg per heat) are removed at three places : directly from the furnace, from the room immediately above the furnace and from the casting area. The extraction network of this ventilation system, removing around 40,000 m^3/h, is equipped with gas cooling devices and three filtration stages. These are a bag filter, a high efficiency filter that can be unclogged, and a final very high efficiency filter.

Around 99% of the dust emitted is removed by the ventilation system and most of it is captured by the bag filter. The dust which is dispersed in the melting room is removed during the weekly cleaning of the room.

After IPSN had assessed the experience gained from one year of operation and the radiological reports, the maximum limit of radioactivity for the material to be melted was increased, for example, for cobalt from 250 Bq/g to 400 Bq/g.

The main radiation risk in the foundry relates to the complete loss of filtration which could lead to a release of contaminated gases and aerosols by the stack, which could reach 2.7 GBq. However, this accidental condition would have no significant effect on the environment.

REDUCING THE VOLUME OF WASTE

In order to reduce the total volume of dust generated by melting and captured by the bag filter, the dust is returned to the furnace to be melted. Experience has shown that this recycling system does not significantly increase the quantity of dust produced by each melting operation. The number of times it can be recycled is, of course, limited, owing to the increase in radioactivity of the resulting dust.

As regards slag, tests showed that its final volume cannot be significantly reduced by recycling, so it is not.

Melting enables the initial volume of metallic waste to be reduced by around a factor of 10. Furthermore, recycling scrap iron in the nuclear field avoids to use new materials (not contaminated) and, therefore, to increase the final volume of wastes.

FOUNDRY OPERATION REPORT

Radioactive elements are mainly emitted when the scrap iron melts, and not when it is put into the furnace. As this melting phase always occurs when the furnace is closed, and the gases are removed directly from the furnace cup, the dispersal of radioactive materials in the melting room is relatively low. It must be noted that scrap iron is loaded four to eight times during each melting operation.

During melting, the behaviour of each radioactive element depends on its metallurgical properties and on the physical and chemical conditions under which melting is carried out. Therefore, some radioactive elements are systematically lost from the melting bath as aerosols, such as caesium, or as gases, such as tritium. Others are found in varying proportions in the slag and dust. Others, such as cobalt, remain almost totally in the cast iron.

The behaviour of certain radioactive elements such as uranium can vary, depending on the characteristics of the melting bath. Indeed, in the case of reducing conditions (when carbon is added at the beginning of melting), a large proportion of uranium remains in the cast iron. However, during oxidizing melting (without

adding carbon), a large proportion of this radioactive element is found in the slag (70%) and in the dust (30%). Moreover, the addition of oxidizing products during melting increases the amount of uranium recovered in the dust. It is therefore possible to extract the uranium from the cast iron produced by melting.

Experience has shown that radioactive elements remaining in the cast iron, mainly cobalt 60, are uniformly distributed in the material.

Since the foundry started operating in 1992, and until the end of 1995, the total exposure of the foundry personnel was 35 man-mSv. This dose equivalent was shared between around 20 to 25 workers and represented, in comparison, less than 6% of the collective dose equivalent received during all the partial dismantling operations of the G2 and G3 reactors. Finally, the radiological cost per tonne of scrap iron melted in this furnace was on average of 7×10^{-3} man-mSv/t.

OPERATING INCIDENTS

Since the foundry began operation, three significant incidents occurred.

1. Simultaneously loading, for test, sections of pipes and 25 kg bags of manganese compounds into the furnace led to high pressure in the furnace which was channelled towards the top through the sections of pipe, thus lifting a melting room ceiling slab. However, no dispersal of contamination was revealed in the hall outside the melting room. The incident was most likely due to the bags being damp. The adding of manganese is given up.

2. 2.5 tonnes of steel tubes with concrete still adhering to them were put in the furnace which already contained 10 tonnes of molten steel, resulting in a violent reaction. Several slabs of the melting room ceiling were lifted. In this second incident, only the external part of the melting room ceiling was slightly contaminated around the damaged area. This incident was caused by the presence of concrete. The adding of concrete is prohibited.

3. Following one melting operation, a fire was revealed in the bag filter of the gas extraction network. Around 40% of the bags were partially burned. The two other filters placed further down the network were smokelogged, which lead to an increase in their pressure loss and to their deterioration. The quantity of radioactivity released through the stack was less than 540 kBq of ^{137}Cs. This incident was due to the introduction of reactive metals (magnesium, zinc, zirconium, ...). The checking of absence of reactive metals is enhanced.

CONCLUSION

The operating experience of the foundry shows that the risks of dispersal of radioactive material and of exposure to ionising radiation are properly controlled and that the main difficulty lies in acquiring a sufficient knowledge of the chemical composition of the scrap iron so as to avoid introducing materials which could lead to violent reactions during melting. In addition, in comparison with conventional steelworks, the effects of violent reactions are accentuated by the small size of the melting room.

Notwithstanding these incidents, it appears that melting radioactive scrap iron in an arc furnace can be carried out in satisfactorily safe conditions providing their radioactivity and above all, their physical and chemical compositions are known. Setting up a quality assurance programme for the reception of scrap iron at the foundry, systematic analysis of test pieces and rigorous visual monitoring enable the properties of the materials for melting to be better discerned, and thus avoid the presence of materials which could lead to violent reactions during melting of the scrap iron.

Recycling steel in the nuclear industry in the form of biological shielding, containers or other semi-finished products, appears to be an economically viable solution and poses few radiation risks. The mean radioactivity per unit of mass of recycled steel in the plant is about 10 Bq/g.

At last, melting contaminated steel reduces its contamination level and concentrates the radioactive elements into a significantly reduced volume with a better confinement.

RADIATION-HYGIENIC CONCEPTIONS ANALYSIS OF HANDLING WITH REACTOR COMPARTMENTS OF BEING UTILISED NUCLEAR-POWERED SUBMARINES

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Elaboration of conception for disjunction and utilisation of ships and vessels with nuclear-powered plants (NPP) removed out of exploitation includes the following main decision:

- Tentative estimation of time-limit after removal of concrete object out of exploitation on the expire of which reactor's compartment (RC) utilisation will come possible, and utilisation technology option;
- The choice of version for RC temporary storage or burial;
- The choice of disjunction and utilisation technology of bow and stern compartments.

The key moment under these questions decision is the radiation state of removed out of exploitation and prepared for utilisation of ship NPP. After discharge of liquid and solid radioactive waste there remain the sources of 2-types radiation in reactor compartment volume: volumetric-radiating-activated material of NPP equipment and superficial-radiating-contaminated by radionuclides equipment's and constructions' surfaces.

Incidentally these sources integral activity with 1-3 years delay time reaches $(1\text{-}2)\cdot10^{15}$ Bq. More than 90 % of long-lived radionuclides are localised in intrareactor constructions that amount to not more of 5-7 % of total radioactive materials mass.

Radiation contamination's level of intrareactor constructions surfaces may reach 2 millions Bq/sm^2, of internal surfaces of the 1-st circuit equipment elements - 100 thousand Bq/sm^2, of the surfaces of the 3-d circuit equipment elements, of desiccation and repair cooling tanks - 20 thousand Bq/sm^2 depending on construction period the surfaces contamination's activity can be conditioned by corrosion products (by Co-60 on the whole) or by radionuclides of fragmentation origin: Cs-137, Cs-134, Ce-144, Ru-106, Sr-90.

Actinide radionuclides activity in the 1-st circuit communications volume of water-cooled ship's NPP can reach $4.5\cdot10^{11}$ Bq.

By 1-3 year delay-time induced activity of construction of exploitation ship's NPP in the places most close to the active zone for reactor casing reaches $2.6\cdot10^{11}$ Bq/kg order value, for reactor caisson - $1.5\cdot10^{11}$ Bq/kg, for steam generator case - $4\cdot10^{8}$ Bq/kg, for pressure hull under reactor - $2\cdot10^{10}$ Bq/kg.

Two conceptions of handling with RC radioactive saturation have been discussed: the first one consists of the absence at present of enterprises with appropriate equipment for disjunction of RC with high-activity equipment and their long-term keeping is necessary for natural activity decay (during 50-80 years) defined by method worked out in our institute. After long-term keeping RC can be disintegrated without large dose expenditures and as a rule - without special remote and protective equipment. Thus for 70 years of keeping the rate of activated gamma-radiation sources will reduce in 1000 times on average.

The alternative conception provides for complete utilisation (remelting) of RC equipment and case.

However RC disassembling immediately after ship's removal out of exploitation is connected with considerable dose loading, heightened risk of environmental radioactive contamination and necessity of using special equipment with remote control. Really, at being

repaired NPS the dose rate in reactor caisson was up to 400 mSv/h, near the reactor case 20-40 mSv/h, in steam generator caisson - 2 mSv/h. Thus, the present conception realisation is possible only after creation of specialised ship disintegrating enterprise, equipped with appropriate technology, that requires large-scale capital investments. Owing to the told the idea of disassembling of RC radioactive equipment before its maintenance did not obtain spreading neither in home nor in foreign practice and in spite of its attractiveness can be regarded only as perspective while keeping in containers ship's NPP equipment of NPS pressure hull and pressurising bulkheads upon its face planes has been realised both in our country and in USA. For normalisation of the radiation situation in conformity with sanitary requirements in the lower part of compartment, in the reactor's arrangement zone, separate areas of pressure hull (container) are covered by proper biological protection. The thickness and configuration in every concrete case can be determined by dosimetric inspection results.

Prepared for burial compartment must correspond to IAEA requirements (1) for type "B" radiation packings (by tightness and impact strength) and PTBRV-73 (2) for the 4-th transport category (by external gamma-radiation levels).

In this case equipment of bow-and stern compartments is disassembled, extracted and either comes for utilisation or is reserved for the second employment. Stage by stage ship's constructions are disintegrated in blocks.

Personnel doze loads under reactor compartment cutting in lower point of external line (without additional protection) average 1 MSv/h magnitude order (under 3-5 years delay). The direct works under reactor's external line require supplementary technological protection, otherwise dose limits will be exceeded in several days or even hours. Our experience indicates that by carrying out factory NPS repairs including opening and decontamination of biological protection tanks, personnel irradiation doses do not exceed 20-30 mSv/year. As well in USA personnel was not exposed to gamma-radiation doses exceeding 20 mSv/year under cutting and preparation for burial NPS reactor compartments (3).

By maintain of main principles of conception various versions of prolonged RC keeping organising can be offered:

- afloat at allocated region of water area;
- inundation in shallow water of allocated region of water area;
- in underground workings-galleries;
- in concrete trenches or on the surface-type grounds protected from precipitation;
- in encirclement of artificially frozen ground, in conditions of permafrost and polar climate (at present thus proposal is being working out in reference to conditions of Novaya Zemlya archipelago).

Afloat storage of ship NPP removed out of exploitation (forming part of NPS, three - compartments blocks or reactor compartments) are of radiation-ecological danger and contradict standards and international recommendations (4-7), being in force forbidding storage of radioactive waste on flooded and waterlogged territories, what is more afloat or under water, in shallow water.

So, this practice should be regarded as forced and provisional. The terms of afloat storage as well as quantity of objects should be reduced to a minimum.

Being in force sanitary regulations and ICRP recommendations exclude the possibility of radioactive waste depots arrangement in flooded and waterlogged places, in regions situated nearer than 500 m from the reservoirs, as well as by ground waters level above 4 m from the depot floor (4-6). Specified requirements should be carried out beyond dependence upon the degree of depot waterproofing, that significantly makes difficult the practical realisation of version NPS' RC arrangement in underground workings-galleries.

The most reasonable and ecologically safe version in organising RC long-term storage of NPS removed out of exploitation is their arrangement in concrete trenches (holes), protected from atmospheric precipitation, or in polar climate conditions in permafrost or artificially frozen ground.

Organising RC storage on the ground surface or in shallow holes (10-20 m deep) and in permafrost ground encirclement also does not contradict with sanitary-hygienic documents requirements.

From the point of view of environmental radioactive contamination the version of storage on specially equipped coast region is of less danger. In this case depot should have not less than two bars on the possible environmental radionuclides route entry. The first includes engineering equipment of storage place for prevention of meteorological and hydrological factors influence, which may break compartment's tightness (covering shield) and engineering system for prevention of radionuclides entry in environment in case RC loss of sealing (spreading shield). Geological formations in depot's arrangement place which must limit radionuclides dissemination are the second bar.

Requirements, similar to arrangement and equipping of points for radioactive waste burial are brought for plot choice.

By execution of all necessary requirements for the storage place the danger of activity dissemination in environment is insignificant and is clearly defined primary by possibility of outward reasons influence (hurricane, whirlwind, aircraft's fall).

As the advantage of version for RC burial in permafrost is the unlimited heat capacity of surrounding rocks and absence of activity dissemination with ground waters as in conditions of frozen rocks of arctic geocryologic region free water completely freezes solid forming so called constitutional ice in which migration process physically cannot be realised.

RC burial in five meters deep from the top part of compartment down to permafrost surface ground satisfy IAEA requirements and home regulating documents for radioactive waste burial (4-6).

The final decision about carrying out of utilisation of equipment materials and constructions of RC removed out of exploitation NPS may be accepted only on the basis of results of materials radiometric examination.

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PRODUCTION AND RADIOACTIVE-WASTE DISPOSAL
A MATHEMATICAL MODEL

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INTRODUCTION

Radioactive-waste materials production in Bologna commune is due almost esclusively to research activity (University, C.N.R., E.N.E.A.) or to hospital assistence activity (Nuclear Medicine). Radioactive-waste products - different for half-life, radiotoxicity, but also chemical and biological toxicity - , can follow different ways of disposal and they can finish incontrollably in the environment.

For estimating the radioecological impact resulting from abovementioned activities , a census of users and main used radioisotopes has been done.

Different methods of disposal has been analyzed and radioactive measures in different samples drawed from city sewer dynamic system - expecially at the entry and at the exit of water depurator. Since the greatest contribute to environmental radioactivity rises from Nuclear Medicine diagnostic protocols, a simple mathematical model that correlates the activity administered to patients with the activity at the entry and at the exit of depurator in different matrix (solid, liquid,and gas matrix) has been created and verified .

A good realiability of model for the most important radioisotopes , like Tc-99m, I-131 ang Ga-137 has been verified . Admissible Maximum Concentration (A.M.C.)has been compared with concentration derived by human activities in water and air.

It has been concluded that the risk for the population is now difficult to point out.

RESULTS

Results are reported in the following tables. Beta-emitter radioisotopes mainly used by laboratories of the University,as H-3, C-14, S-35, P-32, Ca-45 etc., were not foundet at the entry of depurator and not even in other "hot points" in the city sewer system, at least with regard to concentrations higher than 1 Bq/litre. This result is not surprising because radioactive waste products of research laboratories are picked up and treated by authorized firms.

Tab.I shows most important radioisotopes used in Nuclear Medicine, their gamma energy levels (keV),their T(1/2) and the annual amounts administered in Malpighi and Maggiore Hospitals in 1993 (MBq). We can see that more than 95% of activity is imputable to Tc-99m.

Radiois.	En.(keV)	T(1/2)	Malp(MBq/y)	% Malp.	Magg.(MBq/y)	%Magg.
Tc-99m	140	6 h	3,7.10^6	95,5	2,8.10^6	95,4
I-131	364	8,04 d	6,7.10^4	1,75	7,4.10^4	2,5
I-123	159	13,3 h	1,6.10^3	0,04	-	-
Ga-67	91/185 300	78,1 h	4,5.10^4	1,1	7,4.10^3	0,25
Ta-201	81	3,1 g	6,3.10^4	1,6	3,3.10^3	1,14
Xe-133	809	5,3 g	-	-	1,9.10^3	0,64
In-111	171	67,4 h	5,5.10^2	0,02	1,9.10^3	0,06

Tab.II enumerates the really used activities during 1993 in Malpighi Hospital because of exams on day hospital's patiens and hospitalized patiens. The real activities are compared with the activities calculated by us starting from measures results(MBq).

Radiois.	Day hos.p.	hospit.p.	%	Tot.Admin.	Tot. Calcul.
Tc-99m	2,32.10^6	1,65.10^6	95,7	3,96.10^6	3,7.10^6
I-131	5,99.10^4	0,48.10^4	1,5	6,48.10^4	6,7.10^4
I-123	1,67.10^3	6,66.10^3	0,2	8,33.10^3	1,6.10^3
Ga-67	2,46.10^4	2,79.10^4	1,3	5,25.10^4	4,5.10^4
Tl-201	3,11.10^4	1,67.10^4	1,2	4,77.10^4	6,3.10^4
In-111	1,85.10^3	1,85.10^3	0,1	3,70.10^3	5,5.10^2

Tab.III schows the activities Weekly administered to patiens (Adm.) and the activities measured on the entry of depurator (Entr.) during each measurement week (MBq).

Radio-act.	Tc-99m	I-131	Ga-67	Tl-201
Adm . '90	163400	2400	1400	2500
Entr. '90	16800	500	200	tracce
Adm . '91	171000	5600	1800	3200
Entr. '91	22400	1200	300	tracce
Adm . '92	142000	3700	1500	1900
Entr. '92	17900	tracce	tracce	tracce
Adm . '93	161000	1700	1500	2400
Entr. '93	19500	500	60	tracce
Adm . '95	168200	3890	962	1010
Entr. '95	15700	900	200	tracce

MATHEMATICAL MODEL

To formulate a mathematical model to value the environment impact starting from the activities administered to patiens we considered: 1) The activities administered to patiens; 2) the half-time T(1/2); 3) the activity distribution in the body; 4) the activity excreted by the patient after three hours since administration; 5) transit time in the sewer city system; 6) activities that,even if escreted, don't reach the depurator (sedimentation, decay, volatilisation).Transit time between the 2 Nuclear Medicine divisions and the depurator was evalued as 3 hours. The treatment of waste waters lasts about 49 hours.

According to the proposed mathematical model, 12% of Tc-99m administered to the patients reaches the depurator and 0,04% leaves the depurator in several ways (ashes,waters,gases).

The final formulation of this mathematical model is :

$$AR = \frac{AS \left(\frac{1}{2^{(TM/T(1/2))}} - K1 \right) \left(\frac{1}{2^{(TT/T(1/2))}} - K2 \right)}{2^{(TD/T(1/2))}}$$

AR = Activity really released in environment
AS = Activity administered to patients.
K1 = (%) not excreted the first time.
K2 =(%) not reaching the depurator.
TM = Average time between administration and 1° excretion (3 h).
TT = Transit time in the sewer city system (3 hours).
TD = Total time required for depuration treatment (49 hours).

Tab.IV reports the good agreement between activities measured each year and values calculated with mathematical model.

Year	Fraction at the entry (%)		
	Tc-99m	I-131	Ga-167
1990	11	21	15
1991	13	22	15
1992	13	22	17
1993	12	72	5
1995	10	23	20
Val.Calculated (Mathem.Model)	12	23	17

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE A Simple and Robust Precipitation Method for Treating Transuranic Waste Solutions

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MAJOR SCIENTIFIC TOPIC NUMBER 4.6 (see page 7)

ABSTRACT (See instructions overleaf)

The purpose of this study was to devise a treatment method for transuranic (TRU) waste solutions that would remove the TRU elements and concentrate them into a small volume that is suitable for deep geologic disposal. A process that uses magnetite (Fe_3O_4) as a carrier for the TRU elements was developed. This compound is precipitated by adding sodium hydroxide to a solution containing Fe(III) and Fe(II) ions in a 2:1 molar ratio, respectively. It is a very efficient carrier for the TRU elements, and gives decontamination factors (DF, initial amount of TRU element in solution/amount of TRU element remaining in solution after the precipitation) of the order of 10^4 in one precipitation over a wide variety of conditions. The magnetite can be filtered, dried, and packaged for disposal or long-term storage. The remaining solution can be treated as low-level waste since it should contain <100 nCi TRU/gram. If one precipitation does not reduce the TRU concentration to this level, an additional precipitation can be performed. The magnetite can be formed by an *in situ* direct strike (the sodium hydroxide added last), by a reverse strike (the iron solution added last), or by adding pre-formed magnetite to the TRU waste solution.

Among the variables studied were the carrier concentration; the ferric to ferrous ratio; the final pH; the effect of stable anions and cations, including organic complexing agents the precipitate digestion time; initial TRU concentration; precipitation temperature; rate of precipitation; and method of precipitation. The effects of these variables on the decontamination factor will be presented.

REMOVAL OF ^{226}Ra FROM WASTE WATERS RESULTED FROM MINING AND PROCESSING OF URANIUM ORES,WITH NATURAL ZEOLITES

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ABSTRACT

The zeolites were of clinoptilolyte type and the used waste waters were: mine drainage waters, waste waters from radiometric sorting and tailings pond waters from a uranium processing plant.

On the basis of experimental results, the parameters of the procedure have been established, wich enables one to obtain a radium removal efficiency higher than 95% at the passage of over 2,000 BEV waste water.

INTRODUCTION

The rasearch was done on laboratory scale and was started to find out an efficient procedure for removal of ^{226}Ra from industrial effluents resulted from exploitation and processing of uranium ores.

For type of zeolitic tuffs from three autochthonous deposits were tested. The zeolite occuring in these deposits is of clinoptilolyte type.

METHODOLOGY

The experiments were done in two steps: in the first step were used synthetic ^{226}Ra solutions to obtain information regarding the removal of the radionuclide by natural zeolites and in the second step using real waste waters resulted from uranium industry.

The experiments were conducted at ambient temperature and atmospheric presure, in dynamics, the ^{226}Ra solutions being passed through sorption columns. The diameter of columns was 10mm, the sorption bed being made up by zeolitic tuff grains of 1-2mm in diameter, previously washed out to remove the fine particles of tuff.

The effluents of columns were collected in fractions having different volumes, depending on the specificity of the experiment. The fractions were analysed for their ^{226}Ra content, being measured the remanent activity of the radionuclide. The radionuclide was measured using its daughter ^{222}Rn.

RESULTS AND DISCUSSION

The experiments were conducted simultanously using the four type of zeolitic tuffs with clinoptilolyte contents ranging between 64 and 77%. The behaviour of the four tuffs was very similar, so that only the obtained average values will be reported.

Removal of radium 226 from synthetic solutions.

Synthetic ^{226}Ra solutions used in the first step of the experiments were prepared by contamination of tap water by Ra Br_2 solution with known activity concentration.

The experiments proved that in the same geometry of the sorbent bed(height-diameter ratio:h/d) the removal efficiency depends on the specific lauding(burden) of the column and implicity on the contact time between the liquid phase and the sorbent. So, passing a solution having 3.7 $Bq^{226}Ra/L$ through a sorption column with h/d=8.4, at different specific burdens were obtained the following results:

Specific burden	Remanet activity	Removal efficiency
[BEV/h]	[10^{-3}Bq/L]	[%]
10	18	99.5
15	18	99.5
20	55	98.5
25	100	97.3
30	174	95.3

It can be noted the very elevated removal efficiency of radium by zeolites, in the conditions when the volume of radium solution passed through the columns was 120 BEV(Bed Equivalent Volume).

The dependance of removal efficiency on the contact time between the liquid phase and the sorbent is shown in Figure1. The optimum time in experimental conditions is of 4 minutes corresponding to a specific burden of 15 BEV/h.

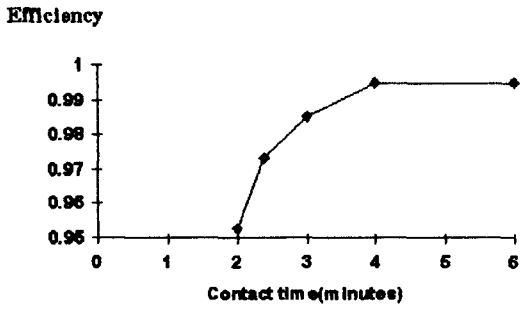

Figure 1. The relationship between removal efficiency and contact time

At this specific burden of 15 BEV/h, synthetic solutions having 0.92, 3.7 and 37Bq ^{226}Ra/L, respectively, volumes of 160 BEV of each were passed through sorption columns having the ratio h/d = 16.8.

The average remanent activities were $7 \cdot 10^{-3}$, $18 \cdot 10^{-3}$ and $451 \cdot 10^{-3}$ Bq ^{226}Ra/L, respectively, the corresponding removal efficiencies being 99.2,99.5 and 98.8%, respectively.Thus, one can note that for a large range of the activity concentrations, the removal efficiency has very high value.

The geometry of sorption columns proved to be unimportant.Thus, at the passing of 120 BEV radium solution of 3.7Bq/L through columns having h/d = 4.2, 8.2, 16.8, and 33.6, at the specific burden of 15 BEV/h were obtained similar efficiencies(99-99,5%).

These encouraging results obtained for the removal of ^{226}Ra from synthetic solutions using columns fieled up with natural zeolites, induced the attempt of using of the same sorbent bed in more cycle after the elution of retained radium. In a test, the same sorbent bed was used in 4 cycle of sorption-desorption , the elution of radium being made using a 5%NH_4NO_3 solution. The specific burdens were 15 BEV/h at sorption and 5 BEV/h at elution, the volumes being 100 BEV for the 3.7 Bq ^{226}Ra/L solution and 20 BEV for the NH_4NO_3 one. In these conditions, even in the fourth cycle, the removal efficiency was over 99%.

Removal of radium226 from true waste waters.

Three uranium industry waste waters were used :mine drainage water, water resulted from radiometric sorting of uranium ore, and effluent resulted from processing of uranium ores.

Due to fact that the activty concentrations the three industrial waters were very low(0.19, 0.56 and 0.17 Bq ^{226}Ra/L, respectively), the sorption test were made using industrial effluent contaminated with 3.7Bq ^{226}Ra/L. So, waste waters having an actual chemical composition and an appropriate activity concentration of radium were obtained. The waste water volumes pased through columns were at least 160 BEV. The experimental results are shown in Table 1.

From the data shown in Table 1 it can be seen that the chemical composition of waste waters has had a negative effect on the removal efficiency of radium, this being the most obvious for the waste water from uranium ore processing. Also, it can be noted that the removal efficiency is not more independent of the columns geometry as for synthetic radium solutions, the decrease of h/d ratio being folowed, generally, by the decrease of removal efficiency.

The detrimental effect of chemical composition of waste waters can be counteract using serial columns, the enhancement of removal efficiency being exactly in the case of processing waste water for which the chemical composition has had the most detrimental effect upon the removal efficiency. Using the same sorbent volume in two columns is more favourable, as for removal efficiency, in comparison to one column.

It can be noted that using the serial column procedure, even at a ratio h/d = 3, the remanent activity concentrations are low, the removal efficiency being over 95% even at specific burdens of 20 BEV/h.

An experiment , using two serial columns with h/d = 3, in which were passed through 2,320 BEV of uranium ore processing waste water, fieled of remanent activity concentration of 0.19 Bq ^{226}Ra/L. Taking into account that in actual Romanian uranium industry waste waters can not be find such high ^{226}Ra activity concentrations, it can be said that in the case of uranium processing effluents it can be achieved an appropriate decontamination using two serial columns (h/d = 3), being able to pass through at least a volume of 2,500 BEV of waste water. These parameters are more assuring for radiometric sorting and mine drainage waters.

Removal of radium from waste waters

Table 1

Column characteristics				Water source					
Number of columns	Sorbent volume (cm^3)	$^h/_d$	Specific burden (BEV/h)	M.D.[1] (3.89Bq/L)		R.S.[2] (4.62Bq/L)		P.P[3] (3.87Bq/L)	
				RAC*	RE**	RAC	RE	RAC	RE
1	10	12	15	0.07	98.2	0.11	97.4	0.39	89.9
			20	0.07	98.2	-	-	0.59	84.7
1	5	6	15	0.14	96.4	0.17	95.7	0.31	92.0
			20	0.17	95.6	-	-	0.39	89.9
1	2.5	3	15	0.36	90.7	0.34	92.0	0.51	86.8
			20	0.34	91.2	-	-	0.76	80.3
2	2x5	6	15	0.06	98.5	0.10	97.6	0.08	97.9
			20	0.05	98.7	-	-	0.12	96.9
2	2x2.5	3	15	0.12	96.9	0.07	98.4	0.12	96.9
			20	0.08	97.9	0.08	98.1	0.16	95.8

1) Mine drainage
2) Radiometric sorting
3) Processing plant

* Remanent activity concentration (Bq/L)
** Removal efficiency (%)

CONCLUSIONS

- Clinoptilolyte zeolitic tuffs have a very high affinity for ^{226}Ra, showing high removal efficiencies;
- The chemical composition of waste water can have a detrimental effect upon the removal efficiency of ^{226}Ra by natural zeolites;
- The inhibitory effect of waste water chemical composition can be counteracted using serial columns;
- Natural zeolite column can be used, if it is appropriate, in more sorption-desorption cycles.

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DOMESTIC SMOKE DETECTORS - A RADIOACTIVE WASTE PROBLEM?

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INTRODUCTION

A common form of domestic smoke detector is the ionisation chamber smoke detector. A small radioactive source provides the ionisation and when smoke enters the sensitive volume the change in voltage is sensed electronically. The use of domestic ionisation chamber smoke detectors is widespread and is recommended by fire authorities and insurance companies. That one million such detectors were imported into New South Wales in 1994 is an indication of the numbers involved. The typical radioactive source in a domestic smoke detector is Am 241, which is regarded as one of the more hazardous radionuclides. If it is such a hazardous material, should it be allowed in the normal household without any control? Or, in a smoke detector, is it in a less hazardous form than assumed in the ICRP recommendations and do its benefits outweigh its possible risks?

In spite of the apparent hazard of Am 241, the possession of the radioactive source in smoke detectors is generally exempt from any form of regulation. Waste regulations, however, set requirements for the disposal of Am 241 and these regulations can be interpreted as applying to smoke detectors. We appear to have a situation where a home owner can legally purchase any number of smoke detectors but when they fail there are Codes of Practice that prevent them being disposed of.

On the other hand, smoke detectors have a direct life saving function and reports indicate that smoke detectors can reduce both injury and property damage by up to 50%. Do these benefits from the use of smoke detectors compensate for any radiological risk?

THE HAZARDS OF AMERICIUM 241

From ICRP Publication 68 (1), the occupational annual limit on intake by inhalation(ALI_{inh}) for an aerosol with an activity median aerodynamic diameter (AMAD) of 1 μm is 500 Bq. The ingestion ALI is 100 kBq. Thus a typical smoke detector with an Americium source of 40 kBq contains about 80 times the ALI_{inh} and about half the ALI_{ing} for an adult worker. Table 1 (based on reference 2) gives the ALIs for inhalation and ingestion for members of the public of different ages. The typical source contains about 2700 times the ALI_{inh} and about 15 times the ALI_{ing} for a one year old.

The majority of Am 241 sources used in domestic smoke detectors are the oxide form. The radionuclide in a gold matrix, sintered at high temperature, is sealed between a silver backing and a thin gold or gold alloy cover, (3). They are robust and can survive quite severe treatment without damage.

Table 1. Am 241 ALIs at different ages.

Exposed person	ALI_{inh} Bq	ALI_{ing} Bq
One year old	15	2.7×10^3
Ten year old	25	4.5×10^3
Adult	26	5×10^3

RADIOLOGICAL ACCIDENTS WITH SMOKE DETECTORS

There do not appear to be any reports of inhalation accidents in the literature.

A case of ingestion by a person who swallowed two Am 241 smoke detector sources of 2.5 μCi (about 100 kBq) each, has been reported (4). The sources were later voided with very little physical damage. The report concluded that the sources lost less than 1 % of their activity and that much less than 1.5 % of the activity released was absorbed into the blood. The authors of the report concluded: "If the sources of Am 241 involved in this incident are representative of those incorporated in domestic smoke detectors, then the most important conclusion that can be drawn is that they are remarkably secure." Current Am 241 smoke detector sources are similar in physical structure, but smaller activities are now used.

CALCULATED RADIATION DOSES FROM THE USE OF SMOKE DETECTORS

External radiation

The OECD (5) has estimated an external dose of 160 nSv from Am 241 to the inhabitants of a protected house.

Inhalation

Table 2. Estimated doses from a fire involving a smoke detector source.

Age	Breathing rate m^3/h	Intake Bq	Dose mSv
1 year	0.35	0.88	0.059
10 years	1.12	2.8	0.11
Adult	3.0	7.5	0.29

For inhalation to occur the source material must become airborne. No inhalation doses will be received under normal operating conditions. The most obvious way for inhalation doses to occur is in a fire. According to the OECD (5) less than 1 % of the Am 241 in the smoke detector source is released in a 925 ^{0} fire. If 0.5% of the material is released and assumed to be distributed in a small room of 20 m^3, then the concentration would be 10 Bq/m^3. ICRP Publication 66 (6) gives an 1 year old child's breathing rate of 0.35 m^3/h during light exercise and a male adult breathing rate of 3 m^3/h during heavy exercise.

Assuming exposure in the small room of no more than a quarter of an hour, the child's intake would be 0.9 Bq and the adults intake would be 8 Bq. Using these values the adult dose would be no more than 0.3 mSv or an child's dose no more than 0.06 mSv. (See Table 2).

If a smoke detector is in a waste stream that goes to an incinerator, there could be a release of the source material if the incinerator temperature is high. Any Am 241 not released will report to the ash and probably go to a municipal tip.

Table 3. Estimated doses from high temperature incineration of a smoke detector source.

Age	Breathing rate m^3/h	Intake Bq	Dose µSv
1 year	0.35	1.75×10^{-4}	1.2×10^{-2}
10 years	1.12	5.6×10^{-4}	2.2×10^{-2}
Adult	3.0	1.5×10^{-3}	5.8×10^{-2}

If 1 % of the source material is released and discharged through a 20 m high stack, and the release occurs over 10 minutes, then the airborne concentration on the ground 50 m downwind would be about 3×10^{-3} Bq/m^3. At that location a child's dose would be 12 nSv and an adult's dose would be 60 nSv. (See Table 3.)

Ingestion

Using the value from Rundo *et al.*(4) for the activity released from an ingested smoke detector source, then the "intake" from a swallowed source would be no more than 400 Bq resulting in an adult dose of 80 µSv or a child"s dose of 150 µSv. A smoke detector disposed of in a municipal tip could leach radioactivity to a local water course. Immersion tests (7) have indicated very slow leach rates. The movement of Am 241, once leached from the source, has been estimated (5) to be 10^{-4} of the ground water velocity. Public radiation doses due to leaching from a tip would be negligible.

Table 4. Estimated doses from use of smoke detectors.

Age	External dose. µSv	Inhalation dose from incinerator. µSv	Accidental inhalation dose from fire. µSv	Accidental ingestion dose. µSv
1 year	0.16	1.2×10^{-2}	59	150
10 years	0.16	2.2×10^{-2}	110	90
Adult	0.16	5.8×10^{-2}	290	80

RISKS FROM THE USE OF SMOKE DETECTORS

Table 4 summarises the possible doses from each pathway discussed above. The once in a lifetime inhalation dose from a fire is the most significant dose. How many people are assumed to be exposed to inhalation of Am 241 in a domestic fire is therefore critical in assessing the overall risk.

THE BENEFITS OF SMOKE DETECTORS

The OECD report (5) indicated that 40 % of fire deaths could be prevented by the use of smoke detectors. Gratz and Hawkins (8) indicated up to 39 % reduction in death and injury and the US National Fire Data Center (9) indicated about 50 %. Similar percentages are reported for reductions in property loss. The OECD report quotes a figure of 18 fire casualties per million population per year, 70 % of which occur in private dwellings. If these data apply to Australia, there are about 13 fire casualties per million per year in private dwellings and about 6 of these could be prevented by the widespread use of smoke detectors.

ALARA

Assuming that each year one adult person in a hundred is involved in a domestic fire inhalation event, one adult person in a thousand is involved in an incinerator fire inhalation event and one 10 year old in a hundred ingests a source, the average annual dose would be about 0.16 μSv. (That is, for the assumptions used, the inhalation and ingestion accident doses have negligible effect on the average dose.)

Using the ICRP (10) probability for stochastic effects of 7.3 % per Sv this would result in 1.2×10^{-2} stochastic effects per million per year. It must be pointed out that natural background radiation is about 2 mSv per year so the calculated increase in annual dose would not result in any observable health effects.

It has been estimated that there are about 13 fire casualties per million per year in private dwellings and that 6 of these could be prevented by the widespread use of smoke detectors. The ratio between probable reduction in fire casualties and possible radiation detriment is 500 to one. This takes no any account of the reduction in property damage.

CONCLUSIONS

The benefits from the use of domestic smoke detectors are at least 500 times greater than the possible harm.

The application of ALARA requires that the radiation detriment be reduced until further reduction is not cost effective. The radiation dose calculated here is so small that any cost to reduce it further would not be cost effective.

If ionisation smoke detectors are the most effective type, and if Am 241 remains the nuclide of choice for domestic smoke detectors then regulators must recognise the fact that the method of construction of the source material makes significant accidental exposure unlikely, the benefits far outweigh the risks, and the exemption that applies to ownership should be extended to the disposal of these sources.

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A STUDY OF BPEO IN THE CONTEXT OF LIQUID RADIOACTIVE WASTE GENERATED AT BNFL SPRINGFIELDS FUEL MANUFACTURING PLANT

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ABSTRACT

Techniques for establishing Best Practicable Environmental Option (BPEO) in relation to the management of radioactive wastes are undergoing development. Recent experience has been gained via a study at BNFL Springfields fuels manufacturing plant involving identification of options for treating liquid waste, screened evaluation and sensitivity analysis to aid selection. Confidence in the process was such that an interim finding was accepted by the regulators and implemented as the BPEO by BNFL ahead of the finalised work on the study.

INTRODUCTION

Radioactive aqueous waste containing alpha and beta/gamma emitters associated with U-238 and Th-232 decay chains are discharged by BNFL Springfields to the tidal river Ribble in Lancashire under joint authorisation of Her Majesty's Inspectorate of Pollution (HMIP) and the Ministry of Agriculture, Fisheries and Food (MAFF). The dose impact of the discharges amounted to around 0.02 to 0.03 mSv to the critical group during the early 1990s (1). During a review of the authorisation in 1991 a number of options were identified as having potential for reducing further the impact of the discharges and in order to evaluate such options, HMIP and MAFF placed a requirement on BNFL to undertake a BPEO study. This paper describes the methodologies used, the potential difficulties encountered and how they were addressed.

METHODOLOGY

The Royal Commission on Environmental Pollution (2) have provided a conceptual approach for conducting a BPEO study. This approach formed the framework within which the study at Springfields (3) was developed and in essence consisted of option generation, screening, evaluation by multiattribute analysis, initial derivation of the BPEO and confirmation by sensitivity analysis.

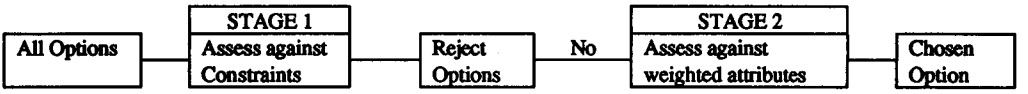
Identification of Treatment Options

Prior to considering specific options, it was considered essential that a full understanding of the waste streams should be obtained, including identification of all sources of contaminants, physical and chemical properties and waste volumes. A fully open minded approach was maintained towards identification of possible treatment options. These ranged from the obvious to the novel including the existing arrangements; options were grouped into common themes, namely (i) limiting impurities in feed materials, (ii) physico/chemical treatment, (iii) hold up to allow radioactive decay and (iv) dispersal by means of discharge. Within these groupings, a total of twenty-one options were arrived at with some options being earmarked as potentially suitable for combining.

Evaluation

In view of the number of options, a two staged approach was adopted in which low ranking options were first eliminated by screening, see Figure 1. This provided an opportunity to target the more potentially fruitful options with more detailed development and costing.

Figure 1. Schematic of Springfield BPEO Methodology



Stage 1 - Screening using Constraints

All options were first assessed against a set of constraints covering statutory obligations, technical feasibility and availability, and financial aspects. It was imperative that implementation of the BPEO did not infringe any legal requirements. Judgements on technical availability were made on the basis that the technique or process had to be accessible, proven or capable of being developed and installed within a reasonable timescale. A six year time limit was used based on a judgement that potentially effective options would not be eliminated due to the development time needed. Costs were an important factor and to avoid bias, all options were sufficiently developed so that they could be compared on an equivalent basis. The detrimental costs following discharges were taken into account and the base line figure recommended by NRPB (4) of £20,000 per man Sv was used together with other health, safety and public concern factors to identify a reasonable level of expenditure. Financial considerations were responsible for the rejection of 52% of the options in the Springfields study compared to 28% for technical constraints.

Stage 2 - Performance against Attributes

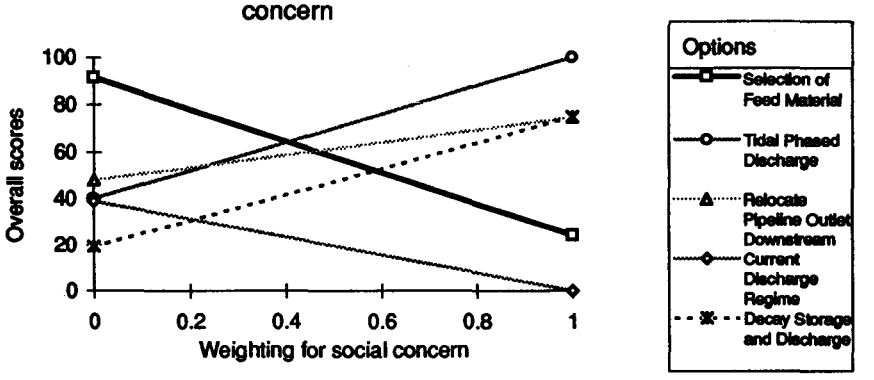
This stage was designed to assess the options which remained after screening using multi-attribute analysis. Attributes were chosen *inter alia* from a review of the constraints. Seven attributes covering quantitative and qualitative factors were arrived at, namely collective dose, skin dose and committed effective dose equivalent to the critical group, timescale, cost, non radiological environmental impact and social concern. For each option in turn, a score was derived according to its performance against each attribute. Linear functions were set by assigning the best performing option (or options in the event of a tie) a score of 100 and the worst performer a zero score. Alternatives to linear relationships were difficult to justify in the present case. A range of weights was chosen for the attributes to take account of their relative importance and the scores were weighted and summed. The highest score was indicative of the likely BPEO.

Assessing relative weights for each attribute is arguably the most difficult and potentially controversial part of the BPEO process. A balanced view can be obtained through consultation between operator and regulators who took account of local and environmental concerns. The outcome of consultation for the Springfields study was that dose was the most important factor with a average relative weighting of 0.4 (85% CEDE, 10% collective and 5% skin), cost was next with weighting of 0.25 and the remaining 0.35 shared roughly equally between all other attributes.

Sensitivity Analysis

The sensitivity of the BPEO to changes in input parameters and assumptions was examined to ascertain that it was reasonably sound. The choice of these parameters depends on how the study was conducted. In this case the weights assigned to each attribute were less well established than the input data for each option and hence the better target for sensitivity analysis. The weights were adjusted by a factor of 4; this accommodated around 90% of the suggested weightings from the consultation exercise and hence was considered to be a reasonable range. This variation did not alter the BPEO arrived at; this was an option under (i) above entailing restricting impurities in the process material. An illustration of this is given in Figure 2 which relates to the attribute social concern and the options remaining after Stage 1 screening. The sensitivity work also indicated that combining the BPEO with the option to change the operating regime in connection with dispersal of radionuclides on discharge could enhance the overall benefit and hence the latter option was earmarked for further evaluation.

Figure 2. Overall scores in relation to weighting for social concern



ISSUES ARISING

A number of difficult issues arose during the course of this study some of which have been already indicated. Some additional issues are highlighted below to assist in future studies.

- A balance may need to be considered between radiological benefits and potential worsening in environmental impact from discharge of chemical species.
- A balance between public and operator doses may have to be addressed.
- Implications for commercial viability and related employment factors may need to be addressed in admitting costly treatment options.
- Public perception stemming from export of waste and hence risks to a different population may have to be addressed.
- There may be a tendency as a result of public or political expectations to implement the best environmental option rather than the BPEO.

CONCLUSION

An open, structured and well documented approach towards BPEO studies with appropriate consultation and demonstration by sensitivity analysis is likely to produce a result which is more universally acceptable. Support for this can be drawn from the present study.

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GUIDELINES ON THE HANDLING OF RADIOACTIVE MATERIAL AND WASTE IN HOSPITAL AND BIOLOGICAL RESEARCH CENTRES.

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This document sets out to provide a guide to the safe handling of radioactive waste produced in medicine and biological research.

Naturally in order to know how to handle radioactive waste properly we must first establish the exact nature of such waste and how it has been produced.

This Guide is basically divided in three parts. The first and main part establishes the correct procedures to be followed in the generation, distribution, manipulation and disposal of radioactive materials.

The second part, which appears in the form of an appendix, offers the norms for protection from radioactivity in the workplace, and refers to methods of decontamination and criteria established in the design of installations as well as appropriate training for employees working directly or indirectly with radioactive materials.

The third and final part describes the process by which the relevant authorities are involved in the disposal and declassification of radioactive waste.

The Guide draws on the experience of professionals working in radiological protection in medicine and biomedical research, and serves to homogenize methods of handling radioactive waste. Furthermore, it offers a new method of characterizing such waste.

In order to manipulate radioactive material, we present operational radiological protection rules, detailing a method to evaluate waste generated in radioisotopes application techniques, based on an analysis of the material used in each technique, its ways of manipulation and quantifying some of the resultant waste. So, by this kind of analysis for each technique, it will be possible to perform a characterization of the resultant products and also a valuation of the total volume produced in the centres.

The radioactive waste management is subdivided in waste that can be declassified directly or after a decay process, and waste that cant be declassified. The latter ones must be managed by an authorized enterprise.

To manage the declassified waste, we propose to carry out an authorization request, to make the pouring by conventional means, to the appropriate organism by comparison between the waste activities and the published international regulations, until the national regulatory organism publishes its own declassified regulations.

We also provide a procedure to control the decay of the waste that must be kept in temporary storage for a period of time until they will be under the limits to perform its management by conventional means.

We establish a control procedure before evacuation to let the user assure that the wastes are under the limits, have the expected destiny and are in accordance with the authorization request.

As in the case of wastes to evacuate by conventional means, we expose a procedure to control wastes before the evacuation by the authorized enterprise.

IRPA9
1966 International Congress on Radiation Protection
April 14-19, 1966
Vienna, Austria

FOR SUBMISSION OF ABSTRACTS
(Instructions for preparation on reverse)

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PAPER TITLE **LONG LIVED, LOW LEVEL RADWASE FORECAST IN SAUDI ARABIA**

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MAJOR SCIENTIFIC TOPIC NUMBER (see page 7) Low Level Radioactive Waste
ABSTRACT (See instructions overleaf)

LONG LIVED, LOW LEVEL RADWASE FORECAST IN SAUDI ARABIA

Samir Abdul-Majid, Ibrahim Kutbi and Abdullah Al-Marshad

Most of the open type radwastes are generated in medical and educational centers in Saudi Arabia; their total numbers are about 15 and 5 respectively, while sealed spent sources are coming from industry. The volume of long lived radwaste was estimated up to the year 2020 for the proper, design of interim storage and radwaste treatments and conditioning facilities purposes. It was anticipated that two main parameters affect the increase in radwaste in the future. The first is the increase of radionuclides use in hospitals in diagnostisis and therapy in the country. The extrapolation of the increase of ^{99m}Tc-^{99}Mo generators over the last few years was used as a growth indicator for this parameter. The second is the increase in population which should be associated with increase in medical services in general. The population is expected to increase about 1.8 times in the year 2020. The annual, long lived waste, that need treatment, conditioning and storage as a function of time is expected to follow the relation, $V = 10 + 0.048t^2$, where V is the waste volume in m^3 and t is the time in years after 1995. The radwaste production rate is expected to be 4 time its current value in 2020. The expected long lived cumulative treated, conditioned and liquid wastes, in that year, if not subject to volume reduction, in m^3 will be: 500, 75 and 100 respectively. Comparisons were made with IAEP waste volume expectations for countries of similar conditions; the results were in reasonable agreement.

Environmental impact of a Very Low Level Waste specific landfill

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I - INTRODUCTION

Operating enrichment plants, nuclear power plants and reprocessing plants and the decommissioning of nuclear facilities will give rise to large volumes of waste material (concrete, steel and others metals, technological wastes heat insulators...) and most of them, in term of quantities, will be categorized as Very Low Level Wastes.

This paper deals with the environmental impact of a specific landfill as a final destination for the Very Low Level Radioactive Waste (VLLW) with the aim of providing technical elements for safer workers practices during the operational and the monitoring phases and for a public occupation after closure of the site. This study has been made on the basis of inventories in terms of estimated quantities and spectra of the French VLLW for a set of scenarios which are representative of practices in a landfill.

II - POSITION OF THE FRENCH AUTHORITIES

For very low level wastes, it will be necessary in a near future to define practical rules to achieve a coherent and standardized management of VLLW.

The nuclear regulatory agency DSIN proposes to establish controled disposal channels with a total treacability of all waste produced. In particular, this process will lead to the elaboration of studies on VLLW, the definition of "zoning" in nuclear installations to identify areas having a potentiality to generate radioactive wastes, the definition of adapted channels based on impact studies, for each kind of radioactive waste and the creation of VLLW specific landfills with accurate regulatory controls.

A concrete and approved solution is not finalized, but a specific landfill, concept between an industrial landfill and a low-and-medium level waste repository, seems to be a realistic solution.

III - ORIGIN AND POTENTIAL QUANTITIES OF VLLW

Low specific activity materials can arise during most maintenance activities carried out in nuclear fuel cycle facilities, however the largest potential source of such materials is from the decommissioning of nuclear facilities, especially the commercial power reactors.

Origins of the Very Low level wastes are multiple and theirs specification also. They result especially from :

- operating of the nuclear facilities, enrichment plants, nuclear power plants and reprocessing plants,
- the dismantling of the same nuclear facilities from 1970 to 2090,
- the radioactive wastes from the "small producers" , hospitals and universities,
- the uranium mine and mill tailings wastes,
- the wastes issued from fabrication processes in the classical industry but using radioactive materials.

The three last categories have not been taken into account in this study for the following reasons :

- the radioactive wastes from the "small producers" are incinerated,
- the uranium mine and mill tailings wastes are disposed of at the extraction site because of very large volume and specific nature of the radionuclides,
- the wastes originated from fabrication processes in the classical industry but using radioactive materials are submitted to special treatment case by case.

We consider that in 2050, all French gas-graphite reactors will be dismantled, as well as uranium enrichment and reprocessing plants operating at present time. We consider also that the stage 2 of PWR dismantling will be over. Depending on the delay of 40 years taken into account between the stage 2 of decommissioning and the stage 3 (final phase) for PWR power plants, the VLLW production will drop down for 25 years between 2050 and 2075.

IV - DATA AND PARAMETERS OF THE IMPACT STUDY

IV 1 Source term

The estimated VLLW volumes are 60 000 m^3 for 1994-2015, 128 000 m^3 for 2016-2030 and about 60 000 m^3 for the last period, 2031-2049. The total volume of waste would be about of 250 000 m^3 for a total weight of 866 000 tons.

The drums are put in trenches dug below the original ground level and covered with the concrete and rubble wastes. The dimensions of the landfill are 350 m length, 100 m width and 7 m depth without the ground cover.The mean activity will be of 10 Bq/g with a mean spectrum depending of the origin of the waste.

The main radionuclides likely to be found in VLLW are H3, Mn54, Co58, Co60, Ni63, Sr90, Cs137, U234, U238 and α-total. The spectra are mean spectra, weighted by the potential quantities of the four main materials: concrete and rubbles, metals (steel, copper, aluminium...), technological wastes and heat insulators.

The release of activity from the source to the geosphere is modelled by the Annual Leaching Fraction (A.L.F) for each radionuclide.

IV.2 Description of the geosphere

Calculations have been made for a generic site, the hydrogeological characteristics being quite representative of a surface site disposal [1]. The main outlet is a river with a flow rate of 1.10 m^3/y located at 250m of the nearest landfill boundary. The medium is clayey-argileous, equivalent to a porous medium. The radionuclide transport is slowed up by the sorption of species in the medium and modelled with a distribution coefficient Kd [1].

IV.3 Description of the biosphere

The river water is used for irrigation and for drinking (man and cattle). A conservative diet [1] is defined with meat, fish, vegetables and drinking water comsumption; all these products are coming from the site (autosuffisance). The biosphere data, transfer factors in the differents boxes of the biosphere are taken from litterature and International exercices [2] [3].

IV.4 Exposure pathways

The workers involved in the operating processes will inhale and ingest dust or soil deposited on their hands and face, and will be submit to external exposure from the waste scraps. We considered a driver of handling truck and a chemist who makes measures on small samples of wastes.

We considered that thirty years after closure of the landfill, an individual would live on the site in a house with a garden around (residence scenario). This individual could be an adult, a 10 years old child or an 1 year old baby. The scenario takes into account all the possible pathway exposures.

We have also considered the use of contaminated water, during the operating and post-closure phases, for the irrigation of crops, for fishing areas, by consumption of contaminated drinking water and by consumption of meat from cattle eating pastures and drinking water at the discharge point.

V - IMPACT RESULTS

V.1 Methodology of dose calculations.

The methodology for estimating the individual dose resulting from the exposure pathways described above, is to calculate the impact of a set of scenarios representative of practices using the best-estimate values for related parameters (time of exposure in the practice, inhalation rate, concentration of particles, ingested quantity of dust, human diets,...).

Calculations of the doses during the operational phase of the landfill and for the residence building after closure, are made with the computer code CERISE [4] which determines for various individual exposure situations, the relationship between the effective dose and the radioactivity of a material handled by or located near the person considered, or his radioactivity intake.

Calculations of the radiological consequences from the landfill activity flow rate to the river outlet uses the geosphere GEOS code [5] which estimates the activity flow to an outlet, river or well, coupled to the biosphere ABRICOT code [6].

V.2 Results of dose calculations.

The maximum annual dose occurs in the landfill aera during thethird operational phase where the annual flux of waste is maximum. The total massic dose is of $3.7.10^{-1}$mSv/y for the manual worker, for all radionuclides and all practices taken into account. 65% of this dose is due to Co60 ($2.4\ 10^{-1}$ mSv/y).

For the public scenarios, the maximum dose due to the residence scenario is $2.4\ 10^{-2}$ mSv/y for the adult and 3.10^{-2} mSv/y for the ten year old child. This dose is due for 30% to Pu239 and for 20% to Cs137. The residence has been considered to be built 30 years after the closure of the landfill and the short-lived radionuclides have strongly decreased.

For the river scenario, between 1994 to 2094, the maximum dose of $6.5\ 10^{-4}$ mSv/y is due to Tritium and the maximum occurs very early (about 2040) because of the absence of sorption in the geosphere for this radionuclide. This dose is essentially due to the consumption of vegetable products (~100%), especially cereals (62%) and fruits (21%).

The absolute maximum dose of 5.10^{-3} mSv/y occurs about 6500 years after closure of the landfill and is due in equal proportion to the uranium isotopes U234 and U238. The main pathway is vegetables consumption for 90%, water drinking for 5% and 1.5% for external exposure.

VI - CONCLUSIONS

The deterministic calculations of individual dose were performed for a low isolation capacity. It should be mentioned that it is possible to add engineered barriers to improve the safety of these disposal facilities :

- barriers to make the intrusion of humans, animals or plants more difficult. These include : greater thicknesses of earth cover and reinforced concrete ;
- barriers to prevent the ingress of grounwater, surface water or precipitation. These include : clay covers, synthetic impermable barriers, draining layers and hydraulic bypasses around the facility ;
- barriers to prevent the release of radionuclides from the disposal. This include : impermeable clay or man-made barriers and buffers which let water out but retard radionuclide migration.

Besides the possibility to improve the landfill safety by the implementation of such technical solutions, the generic impact assessment performed shows that the effective doses undergone by persons of the public are well below 1mSv whatever operational or post-closure phase is considered. The highest calculated exposure concerns mainly engine drivers submitted to external irradiation. It should however be stressed that in this case, the range of doses undergone by workers are again low (about one order of magnitude below 1mSv, and mainly due to Co60). Since exposures are only relevent to ponctual operations, they can easily be reduced if necessary by additional temporary protection (such as radiation protected engins or temporary shields above wastes).

According to former results, it is therefore foreseeable that VLLW are likely to be disposed of safety in specific landfills of very simple conceptual design (such as for example the back filling of quarries). Some favorable site characteristics should be looked for but since the radiological impact of the landfill is low, the site selection criteria may not be stringent.

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TRANSFER PROCESS STUDIES THROUGH A TOARCIAN ARGILITE

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1. INTRODUCTION

Concerning long lived and high level radioactive wastes, the French Wastes Management Research Act (30th December 1991) has set three ways to be studied : separation and/or transmutation, conditioning processes and volumes reduction, and feasibility of geological disposal. ANDRA, in charge of this last point; has the responsibility to create and develop Underground Laboratories, to elaborate and propose the concept of the future disposal installations with safety demonstration, to be submitted to the French safety governmental authorities (Direction de la Sûreté des Installations Nucléaires). In order to asses these safety demonstrations, to build up its judgement, DSIN will ask technical advice, from Institut de Protection et de Sûreté Nucléaire (IPSN).

To be able to meet these requirements, IPSN is developing, in the framework of its Research and Development safety programs, in situ research concerning the confining properties of geological formations. In order to perform experiments in representative conditions concerning mainly geotechnical and hydrogeological properties of very low permeability rocks, appropriate underground sites have been selected in the past in granite and shale formations. IPSN activities at the Tournemire site are the most important activities regarding specifically research about characterisation of transfers through an argilite formation. This programme is developed in the framework of a cost sharing research contract between IPSN and EC.

2. OBJECTIVES OF THE IPSN TOURNEMIRE TUNNEL PROJECT

2.1 CHARACTERISTICS OF THE TOURNEMIRE SITE

By the end of 1990, IPSN initiated in situ research concerning argillaceous formations, considering the relative lack of knowledge about the behaviour of deep impervious clays. IPSN is studying a geological representative formation constituted by indurated claystones of Toarcian. This site has been selected because of its geological simplicity and also because a former railway tunnel gives access to the centre of the toarcian formation [1]. This is an hundred years old railway tunnel, 1885 meters long, in the close vicinity of the village of TOURNEMIRE (Aveyron), in the South of France, crossing a 200 m thick toarcian clay formation. The overlying limestones are 250 meters thick, so the geotechnical and hydrogeological conditions can be considered as representative of those of a deep repository.

The geological formations of the site are constituted by sub-tabular sedimentary layers of the lower and middle Jurassic Period. The stratigraphic formations involved in the Tournemire environment are as follows : Hettangian and Carixian (limestone, dolomites and interbedded marls), Domerian and Toarcian(245 m of marls and argilites), Aalenian, Bajocian and Bathonian (250 m of massive limestone and dolomites).

The prevailing lower limestone formations of Hettangian and Carixian contain the general lower regional aquifer. At the place of the tunnel, this aquifer is in charge below the impervious domerian and toarcian clays. The limestone layers from Aalenian up to upper Bathonian are the reservoir for the upper regional aquifer : at Tournemire, this aquifer lies in the lower part of the Aalenian and should feed the toarcian clay just below. Taking into account the hydraulic heads of the two aquifers, the fluid transfer through the toarcian argilites, which could behave hydraulically as a semi-permeable medium, is supposed to be vertical downwards.

2.2 WORK PERFORMED

The research program concerns hydrogeological properties in order to determine the characteristics of fluid transfers through the clay formation and it intends to perform modelling of these fluids transfers.

The Tournemire programme includes general geological and hydrogeological investigations, drilling operations for detailed geological studies (information about the depth, lithology, stratigraphy and hydrology), including structural studies relative to paleohydrogeological understanding, development of in situ tests including permeability tests at different level (mainly pulse tests), and boreholes equipment for interstitial water pressure measurement in low permeability rocks, sampling of material for laboratory studies on cores including petrophysical conventional identifications, diffusive characteristics, development of isotopic studies of the interstitial fluid and solid phase (matrix and fractures).

During a first phase, six boreholes have been drilled from the tunnel to the bottom and to the top of the Toarcian formation [1]. From the results of this phase, the programme has been developed thanks to long time hydraulic testings into boreholes specifically conceived for rocks of very low permeability, detailed fundamental studies relative to the possibilities for fluid transfer : diffusive transfer, different fluid potentials and associated

fluxes measurements (chemical, suction...), and a natural isotopes measurement programme on the whole formation surrounding the tunnel aiming to characterise transfers both at the geological time scale and at the time scale corresponding to the perturbation due to the presence of the tunnel itself.

As these developments suppose a good understanding of the structural characteristics of the rock, and the achievement of a very high quality sampling phase of the formation, the on going work is based on a coring campaign of height boreholes around the tunnel [2].

3. MAIN RESULTS

3.1 PETROPHYSIC ROCK CHARACTERISTICS

The argilites appear to be very thinly bedded rocks mainly composed of clay minerals (40-50%) such as mica, kaolinite, illite and interstratified clay mineral close to illite, with about 10 to 40% of carbonates (dolomite, siderite and mainly calcite), 10 to 15% of quartz, and a few amount of chlorite, and pyrite.

The water content is very low : 1 to 3%. The free total access porosity measured with mercury is about 3 to 4%. The determination of the pore volume and the distribution size of these pores thanks to the B.E.T. method has shown that the dominant pore size family was centred on the value of 0.0028 µm corresponding to a total porous volume between 0.0071 and 0.0198 cm^3/g. The mean specific surface is about 10 to 25 m^2/g.

Hydraulic conductivity from laboratory tests with water is between 10^{-13} m/s and 10^{-14} m/s. The very low permeability, and the very small dimensions of the porous space lead to conclude that the fluid transfers in this kind of rock are not governed by convective processes, but by processes such as diffusion or others which have to be recognised and quantified.

Mass transfer studies of fresh tritiated water through argilite sample allowed both the characterisation of diffusive transfers, and those (permeation) under an hydraulic head (15 bars). Diffusion coefficient value is estimated around 3.5 10^{-12} m^2/s (with a total porosity between 8.5 to 10.2 %), and permeation coefficient mean value K_p is about 2.5. 10^{-14} m/s, meaning an average permeability coefficient K_H of 9.5. 10^{-15} m/s. These results seem to confirm that water or humidity transfers are essentially diffusive.

3.2 IN SITU HYDRAULIC CHARACTERISTICS

Many in situ permeability tests at different levels have been performed (mainly pulse tests) as well as geophysical loggings. The in situ permeability of the toarcian and domerian clay formations is very low, between 10^{-11} and 10^{-13} m/s [1]. The permeability of the calcareous carixian and aalenian formations is substantially higher : the Carixian has permeability of about 10^{-8} m/s and the Aalenian of about 10^{-6} m/s.

The aquifer heads, in the carixian and in the aalenian limestone correspond with the values that can be deduced from the hydrogeological regional and local system. But it has not been possible up to now to determine the actual head in the argilite formations because of the very long period to reach the steady state after the perturbation induced by drillings resulting from the very low permeability of this kind of rock.

3.3 CHEMICAL AND ISOTOPIC STUDIES FROM INTERSTITIAL PORE WATER

Isotopic and chemical contents of the interstitial fluids from the Tournemire argilites reveal an high heterogeneity of the fluids into the formation [3]. Due to the very low water content (1 to 3% wt) of this rock and to its high degree of induration. it has been impossible to use the squeezing technique for the extraction of total interstitial water. Thus, attempts to evaluate water chemistry have been made by leaching techniques, but interstitial water was exclusively recovered by distillation under vacuum to measure its stable isotope contents (^{18}O et ^{2}H).

Two fluid families, with different chemical facies have been identified (from absolute ethanol leaching) : (i) one $CaSO_4$-Mg type of fluid, representative of both aalenian limestone and high $CaCO_3$ content levels from the upper toarcian, and (ii) fluids with cationic composition dominated by Na coming from the argilites. These changes in the chemical facies probably result from cationic exchange between fluid and the solid matrix.

Four types of pore water can be distinguished from their stable isotope contents (^{18}O et ^{2}H) : (i) meteoric water which is isotopically similar to that of the overlying aalenian karstic aquifer, (ii) meteoric water whose isotopes content ise modified, probably due to exchange with host rock, (iii) water enriched in heavy isotopes possibly due to an evaporation process close to the tunnel and, (iv) water rich in heavy isotopes which could contain a part of residual connate water. The water types (i), (ii) and (iv) show a "randomly" distribution with depth which could probably be controlled by variations in the accessibility to rock porosity. This point has to be clarified, but the direct influence of both water content and $CaCO_3$ of the rock on these two stable isotopes contents has been clearly identified.

3.4 ISOTOPIC STUDIES FROM CALCITE FILLING IN FRACTURES

In addition to the study on pore water, isotopic measurements were made on the secondary calcite that filled the fractures intersected by the boreholes [1] and [2]. Variations of the stable isotopes content (^{13}C and ^{18}O) of these calcites versus depth appear more « regular » and homogeneous than those observed with pore fluid but the measured values imply that several different types of water have circulated in the fractures. An hypothesis to explain the distributions of ^{13}C and Iron contents with depth in fracture minerals would involve both upward and downward circulations, but it has to be confirmed.

All the secondary calcites are ^{14}C free. Their $^{234}U/^{238}U$ ratio are close to one, but they mostly reflect the contribution of a detrital fraction and cannot be used to estimate the age of the calcite. However, the very high $^{230}Th/^{234}U$ ratio measured from the bottom of the profile, probably indicates a secondary sink of ^{234}U. This suggests that water circulation should have occurred in the fractures at the lower part of the toarcian formation, 600 000 years ago at least.

At this stage of the study, it is impossible to establish any link between interstitial fluids and the « paleofluids » identified from the fractures. It could be possible that interstitial fluids reflect a very long and close interaction with the rock itself; and that paleofluids from fractures could represent fluids with very limited isotopic and chemical interactions with the matrix.

4. MAIN CONCLUSIONS

Some time scale indications can be deduced from the measurements on stable and radioactive isotopes both from the liquid and solid phases of the clay :

- after about one century, a distance of 20 à 30 m around the tunnel seems affected by rock desaturation as it can be seen from heavy isotopes values in the interstitial water;
- the existence of fluid which could include connate water indicates very long residence time into the formation. But these old waters seem to remain in coexistence (at some decimetres of distance) with quasi natural meteoric water;
- ^{14}C and U/Th values from fracture minerals lead to think that the clay formation has not been affected by important circulation for more than several thousand years, but it is possible that solutions percolated through a certain number of fractures less than 600 000 years ago.

On the basis of the available data (aquifers heads and permeabilities of the argilites), fluid circulation simulations through the clay were performed [1]:

- results confirm that taking into account only convective transfer, the transient period to reach the steady state after a significant perturbation is very long (several years), and the value of the interstitial water head at different levels in the Toarcian is close to the value of the topographic level at this point. Combined with the very low permeability of the rock, this means that the flow discharge rate by drainage that can be anticipated, specially at the tunnel level, is very low (below 10 cm^3/h) and very difficult to estimate from in situ tests;
- taking into account only transfers thanks to pure diffusion, calculations indicate that the formation could be leached by meteoric water in less than 2 Ma using an effective molecular diffusion coefficient of deuterium close to the one of water into water, or could not be leached after 130 Ma, using a coefficient corrected from porosity and tortuosity. This proves that such simulation would need a better knowledge of the diffusion coefficient of water through clays.

As a result of the coring campaign corresponding to a circular plan (40 m of diameter) perpendicular to the tunnel axis, the argilite formation in this zone appears to be characterised by the existence of two areas structurally different : a west zone relatively highly fractured, and an east zone without any fracture [2]. In some cases, one can observe fault zones, not only infilled with calcite, but mainly constituted of highly disturbed clay material of less good properties (physical and mechanical). The presence and the importance of such structures into argilite formation should be of great interest regarding fluid transfer possibilities.

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PRACTICAL DIFFICULTIES ASSOCIATED WITH INVESTIGATION AND SUBSEQUENT REMEDIATION OF CONTAMINATED LAND

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ABSTRACT

This paper reports on the characterisation and subsequent development of a remediation strategy for a seven hectare site in Oxfordshire, England which was used for the burial of waste contaminated with radioactivity and a range of organic and inorganic chemicals.

The method of site assessment is described and any lessons learnt are highlighted. In particular, the shortcomings of various intrusive and non-intrusive site investigative techniques are discussed.

Anyone faced with the problem of developing a site assessment methodology for contaminated land will find this paper particularly useful.

SITE ASSESSMENT TECHNIQUES

In the case under discussion, the purpose of the site assessment was to determine the extent and location of buried contamination within the site, thus allowing identification of an appropriate remediation strategy. The Site Assessment was divided into three main stages:

- desk study (including a preliminary site visit) to identify past usages which may have given rise to contamination and any particular structures or areas where contaminative processes may have been carried out;
- physical investigation of the site using intrusive and non-intrusive techniques to detect and quantify the extent of any contamination;
- data interpretation in order to determine the extent and magnitude of contamination and therefore the likely disposal route for any waste arisings.

DESK STUDY

The desk study is a cost effective way of investigating whether a site has been put to a potentially contaminative use. Typical information sources include:

- site records, e.g. drawings, maps, investigation data;
- company records, e.g. archival information;
- plant personnel, including former employees;
- local literature, e.g. newspapers, local library;
- regulatory bodies, e.g. local councils, waste regulatory bodies, Her Majesty's Inspectorate of Pollution, National Rivers Authority, Health & Safety Executive.

Company records and recollections of former employees indicated that the site had been used as a repository for redundant items of equipment lightly contaminated with radioactive material. In addition, chemically toxic waste, some of which had traces of radioactive contamination, had been buried in unlined, earth-covered pits during the 1950s and 60s. However, there were no accurate records of the nature and quantity of waste deposited in the pits as there was no legislative requirement to record this information at the time of disposal. Recollections of existing and former employees are particularly important in such situations and proved invaluable in this case.

Records confirmed that a preliminary radiological clearance of the site had already been carried out involving the following key tasks:

- decontamination and demolition of buildings;

- decontamination of roadways within the site;
- examination and removal of drains;
- removal of a low level radioactive waste disposal pit;
- trial pitting to confirm approximate locations and contents of the chemical and beryllium burial pits;
- post-clearance contamination and radiation surveys.

However, no attempt had been made to remediate the chemical and beryllium burial pits.

SITE INVESTIGATION

The site investigation forms the major database for the assessment of the degree of contamination and the extent of remediation required. The choice of sampling strategy will depend upon the findings of the desk study and guidance for optimisation of site investigations is given in Reference 1.

1993 Characterisation Programme

The first stage of the site investigation was to confirm the conclusions of the preliminary radiological clearance study, namely that the site is very unlikely to contain any significant burials of chemical or radioactive waste other than those previously identified within the burial pits. This was done via:

- an extensive soil sampling and analysis programme;
- a walkover radiation dose-rate and contamination survey of the entire site;
- general site inspection.

The sampling programme was agreed with the local council prior to implementation and significantly exceeded the sampling density recommended in DD 175 (Ref 1). The main component of the programme was extraction of cores from over 120 locations using a regular grid sampling pattern. A further 70 cores were also sampled where contamination was judged most likely given the existing knowledge of the site. No core samples were taken from the burial pits.

None of the soil cores analysed contained any significant quantities of hazardous material and it was therefore concluded that the site is highly unlikely to contain any disposal sites or areas of contaminated land additional to those previously identified. This was supported by the walkover dose-rate and contamination survey, the findings of the desk study and site reconnaissance which did not indicate the presence of further burial pits.

However, the sampling programme highlighted a number of practical difficulties which may not be obvious to first-time site investigators. Firstly, the importance of ensuring that samples are adequately labelled and logged should not be underestimated. All samples should be clearly marked in indelible ink with the location number and date and time of collection. This allows greater traceability of records and may avoid significant time delays in the future. In addition, the recovered core samples are not discrete and, if particularly accurate data are required, great care should be exercised to ensure that contamination is not carried from one stratum to another. It is also essential to provide analytical laboratories with samples of sufficient size to allow division into representative sub-samples for analysis.

It should be noted that a 'herringbone' sampling regime has since been promoted by some workers in the field in preference to a rectangular grid. This involves the use of four interlocking regular grids and has been shown to be more statistically significant than stratified regular sampling.

Options for Long Term Management of the Hazardous Inventory

An Environmental Impact Assessment of the site identified two potential long term management strategies for the burial pits consistent with modern standards for safety and environmental protection.

- isolation of the pit contents from the environment by the construction of an engineered cap and curtain walls over and around each burial pit;
- removal of the contents of each burial pit and subsequent disposal via an appropriate route.

Either of these options, or a suitable combination of both, would ensure that health risks to members of the public are acceptable and that the environment is adequately protected from hazardous contaminants known to be present at the site.

Trial Excavation of Burial Pit

A trial excavation of one of the chemical burial pits was undertaken in March 1994 in order to demonstrate the practicability of removing the contents of each burial pit. The pit chosen was believed to be relatively free of radioactive contamination and would therefore pose less risk to workers and members of the public.

The trial excavation was performed in accordance with a fully approved safety case and used a combination of manual and mechanical techniques. The pit was found to be significantly larger than indicated by earlier trial pitting investigations and also contained traces of radioactive contamination (i.e. > 0.4 Bq/g). In addition, the degree of chemical contamination was found to be greater than expected and concentrations were generally above typical levels measured in uncontaminated soil.

It was therefore concluded that existing data on the likely size and contents of the burial pits was likely to be inaccurate and that further site investigations should be performed.

Geophysical Survey and Core Sampling of Burial Pits

A geophysical survey of the site was performed and this confirmed the number and approximate locations of the chemical burial pits. The advantage of such techniques is that they are relatively quick and can be used for initial appraisals. However, care must be taken in the choice of technique and the spacing of traverses. It must also be borne in mind that the techniques identify geophysical anomalies which may not correspond to identifiable features upon intrusive investigation.

Once the geophysical survey had confirmed the approximate locations of burial pits, a core sampling programme of the pits was performed in order to provide further information on the nature and quantity of buried contaminants. The results of this programme are not yet available but will be carefully considered when a decision is made as to which long term management strategy should be implemented (i.e. capping or removal). Some general points on data interpretation techniques are given in the following section. It should be noted that the beryllium pits are being characterised as part of a separate DRAWMOPS project.

DATA INTERPRETATION

Once analytical results are available, they must be examined in order to determine the magnitude of any contamination problem. There are a number of documents which can be used in the interpretation of chemical analysis results. For example, the Interdepartmental Committee on the Redevelopment of Contaminated Land (ICRCL) have produced a number of guidance notes relating to the re-use of sites with previous contaminative uses. These are particularly useful when setting action levels above which further more detailed analysis will be performed. In the case of radioactivity, the need to show that soil is below the Radioactive Substances Act 1993 exemption level of 0.4 Bq/g is particularly important although this can be problematic given the level of natural radioactivity typically present in soil.

ACKNOWLEDGEMENTS

This work has been funded under the DRAWMOPS programme of the UK Department of Trade and Industry (DTI). The results of this work form part of the UK Government programme on decommissioning and radioactive waste management, but do not necessarily represent Government policy.

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URANIUM MINING AND MILLING BY COGEMA
ENVIRONMENTAL IMPACT COMPARED TO 1 mSv LIMIT

S. BERNHARD [1], JL. DAROUSSIN [2], JP. PFIFFELMANN [2]

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CEA then COGEMA have been been operating mines and mills in France since 1948. Total production nears 70000 t of U in the concentrate which were contained in some 85 millions tons of ores (pulp and heap leaching). Many sites are now undergoing remediation [1] and impact on the environment has always been a great concern.

RADIOLOGICAL MONITORING

Throughout the life of the operations, monitoring goes on using a complete network on site and in the neighbourhood. This includes geotechnical monitoring (to ascertain the stability and good confinement of the waste), radiological and chemical monitoring. The radiological monitoring implies :

- water sampling up and down stream (Ra, U analyses),
- integrated site dosimeter measuring Potential Alpha Energy from radon and alpha activity of airborne dust particles,
- thermoluminescent dosimeter for external irradiation measurements,
- periodic analyses in the food chain.

For the evaluation of the impact we need :

- the annual radiological measurements of the main pathways during and after remediation,
- similar values for the natural background (a method is described in [2]),
- the complete description of the critical group,
- in order to calculate the Added Total Annual Exposure (ATAE) which is the sum of the different exposures due to the mining with deduction of the initial natural exposure.
- The regulatory value is the sum of these values rated to their equivalent to the maximum recommended value. In France, the maximum recommended value is 5 mSv according to ICRP 26 recommendations. The Added Total Annual Exposure Rate (ATAER - complete formula herunder) must be less than 1.

$$TAER = \frac{gamma}{1\ mSv} + \frac{PAE\ Rn\ 222}{1.68\ mJ} + \frac{PAE\ 220}{0.56\ mJ} + \frac{IE\ dust}{100\ Bq\ alpha} + \frac{IE\ Ra\ 226}{4500\ Bq} + \frac{IE\ U238}{1\ g}$$

ATAER = TAER (station) - TAER (background) < 1

Based on field measures, the exposure and the regulatory value of ATAER of the critical group are evaluated with the following parameters :

- annual residence time : 7000 hours
- standard breathing rate : 0.8 m3.h-1
- daily amount of ingested water : 2.2 liters of the downstream water. This figure includes water ingested through food consumption.

This standard scenario leads to overevaluation as usually people won't stay so long inhouse neither drink water from the river.

RESULTS - IMPACT ON THE ENVIRONMENT

Figures from the area of ECARPIERE (Vendée) are given, as an example, in the following table. HAUTEGENTE is a location where people live, one kilometer from the impoundment. BEL AIR, further away, is considered as the reference in the area and was set up in 1991 to assess the radiological background of the granitic region.

1994	AIR PATHWAY				WATER PATHWAY		ICRP 26 : 5mSv		ICRP 60-5-8 1mSv	
STATION	EXT.EXP	INTERNAL EXPOSURE								
	Gamma ry nGy.h-1	Rn220 nJ.m-3	Rn222 nJ.m-3	Dust Bq.m-3	Ra226 Bq.l-1	Uranium mg.l-1	TAER	ATAER	TAER	ATAER
HAUTEGENTE	150	17	38	1	0,12	0,1	0,42	0,08	1,64	0,33
BEL AIR (Background)	120	13	29	1	0,05	0,1	0,34		1,32	

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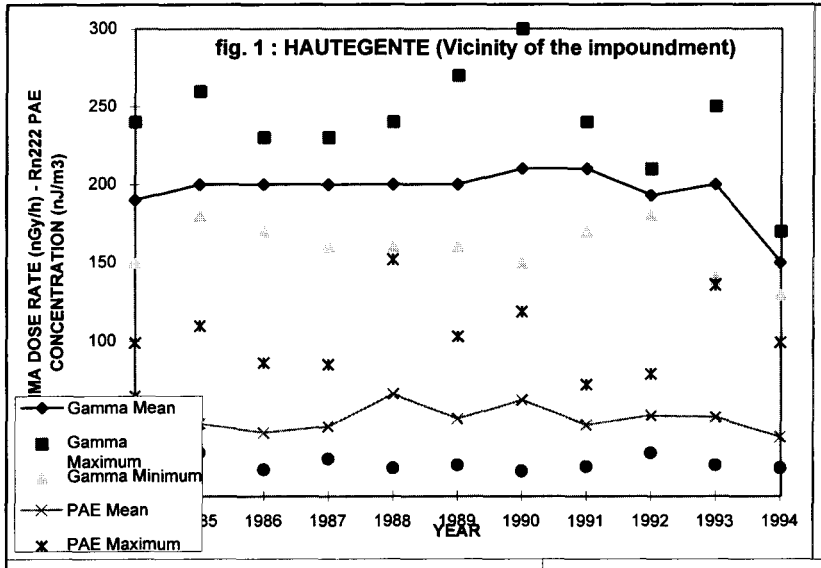
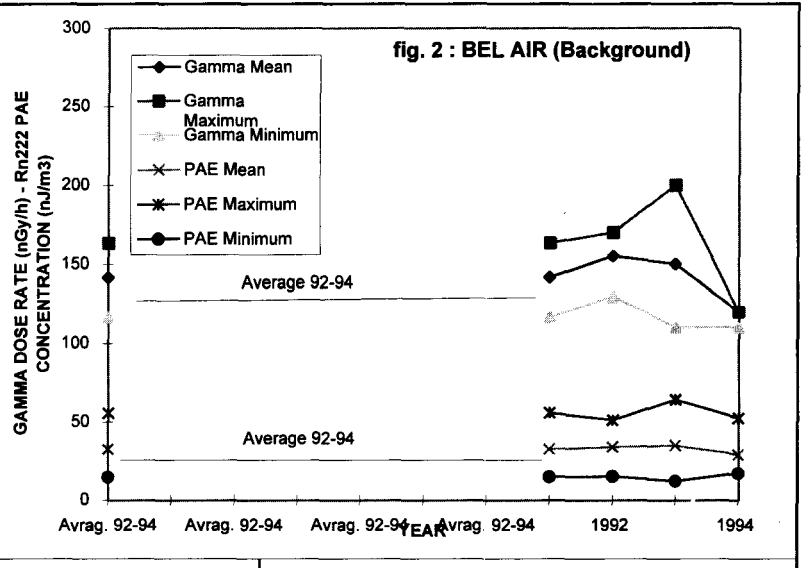
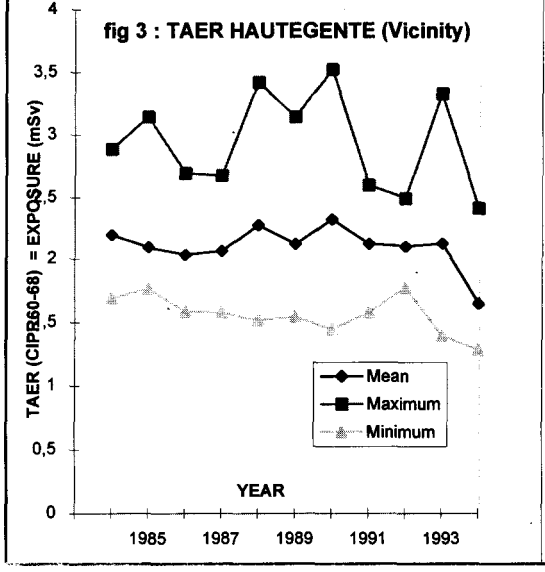
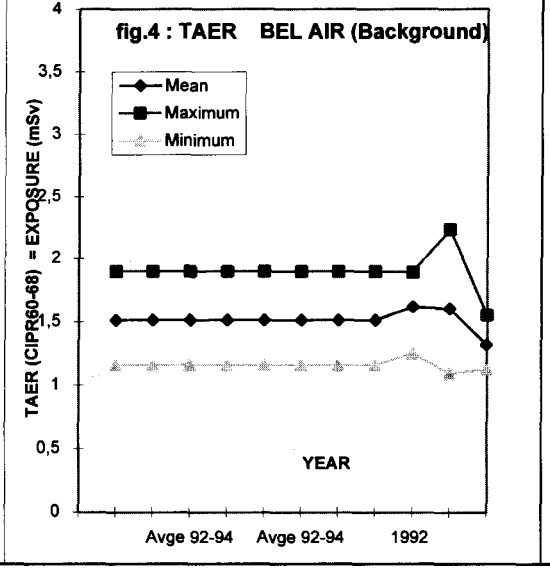
ECARPIERE : fig. 1-2 : MEASUREMENTS

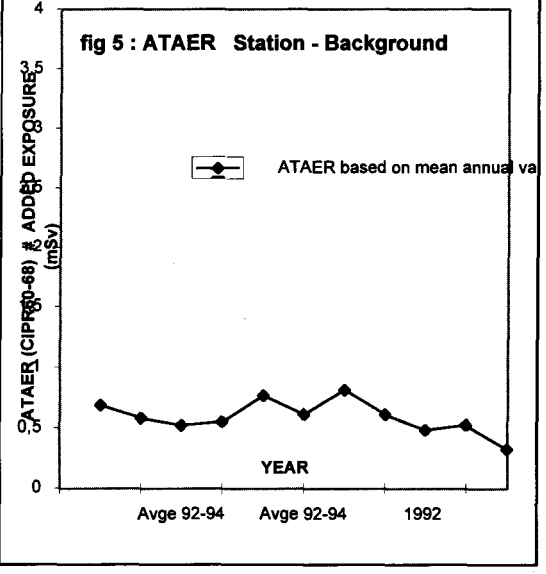
fig 3,4,5 : TOTAL and ADDED TOTAL ANNUAL EXPOSURE RATE (TAER and ATAER)











Although the different measures are slightly higher in Hautegente than in the background, the global impact of the site (ATAER = 0.08 if we consider ICRP 26, 0.33 for ICRP 60-65-68) is limited.
Results and their evolution during the past ten years are plotted on the diagrams for the same two stations :

- figures 1 and 2 show evolution of the measures for the two main pathways : dose rate of external exposure due to gamma irradiation (nGy.h-1) and concentration of Radon 222 Potential Alpha Energy (PAE Rn222 - nJ.m-3). The mean annual value is bounded by the maximum and minimum measurements recorded during the same year.
- figures 3 and 4 display the variation of the TAER using the equivalent to 1 mSv recommended by ICRP 60, 65 and 68. In this case TAER is equivalent to the total exposure.

Evolution of the resulting ATAER is shown on figure 5.

Sets of results concerning different areas are presented on the poster. They show variations of the ATAER up to 1mSv and confirm the following discussion.

DISCUSSION - COMPARISON WITH A 1 mSv LIMIT

Individual measurements (maximum and minimum quarterly or monthly values for respectively external exposure and potential alpha energy) show a great dispersion for both Hautegente and Bel Air. This is due to the natural variability of radioactivity and the uncertainty of the measurements (+/- 20% for Rn222 PAE [3], +/- 10% for external exposure).

Resulting annual TAER (or exposures) range from 1.3 to 3.5 mSv for Hautegente, 1 to 2.2 mSv for Bel Air that is a maximum difference of 2.2 mSv for Hautegente, 1.2 for Bel Air if maximum and minimum monthly or quarterly measures were used to calculate the annual TER.
As plotted on figure 3 and 4, the range is reduced to 1.6 to 2.3 (0.7 mSv difference) for Hautegente and 1.3 to 1.6 (0.3 mSv) if annual means of the different measurements are used for the calculation.

Although use of extreme values for the calculation of TAER is not realistic it confirms that only mean annual values should be taken into account.
In any case all the resulting calculations show that variability of TAE (or TAER) are near to 1 mSv which is the new recommendation of ICRP 60 for a limit to the added total exposure.

CONCLUSION.

- Should the 1 mSv limit be implemented in France, resulting Total Annual Exposure Rate would remain less than 1 in the case of ECARPIERE,
- but we have shown that ICRP 1mSv recommEndation is very difficult to adapt when dealing with natural radioactivity and consequently to the mining industry,
- consequently we think that applicability is linked to :
  - * a special attention to choose the measuring equipments used to monitor environment (active dosimeters)
  - * the use of annual means of the measures (as the seasonnal variability is too much important),
  - * better and realistic definition of the critical group and scenario used for calculation of the total exposure (7000 hours residence time give only a broad evaluation of the impact)
  - * better definition of how to measure, in the natural background, the different parameters used for calculation of the ATAER.

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- REPOWER INSTEAD OF RETIRE - A FOURTH ALTERNATIVE FOR DECOMMISSIONING COMMERCIAL NUCLEAR POWER STATIONS IN THE UNITED STATES

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INTRODUCTION

The three traditional alternatives for decommissioning commercial nuclear power stations in the United States (U.S.) are as follows (1):

- Decontamination (DECON) - the immediate dismantlement, in which the equipment, structures, and portions of the facility and site containing radioactive contaminants are removed or decontaminated permitting release for unrestricted use shortly after cessation of operations.
- Safe Storage (SAFSTOR) - the station is placed and maintained in such a condition that it can be safely stored and subsequently decontaminated (i.e., deferred decontamination) permitting release for unrestricted use.
- Entombment (ENTOMB) - the radioactive contaminants are encased in structurally long-lived material and then maintained under surveillance for an indefinite period of time.

Of the three options provided, the primary reason for deferring the start of decommissioning is to allow the overall radioactivity to decrease by decay, which in turn, will reduce worker risks, costs of dismantling radioactive components, and shipping large volumes of low-level radioactive waste (LLW). Previous evaluations suggest that a 50-year SAFSTOR deferment provides optimum material and radiation reduction benefits (2). However, during this period of deferment, a utility may need to consider replacement power to accommodate its projected electricity demand profile (EDP). Power generators in the U.S. are expected to order 140 Gigawatts (GW) of new generating capacity and put 100 GW into operation by the year 2003 (3). In addition, the average ages for fossil fuel stations and nuclear power stations are approximately 30 years and 16 years, respectively, with an expected useful life of only 30 to 40 years. Therefore, the projected EDP, compounded with the aging base-load capacity, including the untimely premature closing of a commercial nuclear power station will require the utility to address these concerns concurrently through new construction, life extension, or by repowering an existing station. Surprisingly, the concept of repowering was first recognized in 1974 (4), and is receiving more attention than in the past because of the inherent nature of life extension (e.g., equipment replacement and refurbishment) combined with the benefits of repowering (e.g., reduced emissions, an increase in generation capacity). For the purpose of this paper, repower (REPOWER) is defined as adding a new source of fuel to the existing steam-cycle system of a reconfigured commercial nuclear power station to ensure sufficient electrical generation capacity during the SAFSTOR deferral scenario. This paper reviews the various considerations for repowering, such as previous repowering projects, the gas turbine/heat recovery steam generator (GT-HRSG) repowered station, and the evolution of the non-nuclear station.

PREVIOUS REPOWERING PROJECTS

The W.H. Zimmer Station and the Midland Co-generation Station, both designed initially as commercial nuclear power stations and never operated, were reconfigured as a coal-fired station and a natural gas co-generation station, respectively (5, 6). In 1994, the Fort St. Vrain Generating Station was authorized to repower as a gas-fired combined-cycle steam unit consisting of two GT-HRSGs. Initially, this station operated as a 330 MWe high temperature (helium) gas-cooled reactor (HTGR) from January 1979 until August 1989 (7), and after repowering, is expected to generate approximately 471 MWe. A similar combined gas-steam cycle design has also received particular attention in Europe (8).

THE GT-HRSG REPOWERED STATION

Currently, the most common and cost-effective repowering approach is to use natural gas as a source of fuel, add modern GT-HRSGs, and make use of the existing site and internal components of the steam-cycle system (e.g., turbine-generator, condenser, and cooling water systems). Also, as natural gas prices have declined, utilities have taken advantage of several coinciding market trends to further enhance the rise in the application of gas turbines for electric power generation (9):

- favorable construction economics and environmental benefits provided by gas-turbine-based stations;
- achieving very high combined-cycle efficiencies of 54% to 58% using commercially available technology;
- reliability and availability figures of 98% and 95%, respectively, from gas turbine manufacturers; and
- an EDP requiring large blocks of peaking power; i.e., gas-fired stations are more attractive for peaking applications because low-capacity utilization such as peaking service is often best served by a low capital-cost power station.

The common denominator in these projects is to replace the reactor vessel and coolant system by adding several 100-150 MWe GT-HRSGs (Brayton cycle) to a refurbished nuclear steam-turbine-based (Rankine cycle) station. A gas-turbine-based station consists of a compressor, combustion chambers, and a turbine section. This method of GT-HRSG combined-cycle repowering can improve overall plant thermal efficiency by more than 20%, and in some cases triple the total Megawatt output of the original station (10).

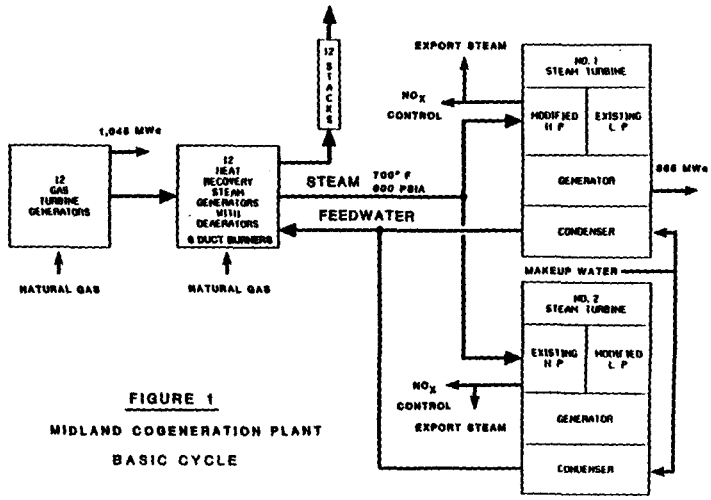

FIGURE 1

MIDLAND COGENERATION PLANT

BASIC CYCLE

Figure 1 represents a general configuration of the Midland Co-generation Venture that was successful in converting two large nuclear steam turbines into a GT-HRSG combined-cycle arrangement (6).

EVOLUTION OF THE NON-NUCLEAR STATION

The containment building will be isolated from the repowered side of the station by installing permanent physical boundaries within the nuclear power block, thereby, enclosing the areas and systems that contain essentially all of the significant radioactive contamination and activated components. The potential of nuclear criticality will have been precluded by transferring the fuel assemblies from the reactor vessel to either the spent fuel pool or an Independent Spent Fuel Storage Installation (ISFSI) (11). Depending on the thermodynamic nature of the repowered system, it may be cost effective to "eliminate and replace" rather than "decontaminate and refurbish" certain components (e.g., pumps, valves, moisture separators). In all practicality, consideration should not be given to decontaminating the small bore piping used for sampling radioactive liquid effluents. In addition, instrument sensors, transducers, and seals having been in contact with steam or condensate containing residual radioactive materials (RRM) will be removed, characterized, and discarded as LLW. Because of the complex geometry of components like the turbine, piping, and associated steam and condensate systems, there will, in all likelihood, be hidden pockets of RRM present creating high radiation dose rate "hot spots" or "crud traps," even following extensive decontamination (12). System perturbations could displace the RRM, that is analogous to a "crud burst" in an operating commercial nuclear power station. Therefore, to ensure that such potential releases are contained, as well as ensure boiler feedwater quality, full-flow condensate ion-exchange polishing capabilities should be maintained for the repowered station. To alleviate stakeholder concerns regarding the potential release of RRM still residing in the steam-cycle system following decontamination, a

modified in-house and environmental monitoring program will be maintained throughout the repowering period. The environmental program will consist of monitoring gaseous, particulate, and liquid effluents continuously, which is no different than monitoring other system parameters (e.g., water chemistry). Nevertheless, the radiation dose to the public from a postulated release of RRM remaining on inaccessible work surfaces (e.g., interior of pipes, drain lines) is expected to be indistinguishable from background levels. Additional station systems foreseen to be required, but downgraded during repowering could include, but are not limited to radiological and security access, the management of LLW, AC and DC electrical power systems, and fire protection/suppression systems.

CONCLUSION

The fourth alternative for decommissioning (REPOWER) focuses on replacing an existing fuel source with GT-HRSGs, and reusing the internal non-nuclear components to yield a combined-cycle power station. In addition, the inherent benefits of modifying the serviceable steam cycle improves efficiency, expands electrical generating capacity up to three times the original station output, while maintaining a favorable environmental profile. Repowering is especially well-suited for stations that satisfy mid-range power demands (e.g., 400-700 MWe), and is expected to have a design life similar to that of a new station (e.g., 40 years). Therefore, the REPOWER alternative allows the utility to accommodate its projected EDP, and reduce the need for new construction by using an existing site already dedicated to the generation of electricity.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

03. Aug. 1995

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Abstract No. 90615
Receipt
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Acceptance
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(Instructions for preparation on reverse)

PAPER TITLE Radiation Dose Control in Decommissioning

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MAJOR SCIENTIFIC TOPIC NUMBER 4.7 (see page 7)

ABSTRACT (See instructions overleaf)

Sellafield in Cumbria [UK] has been in use as a nuclear site for number of years. It now contains a 'second generation' Magnox reprocessing plant, modern storage ponds, fuel dismantling facilities, product finishing lines, and the Thermal Oxide Reprocessing Plant [THORP]. It also houses a complex of waste treatment and storage facilities, including a vitrification plant, and the Calder Hall Nuclear power station. Over the years a number of the facilities on the site have become redundant and there is a programme to decommission these areas. A business unit within British Nuclear Fuels plc, UK Group, Waste Retrieval and Decommissioning (WR&D) has been set up to carry out this work. The first strategic objective in the WR&D Business Plan is 'To retrieve waste and decommission facilities safety, cost effectively and within estimated cost and timescale'. This paper will deal with some of the decommissioning work being carried out by WR&D and the dose control strategies, including ALARP which are being used.

DECOMMISSIONING OF WESTERN EUROPE'S OLDEST REACTOR THE GRAPHITE LOW ENERGY EXPERIMENTAL PILE (GLEEP) AT UKAEA HARWELL

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INTRODUCTION

The Graphite Low Energy Experimental Pile (GLEEP) was the first nuclear reactor to be built in Western Europe and was designed to operate at a maximum power of 100 kW, although it was generally run at 3 kW. It is situated on the UKAEA Harwell site in the UK and was a natural uranium graphite moderated reactor which operated from 1947 until it was finally shutdown in September 1990 making it Western Europe's oldest and longest serving reactor. Decommissioning of the Facility commenced soon after closure. All the fuel elements, activated and ancillary items have been removed as part of decommissioning, leaving only the graphite block housed within a 1.52 m thick barytes concrete biological shield. This paper reviews the Stage 2 decommissioning experience. A more detailed description of GLEEP and of the fuel removal process is given in Reference 1.

DECOMMISSIONING STAGES

Decommissioning was managed by AEA Technology on behalf of UKAEA. Following Stage 2 Phase 1 decommissioning, formal hand back of the facility from AEA Technology to UKAEA took place on 9 June 1995.

Decommissioning of the reactor has been divided into three Stages;

Stage 1 decommissioning was completed at the end of August 1994 and involved the removal of fuel from the reactor using specially designed equipment (since fuel changes were not part of the normal operation of GLEEP). The tasks carried out during this stage of decommissioning are described in Reference 1.

Stage 2 decommissioning has been divided into two Phases. **Phase 1** was completed in April 1995 and comprised the removal of activated items, such as the absorbers and control rods, from the reactor. In addition, other preparatory operations for Care and Maintenance were carried out such as altering access arrangements and the removal of the control room, leaving only the graphite block and concrete biological shield. The facility is currently in **Phase 2** which is an extended period of Care and Maintenance scheduled to last until the year 2008 although there are several other options which may become more favourable. No operations within the reactor biological shield are scheduled for the extended period of Care and Maintenance and therefore the number of safety related items is much reduced and all of which are passive control systems.

During **Stage 3**, decommissioning and removal of the remaining reactor structures will be carried out, thus allowing unrestricted use of the area.

DESCRIPTION OF STAGE 2 DECOMMISSIONING TASKS

Stage 2 Phase 1 decommissioning was divided into Parts 1 and 2. **Part 1** comprised the removal of all items within the biological shield other than the graphiite block. **Part 2** involved removal of the remaining items outside the biological shield so that only the graphite block and concrete biological shield remained for the extended period of Care and Maintenance.

The major tasks that were carried out during Stage 2 Phase 1 Part 1 are summarised briefly below:

- all activated items, including the absorbers and their mechanisms, were removed;
- the redundant fuel unload equipment was removed;
- exposed holes in the walls and roof of the concrete biological shield were filled with expanding foam and screeded with concrete.

The major tasks that were carried out during Stage 2 Phase 1 Part 2 are summarised briefly below:

- all loose combustible items were removed from the GLEEP area;

- all Low Level Waste (LLW) and Very Low Level Waste (VLLW) stored in the GLEEP area was disposed of in an appropriate manner;
- the GLEEP standard radium-beryllium source was removed from the vertical storage hole and removed for storage as Intermediate Level Waste;
- redundant reactor instrumentation, such as the area gamma monitors and fire detection systems, were removed and disposed of;
- all redundant electrical cabling in the GLEEP area was removed;
- the GLEEP ventilation system was decommissioned and disposed of;
- part of the steel galleries/walkways were removed and the control block was demolished;
- the GLEEP workshop and offices were cleared;
- 'fixed' spots of contamination on the floor of the general GLEEP area at faces A and B were removed using standard decontamination techniques. This area, which is external to the biological shield, is now no longer a Radiologically Designated Area.

WASTE ARISINGS

The majority of waste generated during Stage 2 Phase 1 decommissioning was below the exemption level of 0.4 Bq/g given under the Radioactive Substances Act (RSA93) and could therefore be classified as 'free release' waste. In total, 244 m^3 of 'free release' waste, including graphite and concrete blocks, was packaged into skips and disposed of in an appropriate manner. More skips than anticipated were required due to practical difficulties associated with packing the waste which lead to less efficient use of space.

There were two disposal routes for items of Low Level Waste (LLW) which exceeded the exemption level of 0.4 Bq/g:

- 5 m^3 of LLW was placed in an ISO container and removed pending disposal;
- 2.9 m^3 of LLW was packaged into 10 litre waste drums and removed for storage pending final disposal.

In addition, 25 litres of Intermediate Level Waste (ILW) was generated and removed for storage until a suitable disposal route becomes available. Routine operational waste was also generated, such as gloves, overshoes and coveralls. No liquid wastes were generated during Stage 2 Phase 1 decommissioning.

RADIOACTIVE INVENTORY

Prior to commencement of decommissioning, the fuel elements were the principal radioactive material within the GLEEP Facility with an activity of approximately 15.5 TBq (i.e. about 99.98 % of the total radioactive inventory). Removal of the fuel elements and activated items has reduced the radioactive inventory of the facility to that associated with the graphite block, its associated steel base and support structures and the concrete biological shield.

A detailed estimate of the radioactive inventory of the structure materials in the GLEEP reactor was carried out in April 1993. The estimated total activity associated with each of the remaining reactor components is given below:

Steel Base and Supports for Graphite Block	2.5E10 Bq
Graphite Block	3.0E10 Bq
Lower Biological Shield	1.8E10 Bq
Upper Biological Shield	6.5E9 Bq

The estimated total radioactive inventory of the facility remaining for the care and maintenance phase is therefore 80 GBq composed of activation products as a result of neutron irradiation of its constituent materials. There is no significant surface contamination present within the biological shield. It should be noted that tritium contributes most to the inventory.

DOSES RECEIVED

No member of the decommissioning staff exceeded any individual dose limit and the team as a whole complied with the group average target of 5 mSv. The following whole body dose equivalents were received during Stage 2 Phase 1 operations:

Collective Dose	1.94 man mSv
Individual Average Dose	0.32 mSv
Maximum Individual Dose	0.66 mSv

A comparison of the estimated and actual collective doses received by operators during Stage 2 Phase 1 decommissioning operations is given below. It should be noted that Phase 1 Part 2 operations were performed outside the biological shield area (i.e. < 2 μSv/h) and therefore no dose allowance was made.

Estimated Collective Dose	3.19 man mSv
Actual Collective Dose	1.94 man mSv

The collective dose received by operators was approximately 60 % of that estimated in the Decommissioning Safety Case mainly because operations took less time than expected and operators minimised the time spent in higher dose-rate areas within the biological shield. The predicted dose budget was also a significant overestimate as it took no account of the general reduction in dose rates as activated items were removed.

WORK PROGRAMME

Stage 2 Phase 1 decommissioning was completed in April 1995 although it was scheduled to last until mid-June 1995. This is because more operators were used than originally anticipated, thus allowing certain tasks to be performed in parallel. It should be noted that the Decommissioning Programme assumed that either two or three operators would be used, whereas up to six operators were used in reality.

In addition, personnel used during Stage 2 Phase 1 decommissioning were familiar with the facility, having carried out the fuel unload operations. This shows the importance of using experienced personnel who are familiar with the facility and do not require significant training and direct supervision.

Another reason for the operations being completed ahead of schedule is that certain contract variations were implemented which increased the efficiency of decommissioning. For example, the GLEEP boundary fence was removed thus facilitating vehicular access for removal of the control block and demolition of the control room, etc. The decision not to completely remove the steel galleries/walkways also saved time. This shows the benefit of adopting a flexible attitude to decommissioning.

CONCLUSIONS

Stage 2 Phase 1 decommissioning of the GLEEP Facility was successfully completed in a satisfactory manner ahead of programme. Decommissioning staff received less than the estimated collective dose and the operations met all Regulatory, Statutory and Authority accident risk criteria, dose limits and targets.

ACKNOWLEDGEMENTS

This work has been funded under the DRAWMOPS programme of the UK Department of Trade and Industry (DTI). The results of this work form part of the UK Government programme on decommissioning and radioactive waste management, but do not necessarily represent Government policy.

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DECOMMISSIONING ANALYSIS OF A UNIVERSITY CYCLOTRON

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INTRODUCTION

In the widespread use of some medical nuclear facilities, such as cyclotrons for isotope production, life cycle analysis, including decommissioning, was not taken into account. The structual materials of an accelerator and the concrete shielding of the bunker are activated by neutrons. This could yield a considerable volume of nuclear waste and needs radiation protection concern for occupational workers and the environment during some decennia (1).

At the university of Brussels (VUB) a prospective radiation protection and waste analysis is being made for the later decommissining of their cyclotron (2). Only few similar studies have been published (1,3).

In Belgium future nuclear dismantling operations will be submitted to a radiation protection authorisation procedure. Meanwhile the nuclear waste authorities insist on dismantling planning, including financial provisioning.

An optimisation exercise was made at the VUB-cyclotron, taking into account international trends to clearance levels for low level nuclear waste.

Conceptual prevention opportunities e.g. selective material choice could be identified for future accelerator constructions.

ACTIVATION AROUND A CYCLOTRON

At the VUB a variable energy, multiparticle (protons, deuterons, α- and ^{3}He- particles) cyclotron (CGR-560) is in operation since 1985. The maximal current on target is 120 μA for 30 MeV protons. This machine is used for physics research and for radionuclide production in a shielded vault complex with 4 irradiation rooms (bunkers). The most important contribution to activation is due to ^{201}Tl isotope production at 27,3 MeV protons for radiopharmaceutical companies.

The bunker shielding has a thickness of 2.5 m totalling a volume of 2700 m³ of concrete.

The spectral characteristics of fast neutron beams were determined in different directions, showing an average energy of 12 Mev at 0°. Through multiple reflection on walls and elastic scattering in concrete, thermalisation of neutrons occurs in depth. The activation of trace elements in sand e.g. europium and of particular metals yields medium living radioactive products. They are created by capture reactions with high cross sections and by some threshold reactions with lower yield, as indicated below.

^{151}Eu (n,γ) ^{152}Eu	σ_{th} = 5900 barn	$t_{1/2}$ = 13.33 year
^{153}Eu (n,γ) ^{154}Eu	σ_{th} = 390 barn	$t_{1/2}$ = 8.8 year
^{133}Cs (n,γ) ^{134}Cs	σ_{th} = 29 barn	$t_{1/2}$ = 2.06 year
^{59}Co (n,γ) ^{60}Co	σ_{th} = 37 barn	$t_{1/2}$ = 5.3 year
^{45}Sc (n,γ) ^{46}Sc	σ_{th} = 26.5 barn	$t_{1/2}$ = 83 days
^{58}Fe (n,γ) ^{59}Fe	σ_{th} = 1.15 barn	$t_{1/2}$ = 44 days
^{64}Zn (n, γ) ^{65}Zn	σ_{th} = 0.78 barn	$t_{1/2}$ = 244 days

^{55}Mn (n,2n) ^{54}Mn	σ_{max} = 910 mbarn (18MeV)	$t_{1/2}$ = 312 days
^{54}Fe (n,p) ^{54}Mn	σ_{max} = 590 mbarn (10MeV)	$t_{1/2}$ = 312 days

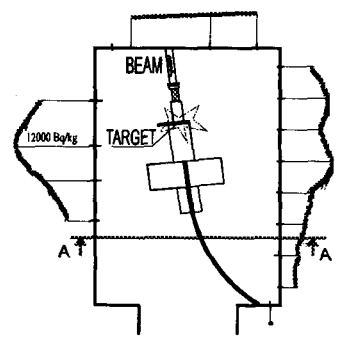

Fig 1 : Spatial distribution of maximal ^{152}Eu in concrete shielding.

The activation distribution was assessed by γ-spectrometry on 5 cm bore samples. They were systematically taken up to a depth of 50 cm throughout the shielding structure presented in fig. 1.

RESULTS OF ACTIVATION MEASUREMENTS

The highest specific activation in concrete occurs laterally near the target at 15 cm depth (fig 2). The iron reinforcement bars in the concrete shielding show maximum activation levels of 32 Bq/g. The activity decrease is exponential for both cases. Prediction of depth profiles could be derived from the measurements (fig 3). The irradiation room infrastructure consists for 70 % of steel. Specific activities up to 360 Bq/g ^{54}Mn and ^{60}Co are measured.

The huge steel accelerator yoke of 80 ton however is showing much lower level activation due to its low ^{59}Co content. These values can be compared with clearance levels of about 0,3 Bq/g (4).

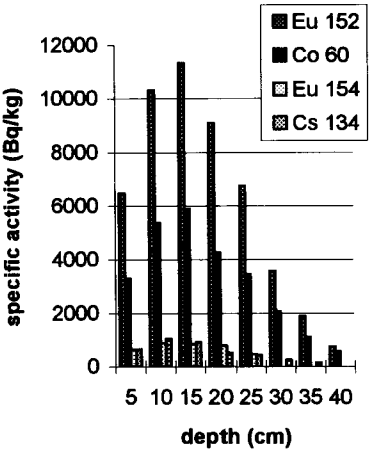

Fig 2 : Depth distribution in a concrete bore sample of specific activity for ^{152}Eu, ^{60}Co, ^{154}Eu, ^{134}Cs

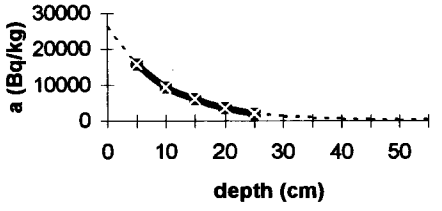

Fig 3 : Measured ^{60}Co activation in iron reinforcement bars

OPTIMISATION OF DISMANTLING SCENARIOS

Dismantling of a cyclotron will be submitted to an authorisation procedure in future Belgian regulations. Waste quantities have to be estimated with specification of their destination. The protection of man and the environment against radiation has to be guaranteed during preparatory work an dismantling operations. Site destination and future use should be specified.

In this preliminary study, the following scenarios were considered :

- Early dismantling of a cyclotron infrastructure, directly after shut-down is confronted with an internal contamination risk by ^{65}Zn. This copper activation product is easily dispersed. ^{65}Zn is detected regularly at acceptable levels in whole body monitoring of cyclotron maintenance workers.
- The waste from metal infrastructures having activities up to 1000 times clearance level could be stored for decay during 35 years in the most activated cyclotron vault. This should be done after the management of the contanimation problems.

- Early decommissioning of the concrete walls to 30-50 cm depths could yield up to 100 m^3 of nuclear waste, applying IAEA proposed clearence levels (4). The dismantling techniques available in nuclear fuel cycle industries allow to remove the activated depths. Since no provisions are made, the cost of this option is too high for an university.
  Dilution techniques, mixing active and inactive crushed concrete and melting iron bars, could be applied in order to arrive at a reasonable cost. The authorisation of such an approach has been given in the UK (5). It is not evident in Belgium, where the obligation of an environmental assessment report could need to take alternative options into account.
- A decay on site of the activity of the concrete rooms and of the metal infrastructures till clearance levels could be performed. Fig 4 illustrates that cooling times of maximum 70 y are necessary for this option.
- Restricted use of cyclotron rooms as controlled areas in the intermediate period was evaluated. Doserates in the documented bunker were calculated from the activation measurements using conservative assumptions. A maximal doserate of 60 µSv/h was derived. Measurements indicate actual doserates up to 15 µSv/h. Occupational use of such rooms yield calculated doses between 6 and 120 mSy/y.
  Surface doses were measured with TLD on concrete samples with a maximum of 2 mGy/y.

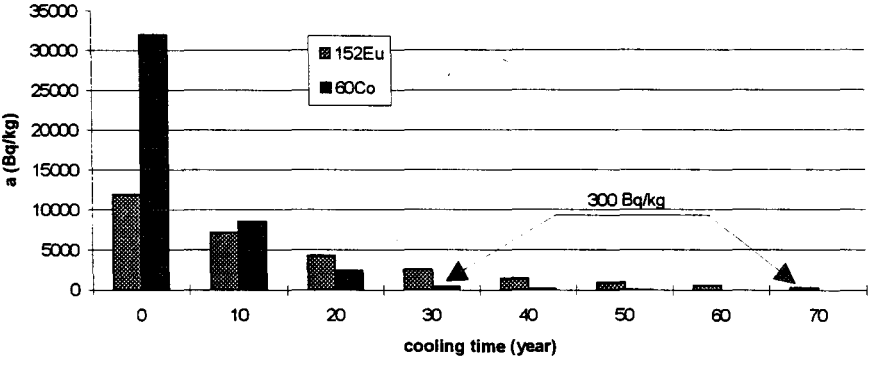

Fig 4 : Decay of Activation of Concrete Bunker

PREVENTION OPPORTUNITIES

The use of iron and scrap metal in the concrete shielding is not recommended for cyclotrons. The use of removable blocks for the inner part of the shielding walls could reduce dismantling costs. Local neutron absorption measures could be taken to reduce the source term. A selection of sand with low europium content for concrete preparation and a preference for low ^{59}Co steel or Al for the infrastructure could be taken into account in cyclotron complex conception.

CONCLUSION

Regarding the high cost of nuclear waste, decay during about 50 years of the infrastructure after shut-down is necessary to allow cooling to proposed clearance levels. Decontamination for ^{65}Zn could be performed after some years.

Considering dose limits for workers and the public, a restricted use of cyclotron rooms as controlled area, seems the most reasonable option. Life cycle analysis techniques should be integrated in the planning of isotope production facilities. It could contribute to reduction of later nuclear waste and to sustainable nuclear development.

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RADIATION PROTECTION ASPECTS OF THE DECOMMISSIONING OF THE LINAC-ADONE STORAGE RING

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Introduction

An e^+ e^- collider, christened DAΦNE (Double Anular Φ factory for Nice Experiments), optimized for operation at a total energy of 1020 MeV, is under construction at the Frascati National Laboratories (LNF) of the National Institute of Nuclear Physics (INFN). The new machine will be placed into the existing buildings which in the past housed the Linac-Adone complex, which definitively ceased operation April 26th 1993 and was at once decommissioned.

The Linac-Adone complex has operated without stopping up to the 26th of April 1993 except for the ordinary maintenance periods.

It was composed by a Linac, capable of accelerating 100 mA of e^- beam peak current to 400 MeV and 1 mA of e^+ beam peak current to 365 MeV, in operation since 1964, and by an e^+ e^- storage ring capable of storing $2x10^{11}$ particles per beam at 1500 MeV, in operation since 1967.

Radiation safety standard and criteria for unrestricted release adopted at LNF

The objectives of radiation safety for the Linac-Adone decommissioning were:

- limitation of personal doses;
- control of radioactive materials either for reuse or for disposal;
- prevention of dispersion of radioactive material during handling and transportation to the final destination.

The limits for personal doses and radioactive material were taken from the recommendations of the Italian National Agency of Environment Protection (1) and from the law in force on the safety and health protection for workers and population against the danger of ionizing radiation field (2), as follows:

- the reference dose for people working in decommissioning areas was 15 mSv/y;
- a limit for unrestricted release for β–γ emitters was set at 1 kBq/m^2 for surface contamination and 1 kBq/kg for mass activity.

Preliminary measurements

Following the shut down of the Linac-Adone complex, the Health Physics Service carried out many measurements to find out the amount of dose rate along e^+e^- beam lines, the specific activity in dust, in cooling water and in metallic components of the machine, in order to obtain useful information for planning the decommissioning itself.

Gamma dose rates, using CaF2 TLD dosimeters (bulb dosimeter mod. 4040 by Harshaw) and a Victoreen ionization chamber mod. 450 P, were measured along e^+e^- beam lines at the distance of about 30 cm from the machine. The results obtained, shown in Figure 1 stress that the dose rate values are higher than the background only near the positron converter and the final part of the e^- beam line before beam dump.

Liquid samples, collected from the primary cooling circuit of the machine supplied with aqueduct water and from the secondary circuit supplied with distilled or demineralized water show (Table I) an appreciable concentration of H-3, as expected, in cooling water of beam dumps.

Tab. I - Measured specific activity in water of Linac-Adone-Leale cooling circuits

Radionuclides	Co-58	Co-60	H-3
Cooling circuits	(Bq/l)	(Bq/l)	(Bq/l)
Linac-Adone primary cooling circuit			3.1±1.3
Linac secondary cooling circuit (5350 l)		1.47±0.07	21.9±1.6
Wiggler secondary cooling circuit (5500 l)		2.00±0.08	21.7±1.6
Adone secondary cooling circuit (5900 l)			20.8±1.6
Leale primary cooling circuit			18.8±1.7
Leale secondary cooling circuit S2 (1500 l)			30.8±1.8
Leale secondary cooling circuit S3 (1000 l)			7.4±1.6
Linac beam dump 1 cooling circuit (2000 l)		0.27±0.03	3.5E04±43
Linac beam dump 2 cooling circuit (6000 l)		1.38±0.06	3.4E04±42

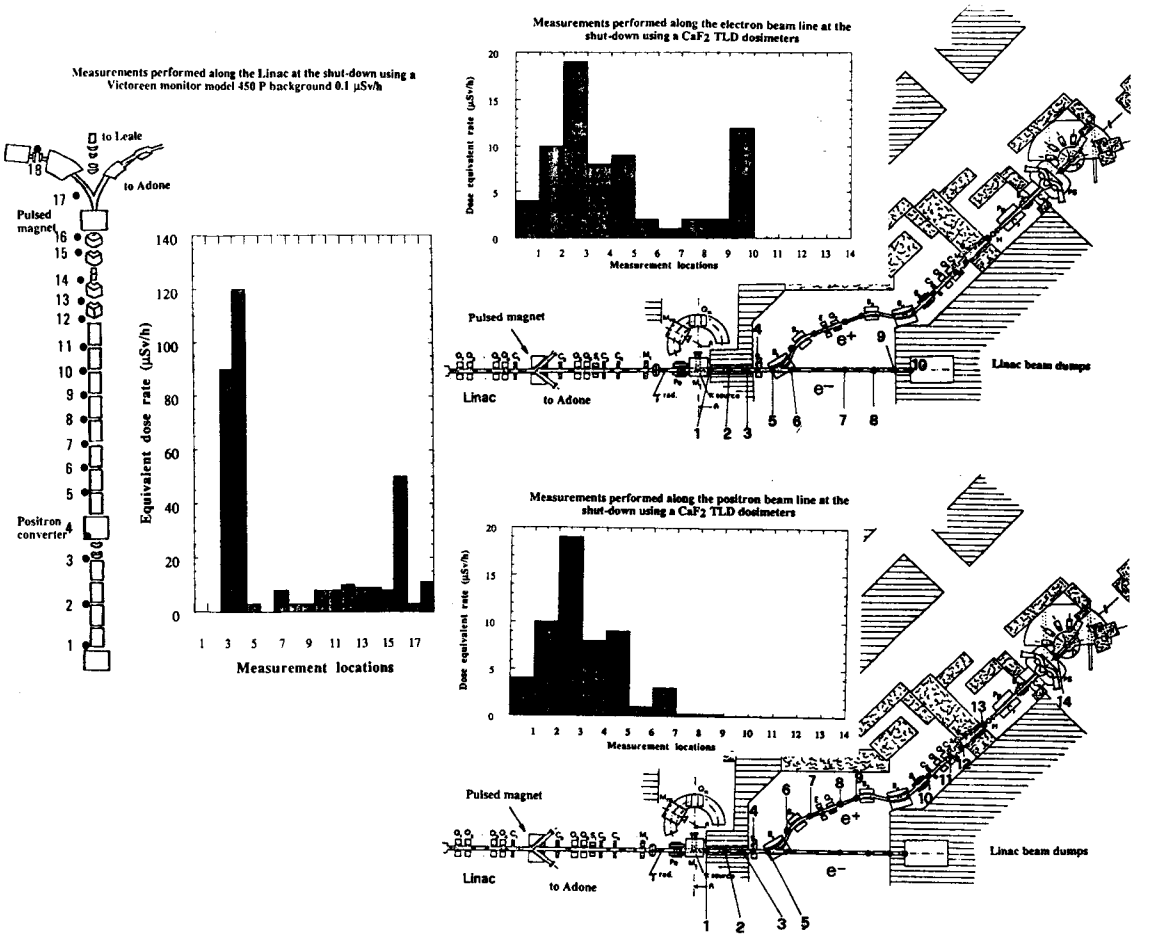

Figure 1 Dose equivalent rate from residual radioactivity

Dismantling of the machine elements and final cleaning

On the basis of the results of the preliminary measurements the decommissioning program, the handling and transportation of materials began and were completed in 3 months without particular protections for people involved except the use of personal dosimeters and disposable dresses used in dusty operations.

During the dismantling the following instruments were used for in the field measurements:

- two portable HpGe EG&G spectrometric systems;
- two environmental Reuter Stokes ionization chambers;
- a Victoreen 450 P ionization chamber;
- a Victoreen Frisker connected with a pancake probe;
- other useful monitors.

Two HpGe spectrometric systems in network with a 486 IBM PC were used for the quantitative analysis of samples.

Materials coming from Linac, including the 12 accelerator sections, the magnetic structures, the vacuum pipe line, the Linac-Adone transfer lines and 43 baskets, filled with remaining materials, were transported to the storage area without any quantitative measurements.

On the basis of the γ rate measurements it was clear that the most part of the materials had a concentration values higher than the clearance levels while the remaining ones not voluminous but very numerous, would have paralized uselessly all Health Physics Service γ ray spectrometric systems. More accurate measurements are foreseen for the future.

TELEROBOTIC DECOMMISSIONING OF THE DIDO HIGH ACTIVITY HANDLING CELL AT UKAEA HARWELL

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SYNOPSIS

This paper reports on the decommissioning of the DIDO High Activity Handling Cell (HAHC) at UKAEA Harwell, focusing on the successful use of a Nuclear Engineered Advanced TeleRobot (NEATER). The use of NEATER achieved a significant reduction in collective dose for the project with a saving of both cost and time, when compared with estimates for traditional decommissioning techniques. The HAHC project has provided a range of experience and awareness of the potential for remote telerobotic operations in decommissioning.

1. INTRODUCTION

The HAHC was used for the handling of irradiated experimental equipment and specimens from the DIDO and PLUTO Materials Testing Reactors at Harwell. The facility was closed in 1984 due to concern over the safety of continued operation of the cell, mainly associated with the deterioration of the ventilation system. The cell remained under care and maintenance until 1989 when a decision was made to decontaminate and decommission the cell and its contents. The decommissioning project started in January 1989 and was undertaken by AEA Technology. The decommissioning project was completed in five Phases, commencing with Phase 1 'Planning and Documentation', with the radiological end point being complete removal of the cell.

2. DESCRIPTION OF FACILITY

The HAHC consisted of the Main Cell (dimensions 7.3m long x 2.4m wide x 3m high) with adjoining Maintenance Bay and Posting Areas located at either end of the cell. The Main Cell was constructed from modules of double walled steel plates, the gap between the plates was filled with sand to provide shielding.

The cell had three pairs of master slave manipulators (MSMs) adjacent to zinc-bromide filled viewing windows. A large power manipulator ran on overhead rails that traversed the full length of the Main Cell and into the Maintenance Bay. A pair of railway lines running the full length of the floor in the facility carried a trolley to allow equipment and waste to be transported into and out of the cell. The major items of equipment in the cell at the start of decommissioning comprised a large steel bench in front of the viewing windows, an auxiliary bench, a set of steel rollers in line with the posting port and a static vacuum unit.

3. INVENTORY

The radioactive inventory of the cell was estimated to be 6×10^{12} Bq cobalt-60 with a small percentage (< 1%) of europium isotopes also present. The majority of contamination was found to be loose, due to large fines and saw cuttings from routine work to cut up cobalt pencils. Radiation levels in the Main Cell were up to 1 Svh^{-1} on contact with parts of the main bench.

4. PRE-DECOMMISSIONING PREPARATION

Phase 2 'Preparation of the Cell for Decommissioning Activities' involved a programme of refurbishment to bring all the systems and equipment required for decommissioning up to modern standards. This included the installation of a new auxiliary ventilation system and a modular containment system.

Specialist foam decontamination techniques developed by AEA Technology at Winfrith were evaluated in Phase 2 during decontamination of the Maintenance Bay. It was found that several applications of foam were required to penetrate the accumulated grease and dirt and that as layers of dirt were removed the radiation levels fluctuated indicating that contamination had been masked.

In Table II are reported, for istance, the results of the measurements carried out on the most radioactive parts of the accelerator.

Tab.II Qualitative gamma-rays spectrometry on Linac components

Material	Dose equivalent rate (μGy/h)	122 keV Co-57 (cps)	321 keV Cr-51 (cps)	810 keV Co-58 (cps)	835 keV Mn-54 (cps)	846 keV Co-56 (cps)	1275 keV Na-22 (cps)	1332 keV Co-60 (cps)
Linac section n°5 at 1 m of distance	750	21			39		1	154
positron converter target at 50 cm of distance	230	8			4		1	170
positron converter flange at 1 m of distance	500	137		1.7	127		197	12

Dose rate measurements carried out with a Reuter Stokes mod RSS112 during the cleaning of the Linac area from the remaining dust and reported in Figure 5, have shown appreciable values in the positron converter area with a maximum of 3 μSv/h .

A quantitative γ spectrometry measurement of a concrete sample, collected from a wall of the positron converter area, stressed the presence of Eu-152, Co-60, Cs-134, Eu-154 (Table IV), due to the concrete activation (n, γ) of the Linac walls (3).

The decommissioning of the Adone area has caused many problems due to the big volumes and weights involved and to the very low value for unrestricted release. The Adone storage ring was formed of 12 identical sectors, each one composed by a straight section, a bending magnet, and a couple of quadrupoles for a total weight of about 35 ton/sector and the concrete shielding weighted about 800 tons. In situ γ spectrometry measurements were carried out on each part of the sectors in order to collect a sample from the most radioactive area. Anyway, the final destination of materials was determined on the basis of quantitative results. All materials coming from Adone were released free except a few vacuum chamber.

Some results of the quantitative analysis, carried out on the most representative magnetic elements and vacuum chamber of the storage ring are reported in Table III. The measurements stress the presence of Mn-54 in the iron of a few bending magnets and Co-60 in the inox steel of a few vacuum chambers, where beams usually were lost (near injection devices).

Tab. III Quantitative gamma-rays spectrometry on some representative Adone components

Radionuclide mass activity	Co-57 (Bq/g)	Co-60 (Bq/g)	Mn-54 (Bq/g)
Sample collected from			
Quadrupol 43		0.00085 ± 0.00007	0.022 ± 0.002
Bending magnet 2		0.0080 ± 0.0007	
vacuum chamber of bending magnet 2	0.011 ± 0.001	0.017 ± 0.002	0.014 ± 0.002
vacuum chamber of quadrupol 43	4.70 ± 0.40	2.14 ± 0.30	9.70± 0.80

Conclusion

The decommissioning of the whole Linac-Adone complex, including radiation protection measurements, has been completed in 6 months.

The continuous survey carried out by the Health Physics Service (1 physicist and two technicians) has obtained two important aims:

1) the movement of the activated material to the area of storage has been carried out avoiding any loss;
2) people involved in decommissioning have received doses, no higher than the background.

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5. REMOTE DECOMMISSIONING OF THE MAIN CELL

Phase 3 'Remote Decommissioning of the Main Cell' began in June 1991. The aim of this phase was to achieve radiological conditions acceptable for man-entry into the Main Cell, by remote decontamination and removal of contaminated items from the cell. These conditions were defined as 500 μSvh^{-1} measured as a general background radiation level in the cell.

During the operation to drive the overhead power manipulator out of the Main Cell an oil seal ruptured spilling a considerable amount of hydraulic oil into the Main Cell and work was suspended while a clean-up operation was carried out. It was originally planned that items would be size-reduced in the cell using plasma-arc cutting equipment deployed by the MSMs. However, this technique was prohibited due to the fire risk from residual spilt oil and the MSMs were not strong or rigid enough to manipulate the alternative heavy cutting gear.

An alternative approach was an extensive remote decontamination programme to reduce radiation levels to permit man-entry, followed by manual size reduction. Efforts were made to decontaminate the Main Cell and principally the main bench by remote techniques, however the traditional decontamination methods achieved only limited success in reducing the background radiation level. This indicated that several cycles of cleaning would be required taking a considerable amount of time, inflating the cost and there was no guarantee that this method would achieve the required end point of conditions suitable for man entry.

6. NUCLEAR ENGINEERED ADVANCED TELEROBOT (NEATER)

The opportunity arose to use a remotely operated telerobotic system, the Nuclear Engineered Advanced TeleRobot (NEATER), to continue with the decontamination and decommissioning operations. This was funded partly by the Commission of the European Communities (CEC), who wished to support a proposal to demonstrate and test the capabilities of the telerobot for active decontamination and decommissioning tasks, and the Department of Trade and Industry via UKAEA's Corporate DRAWMOPS Directorate (CDD).

NEATER is based on an industrial telerobot model, modified for use in the high radiation environment found in nuclear decommissioning operations. The telerobot was used for size-reduction of items, monitoring, sorting and packaging of waste, and decontamination of surfaces. Special radiation tolerant tools were developed including an electric saw, pneumatic gripper, foam cleaner and vacuum equipment. A remote automatic tool changing station mounted on the telerobot trolley enabled tools to be changed in the cell.

Computer simulations were performed to demonstrate the feasibility of the proposed operations. Inactive trials were carried out on a full size mock-up of the cell and equipment to train operators and optimise the efficiency of operations, which prove invaluable.

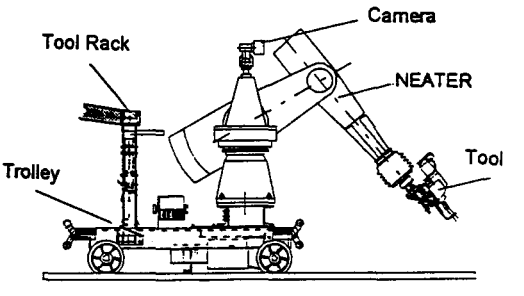

Diagram of NEATER mounted on the trolley.

Installation of NEATER began in July 1993 and the telerobot was conveniently deployed into the Main Cell on a trolley using the existing floor rails. A gaiter and strippable coating were applied to NEATER before it entered the cell to reduce contamination of the telerobot surfaces.

7. EXPERIENCE OF USING NEATER

The use of NEATER for remote decontamination and decommissioning operations is compared with the original plan of using traditional remote techniques (assuming five cycles of cleaning would have been sufficient).

- **Dose**

The collective dose for NEATER operations was 8 man-mSv, an estimated reduction of 90 man-mSv over traditional techniques. Such a reduction will be of increasing value in future projects as regulatory bodies set more stringent dose limits.

- **Waste**

The robot was used to monitor waste in the cell and segregate it for disposal to the appropriate route. The amount of Intermediate Level Waste (ILW) was kept to a minimum by decontamination techniques and by cutting out areas with high radiation levels. Low Level Waste (LLW) was packed directly into half-height ISO containers for dispatch to Drigg instead of intermediate handling and storage on the Harwell site. The elimination of size-reduction and double-handling activities reduced the collective dose for waste handling.

- **Telerobot Reliability**

There was no significant breakdown of the telerobot during an estimated 660 hours of operation. An availability of greater than 90% was achieved, which is considered to be excellent for such a complex piece of equipment.

- **Time Scales**

The schedule for telerobotic decommissioning was for 9 months of NEATER operations in the cell. Actual operations were completed in under 7 months, although the training time had been extended before installation. The actual time from start of planning the telerobotic operations to reach radiation levels compatible with man-entry was 20 months. This time scale can be compared with an optimistic estimate of 40 months for traditional techniques to reach the required levels.

- **Redeployment**

When the telerobot operations had been completed, NEATER was removed from the cell, the gaiter and strippable coatings removed and disposed of as LLW, and the telerobot monitored. Minimal contamination was detected indicating that the protective measures were effective and NEATER could be recovered and refurbished for further use.

8. DECOMMISSIONING END POINT

NEATER was withdrawn from the cell in February 1994 when background radiation levels were below 500 μSvh^{-1} and it was accepted that any remaining areas of high radiation were best decontaminated by manual operations.

Phases 4 and 5 'Man Entry into Main Cell' and 'Achievement of the Decommissioning End Point' were combined in an accelerated programme to complete manual decontamination of the cell, followed by dismantling of the cell structure. The Decommissioning End Point of complete removal of the HAHC, was achieved successfully by August 1994.

9. SUMMARY: BENEFITS OF USING TELEROBOTIC TECHNIQUES

- The HAHC project has provided a successful demonstration of telerobotic techniques for remote decontamination and decommissioning, yielding significant advances in telerobotic decommissioning methodology.
- Use of the NEATER telerobot in decommissioning the HAHC has achieved a significant saving in operator dose, cost and time, when compared with estimates for traditional techniques.
- The extensive use of a mock-up facility for optimisation of telerobotic operations and training of operators proved invaluable in the success of the project.
- The use of a remotely operated telerobotic system reduces the risks associated with projects where man-entry to a facility is restricted and hands-on techniques may not be possible to achieve the required end point.
- The HAHC project has provided a range of experience and awareness of the potential for remote telerobotic operations and it is recommended that telerobotic techniques be utilised in decommissioning operations in radioactive environments.

ACKNOWLEDGEMENTS

This work has been funded under the DRAWMOPS programme of the UK Department of Trade and Industry (DTI). The results of this work form part of the UK Government programme on decommissioning and radioactive waste management, but do not necessarily represent Government policy.

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PAPER TITLE Recovery of wastes from an old storage silo

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MAJOR SCIENTIFIC TOPIC NUMBER 4.7 (see page 7)

ABSTRACT (See instructions overleaf)

The dry silo at the Sellafield site, British Nuclear Fuels Limited [BNFL] was built in the late 1940s to receive solid intermediate level wastes from the first fuel decanning and reprocessing plants in the United Kingdom. It operated from the early 1950s until the mid 1960s, and the building has been under care and maintenance surveillance from then until the present. The building is a single walled structure, with six internal compartments and a roof superstructure that positioned a transfer flask over the compartment being filled. To minimise external doserates, additional shielding was placed on and near to the silo walls. BNFL has now decided to retrieve the contents of the silo. There has been investigative work to confirm the state of the structure and to confirm the nature of the contents, since contemporary tipping records were not detailed. A remotely operated method for retrieval has been developed and a processing building is to be erected adjacent to the existing silo. Here there will be some initial sorting, identification and processing of wastes to ensure that they are safe for transport to a further building where they will be encapsulated. The poster will outline some of the safety concerns addressed by the recovery facility design such as the potential for fire, the seismic robustness of the silo and the dose reduction measures necessary for construction in the existing doserates from the silo.

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PAPER TITLE Research on the disposal methods of uranium-bearing spoils in Eastern China

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 4.7

ABSTRACT (See instructions overleaf)

The spoils (produced by geological exploration,especially by tunnelling) in uranium deposit contain certain amount of uranium ore, it may have impact on the environment and pose a potential radiation health hazard to the public because of the presence of spoils piles, the amount of spoils at each site ranges from only residual contamination to 92 thousand tons in Eastern China. This paper develops a series of methods for the disposal of uranium spoils, such as applying an earthen cover to control release of radon because radon is the most hazardous constituent of uranium-bearing spoils, building drainage system to avoid water erosion and supporting to maintain the stability of spoils piles, which is based on the summarization of environmental impact analyses in 18 uranium deposits,various factors including health,resource, ecological environment protection and meteorological condition.

By the use of above methods that make the implymentation easily and less costly, we have finished the ultimate disposal of 21 spoils piles and gained ideal results that radon emission rate and gamma radiation are decreased significantly (achieving the demands of pertient standards),natural environment is restored fully, water erosion and misuse are avoided effectively .This paper also presents some data to evaluate the effect of remedial actions.

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PAPER TITLE Radium Luminising: The Continuing Implications

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MAJOR SCIENTIFIC TOPIC NUMBER 4.7 (see page 7)

ABSTRACT (See instructions overleaf)

The use of ^{226}Ra in luminising materials is well known. Even though this work ceased by the 1960s, it continues to have implications for British industry today. This paper looks at the financial and legal problems faced by landowners wishing to sell or develop sites that were formerly used for radium luminising purposes.

The potential liabilities, including the costs of decontaminating the land, are vast and must be considered fully when planning future use of affected land.

CHARACTERISTICS OF GENERATED AEROSOLS AND ESTIMATION OF AIR CONTAMINATION IN DECOMMISSIONING OF THE JAPAN POWER DEMONSTRATION REACTOR

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INTRODUCTION

A decommissioning program of the Japan Power Demonstration Reactor(JPDR) started in order to establish technical bases for future dismantlement of commercial nuclear power plants. Some techniques of reactor dismantling were developed. Various cutting methods were tested during the dismantling of the JPDR, which had variety of structures, materials and contamination level. JAERI developed an underwater plasma-arc cutting system which is capable of effective cutting highly activated reactor components and consequently reducing radiation exposure of workers. The extent of air contamination generated from underwater cutting had been unknown because of insufficient experiences of application of this technique. Methods for estimating the level of air contamination which results from the underwater cutting are necessary to evaluate radiation exposure of workers. A mathematical model of estimating air contamination level was proposed for aerosol generation during in-air and underwater cutting processes. The model was examined and some characteristics of the generated aerosols were investigated.

MATHEMATICAL MODEL FOR ESTIMATING AIR CONTAMINATION LEVEL

The air contamination is caused by the aerosols due to the cutting. An air contamination estimating model for the reactor dismantling is shown in Fig. 1. Formulas to calculate the total radioactivity of material generated during cutting operations were given by Smith et al.(1). The evaluation method of airborne activity resulting from radioactive surface contamination was developed by Shapiro(2). Both methods were applied to our model. The dispersion rate for in-air cutting was defined as activity ratio of material generated during the cutting to dispersed aerosols, and was given by:

$$\lambda a = Ca\,(Vs \cdot S + F)\,T / A \qquad (i)$$

where: λa = dispersion rate for in-air cutting

Ca = airborne concentration for in-air cutting (Bq/cm^3)

Vs = settling velocity of aerosols (cm/min)

S = area of floor (cm^2)

F = ventilation rate (cm^3/min)

T = working time in cutting operation(min)

A = total radioactivity of material generated during cutting operation

The dispersion rate for underwater cutting was defined as radioactivity ratio of material generated during underwater cutting to dispersed aerosols from the water into the air, and was given by:

$$\lambda w(d) = Cw\,(Vs \cdot S + F)\,T / A \qquad (ii)$$

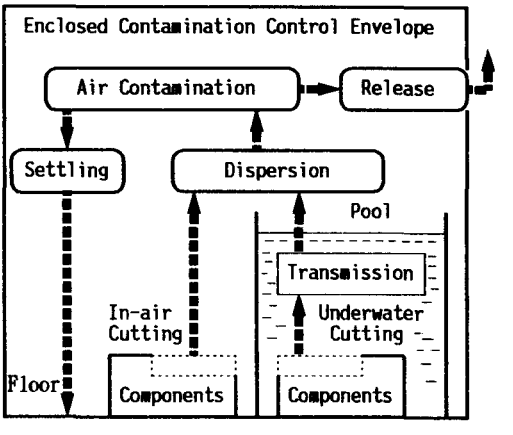

Fig. 1 Air contamination estimating model for the reactor dismantling

where: λ w(d) = dispersion rate for underwater cutting

Cw = airborne concentration for underwater cutting (Bq/ cm^3)

d = water depth at cutting operation (m)

Aerosol transmission factor (P (d)) was defined as:

$$P(d) = \lambda w(d) / \lambda a \qquad \text{(iii)}$$

MEASUREMENTS OF AEROSOLS

The radioactivity air concentration and size distribution of aerosols were measured to investigate characteristics of dispersed aerosols in a thermal cutting process. The reactor internals and pipes connected to reactor pressure vessel(RPV) were cut in the air with oxyacetylene and plasma-arc torches. The highly activated internals were cut contourly at water depths between 1 m and 8 m in the RPV and spent fuel storage pool with underwater plasma-arc cutting systems(Fig. 2). Typical surface or specific radioactivity of contaminated components are listed in Table 1.

Radioactive aerosols generated by the cutting were collected on glass fiber filters with a low pressure cascade impactor and a dust sampler. The aerosols were sampled at an exit of an enclosed contamination control envelope. Radioactivity on the filters were measured with a GM counter and a high purity Ge γ -ray spectrometer.

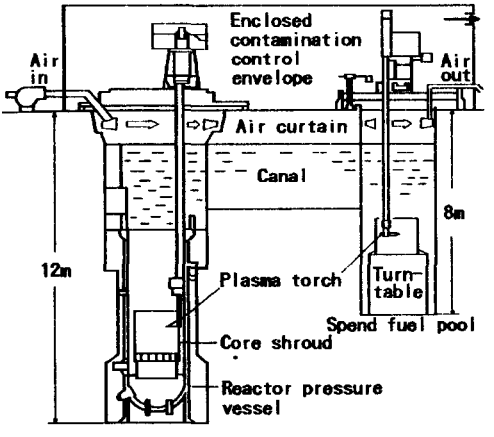

Fig. 2 Schematic view of cutting of reactor components in the water

Table 1 Radioactivity levels in major contaminated reactor components

Code	Component	Radioactivity
a	Pipes connected to RPV	2.9×10^{2} Bq/cm^2
b	Channel box	2.0×10^{4} Bq/cm^3
c	Core shroud	2.9×10^{7} Bq/cm^3
d	Bottom grid	3.0×10^{5} Bq/cm^3
e	Core support	1.5×10^{5} Bq/cm^3
f	Control rod	3.0×10^{7} Bq/cm^3
g	In-core monitor tube	2.0×10^{10} Bq/cm^3

RESULTS AND DISCUSSION

1) Aerosol size and size distribution

Figure 3 shows size distribution of aerosol generated by thermal cutting in the air and in the water. In the case of in-air cutting, bimodal distribution was observed, which has peaks at 0.05 μ m and 3 μ m of aerosol size. On the other hand, in the case of underwater cutting, only a single peak was observed at 0.05 μ m. This property may be explained by the fact that the big-size aerosols in bubbles were easily

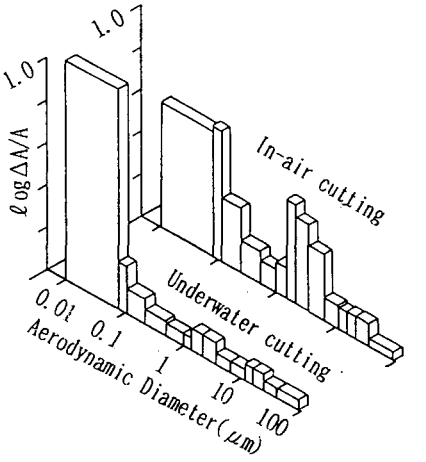

Fig. 3 Size distribution of aerosol generated by thermal cutting in the air and in the water

captured by the water.

2) Dispersion rate during in-air and underwater cutting

Dispersion rates for in-air cutting were calculated by Equation (i) using data measured in the pipe cutting. The results are plotted at 0 m of water depth in Fig. 4. The dispersion rate ranges from 3×10^{-4} to 3×10^{-2}. The geometric mean of the rate is 3×10^{-3}. Dispersion rates for underwater cutting were calculated also by Equation (ii) using data measured in the cutting of internals. The dispersion rate decreases exponentially with the water depth as shown in Fig. 4. The geometric means are 6×10^{-5} and 3×10^{-7} at 2 m and 8 m water depth, respectively.

3) Aerosol transmission factor

The typical transmission factors as given by Equation (iii) are 2×10^{-2} at 2 m, 1×10^{-3} at 4.5 m and 1×10^{-4} at 8 m depths in the water. These results show that the dispersion volume of aerosols depends on the water depth of cutting operation.

CONCLUSION

The characteristics of the aerosols generated from some structure-cutting operations were studied through the dismantling works of the JPDR. A model was proposed to estimate air contamination level due to aerosol generation. The transmission factors of aerosols in the water were obtained and found to decrease drastically with the water depth. The model and accumulated data will be helpful for accurate estimation of occupational radiation exposure and planning of reasonable dismantling methods in future decommissioning of commercial nuclear power plants.

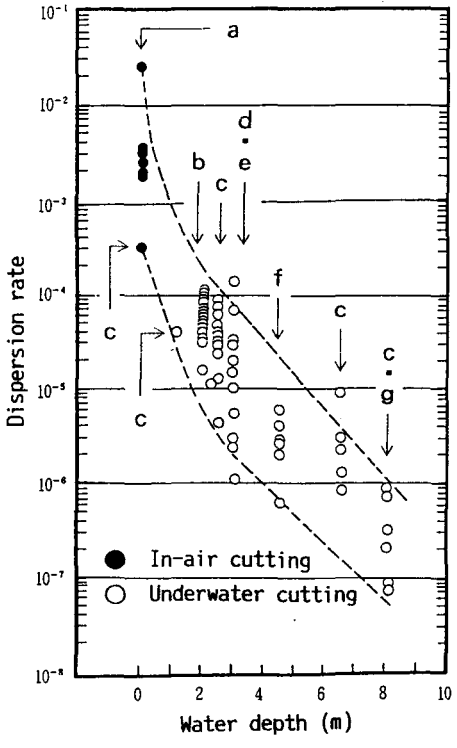

Fig. 4 Dispersion rates for in-air and underwater cutting (a-g: components in Table 1)

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This work was performed by JAERI under contract with the Science and Technology Agency of Japan.

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PAPER TITLE Use of a Single-Channel Analyzer and Sodium Iodide Detector to Resolve Quantification Difficulties During Remediation of a Radium-Contaminated Site

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MAJOR SCIENTIFIC TOPIC NUMBER 4.7 (see page 7)

ABSTRACT (See instructions overleaf)

Difficulties with using hand-held survey instruments to quantify radium concentrations in soil became apparent during a recent remediation effort. A major contributing factor was the presence of illite, a potassium-rich clay. Pure illite can have a ^{40}K concentration of 1.8 Bq/gm (49 pCi/gm). The 1.46 MeV gamma ray emitted by ^{40}K has a transition probability of 11%. In illite-rich soils, these gamma rays can obscure detection of the 186 KeV photons emitted from ^{226}Ra. This can, in turn, result in inaccurate field assessments of soil contamination.

Another difficulty encountered was radium contamination at some depth as well as soil overlaid with asphalt. The shielding properties of soil and paving material can lead to serious underestimates of anticipated remediation volumes.

Finally, proper manifesting of shipments from these sites requires activity estimates for each container. Performing such estimates can be done analytically, but in large containers, such as rail cars, it is difficult, time-consuming, and expensive to obtain and analyze samples.

It was found that use of a single-channel analyzer, in conjunction with a 2"x2" NaI(Tl) crystal, overcame most of these problems. An analyze window centered on the 1.76 MeV gamma ray emitted by ^{214}Bi progeny of ^{226}Ra eliminates interference from ^{40}K emissions. In addition, the 1.76 MeV gamma ray is very penetrating. This allows quantification to a depth of at least one meter in soil and within large shipping containers if soil homogeneity can be assured.

LEACHING AND ADSORPTION CHARACTERISTICS OF RADIONUCLIDES IN ACTIVATED CONCRETE WASTE

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1. INTRODUCTION

Full-scale dismantling of nuclear power plants will come in the near future. In the decommissioning, very large amount of activated concrete waste with very low-level radioactivity will be generated. One of the disposal methods of such kind of wastes is shallow land burial. Radiation exposure to the general public through groundwater is expected to be one of the main exposure pathways in case of the shallow land burial. Leaching characteristics of radionuclides from concrete waste are very important in assessing the exposure due to the shallow land disposal. There are many reports on leaching mechanism focusing on the solidified low-level waste, but not on activated concrete waste (1). Leaching is caused by diffusion, dissolution, ion-exchange, corrosion and so on. The leachability also strongly depends on the environmental conditions, such as form of waste, waste materials, substances dissolved in water contacted with the waste and nuclides of interest. In decommissioning of a nuclear power plant, concrete wastes will be vary in forms such as block, debris, granule and powder. It is, therefore, very difficult to estimate leaching of radionuclides from concrete wastes precisely.

The objects of this study are to obtain diffusion coefficients and distribution coefficients of elements for leaching from activated concrete wastes, to estimate the leach rates of radionuclides from activated concrete wastes by three different models, and to evaluate the leaching models for disposal of activated concrete wastes generated from decommissioning of nuclear power plants.

2. EXPERIMENTAL METHOD AND MATERIALS

2.1 Measurement of effective diffusion coefficients, D_{eff}

The effective diffusion coefficients (D_{eff}) were measured by two methods as follows;

Method 1: Leaching test procedures were similar to the IAEA method(2). A concrete specimen with 2 cm in thickness and 8 cm in diameter (230 g) was activated by a research reactor, and was placed into the distilled water container (500 cm^3). Samples of leachant were withdrawn and analyzed at designated intervals by gamma-spectrometry.

Method 2: Leaching test procedures were the same as IAEA method(2). 10 specimens of non-activated concrete with same dimension as above were placed into water container (5000 cm^3). The concentration of nuclides in leachant were measured by activation analysis.

The leach test results were expressed by a plot of the cumulative fraction of radioactivity leached from the specimen as a function of the total time of leaching thus, $\Sigma a_n/A_0$ versus $t^{1/2}$, where a_n is amount of nuclides of interest leached during the leach interval (Bq or mol), A_0 is total amount of species in test specimen (Bq or mol), t is total elapsed time since start of leaching (day). D_{eff} was delived from the slope using following equation(3): $(\Sigma a_n/A_0)(v/s)=2(D_{eff}/\pi)t^{1/2}$, where s is geometric surface area of specimen (cm^2), v is volume of specimen (cm^3).

2.2 Measurement of distribution coefficients for leaching, K_{dl}

K_{dl}s were measured by different two methods as follows.

Method 1: 50 g of non-activated concrete granule (1.2 - 2.4 mm in diameter) and 200 ml of distilled water was mixed and elapsed for 400 days. The concentration of nuclide in the solution and the concrete was measured by activation analysis. K_{dl}s were calculated by the equation, $K_{dl} = C_m / C_l$, where C_m: concentration in concrete (ppm), C_l: concentration in solution (ppm).

Method 2: 50 g of activated concrete granule(1.2 - 2.4 mm in diameter) or powder(<150 μm in particle diameter) and 200 ml of distilled water was mixed and was elapsed for 400 days. The radioactive concentrations of radionuclides in the solution and the concrete were measured. K_{dl}s were also calculated the above equation.

2.3 Measurement of distribution coefficient for adsorption, K_{da} (4)

10 g of non-activated powder(<150 μm in diameter) concrete were mixed in 200 ml of solution containing ^{134}Cs, ^{60}Co, ^{45}Ca and ^{152}Eu and were shaken for 24 hours. The concentration of the radionuclides in solution were measured. Kda was calculated by the equation: $Kda = (C_0-C)V_S/(CM)$, where C_0: initial concentration in solution (Bq/ml), C: concentration in solution after reaction (Bq/ml), V_S: volume of solution (cm^3) and M: mass of concrete(g).

All samples for leaching and adsorption tests were served from bioshield concrete of Japan Power Demonstration Reactor (JPDR) which is now under decommissioning.

3. RESULTS AND DISCUSSION

Effective diffusion coefficients, Deffs and distribution coefficients Kdl and Kda are listed in the Table 1. Only the Deff of Cs was obtained by Method 1 due to the detection limit. On the other hand, many Deffs of elements were obtained using the method 2. There seems no discrepancies in Deff of each element between two methods. This means that there is no significant difference between the diffusion mechanism of activated elements and that of non-activated elements. The Deffs were Ca > Cs > Co in order of magnitude. Deff of Eu seems to be far smaller.

There were no significant discrepancies of Kdls due to measurement methods. The Kdls measured by both the method 1 and 2 were Ca > Cs > Co > Eu. Kdls for powder concrete were about 10 times as large as that for granular concrete. A relation between Deff and the inverse of Kdl measured by method 1 showed that the Deff increases with decreasing Kdl linearly on the log-log plot. Therefore, these results indicate that a diffusion mechanism is a dominant mechanism of leaching of nuclide from activated concrete waste.

The Kdas measured were Co > Eu > Cs > Ca. The Kdas measured were one order of magnitude smaller than distribution coefficient of equilibrium, Kdls.

In decommissioning of a nuclear power plant, concrete wastes will be vary in forms such as block, debris, granule and powder. It is, therefore, very difficult to estimate leaching of radionuclides from concrete wastes precisely. Following simplified estimation models of leaching are proposed to assess the leaching: a) an instantaneously leach model, the most conservative model, assuming that all activity leaches immediately after disposal, b) an equilibrium model assuming that there is an equilibrium between waste and leachate, this model is apply to the waste in form of granule and powder, and c) a diffusion model which will be apply to the waste in form of blocks.

To evaluate the leach models for disposal of activated concrete wastes generated from decommissioning, leaching amount were calculated by three models using measured data under simplified conditions as follows:

a) Instantaneously leach model: $L = CwV\rho/Q_0$, where L is a leach ratio defined by the ratio of amount of nuclide leached from waste to the initial total amount of nuclide in the waste, C_W is concentration of nuclides in the waste (= 1 Bq/g), V is the volume of waste (= 10^6 cm^3) and ρ is the density of waste (1.0 g/cm^3), and Q_0: initial amount of activity in the waste ($=CwV\rho$).

b) Equilibrium model (5): $L = Kd^{-1}CwPA/Q_0$, where Kd is distribution coefficient ,Kdl or Kda (cm^3/g), P is precipitation (= 10^2 cm/y), A: precipitation area for waste (= 10^4 cm^2).

c) Diffusion model (3): $L = 2(S/V)(Dt/\pi)^{1/2}$, where D is effective diffusion coefficient (cm^2/day), S is total surface area of waste (= $6x10^4$ cm^2, assuming a cubic form waste with side of 1m), t is duration of leaching (= 365 days).

Table 2 shows the leach ratio calculated by different three calculation models under the above conditions. The instantaneous leach model is not based on real physico-chemical process and is the most conservative. The instantaneous leach model is, however, useful when there are no data used for other sophisticated models or the evaluation is for tentative purposes. When the waste form is granule or powder, the equilibrium model is more reasonable than the instantaneous model. Table 2 indicats that the leach ratios calculated by the equilibrium model are several order of magnitude smaller than that by the instantaneous leach model. For example, leach ratio of Co from powder concrete was $3x10^{-4}$ times as large as that by instantaneous leach model.

Estimation of leaching using the distribution coefficients for adsorption (K_{da}) is an alternative method when distribution coefficients for leaching (K_{dl}) are not available. The results, however, shows that the leach ratios calculated using K_{da} are one order of magnitude larger than that using K_{dl}.

Table 2 shows that the leach ratio of elements from cubic concrete waste estimated by diffusion model was 10^{-4} to 10^{-6} times as large as that by the instantaneous leach model.

4. CONCLUSIONS

1) Leachability of nuclides from an activated concrete is Ca > Cs > Co > Eu.
2) The leaching amounts from granular concrete waste estimated by equilibrium model are 4 to 5 order of magnitude as large as those by the instantaneous model.
3) The leach amounts calculated using Kda are one order of magnitude larger than that using Kdl.
4) The Leaching amountsestimated by diffusion model are five order of magnitude as large as those by the instantaneous leach model.

Table 1 Effective diffusion coefficients and distribution coefficients

Element	$D_{eff}(cm^2/day)$		$K_{dl}(cm^3/g)$			$K_{da}(cm^3/g)$
	Method 1	Method 2	Method 1 (powder)	Method 1 (granule)	Method 2 (granule)	(powder)
Cs	$4x10^{-10}$	$2x10^{-10}$	$3x10^{3}$	$1x10^{4}$	$1x10^{4}$	$5x10^{2}$
Co	*	$6x10^{-12}$	$1x10^{4}$	$2x10^{4}$	$5x10^{4}$	$4x10^{3}$
Ca	*	$3x10^{-8}$	-	-	$1x10^{3}$	$3x10^{2}$
Eu	*	*	$>5x10^{4}$	$>1x10^{5}$	$>2x10^{5}$	$2x10^{3}$

*: lower than detection limit, -: not measured

Table 2 Leaching rates estimated by different models

Element	Leach ratio*				
Calculation Model	Instantaneous Leach model	Equilibrium Model**	Equilibrium Model***	Equilibrium Model***	Diffusion Model
Dimension of waste		Powder	Powder	Granule	Cubic
Cs	1.0	$2x10^{-3}$	$3x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$
Co	1.0	$3x10^{-4}$	$1x10^{-4}$	$2x10^{-5}$	$3x10^{-6}$
Ca	1.0	$3x10^{-3}$	-	$1x10^{-3}$	$2x10^{-4}$
Eu	1.0	$5x10^{-4}$	$<2x10^{-5}$	$<5x10^{-6}$	-

* The ratio of amount of element of interest leached from a concrete waste to initial total amount of element in the waste.
** Distribution coefficients for adsorption, K_{da} are used.
*** Distribution coefficients fro leaching, K_{dl} are used.
- Not calculated due to lack of data.

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This study was carried out under a contract with the Science and Technology Agency of Japan.

COLLIMATED IN-SITU GAMMA SPECTROMETRY: A NEW METHOD FOR FAST CLEARANCE MEASUREMENTS OF LARGE AREAS OR BUILDING STRUCTURES OF NUCLEAR FACILITIES UNDER DECOMMISIONING [1]

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INTRODUCTION AND MOTIVATION

The ideas in the late sixties which led to the use of bare spectrometric radiation detectors like NaI(Tl), Ge(Li) or nowadays high-purity germanium detectors (HPGe) for field measurements (in-situ gammaspectrometry) were to get easy and rapid informations about the radiological state of outdoor grounds after nuclear weapon tests or to estimate dose rates created by natural radioactive nuclides in the soil. In this cases, it was assumed, that there was no disturbance of the source-detector geometry for many hundred square meters around the detector. After the nuclear accident in Chernobyl in 1986 these advantages focussed strong scientific interest at the in-situ technique and in 1993 it was established in the german regulatory for immision surveillance after significant radioactive emissions. For unrestricted release in decommissioning "in-situ gammaspectrometry may be the only method of achieving validation of the release criteria, particulary for large areas outside the buildings" [1].

In a late phase of the decommisioning of a nuclear power plant all components containing a significant inventory of radioactivity are removed, leaving large surfaces with often poorly known contamination levels. Taking in account only buildings of restricted areas in the next 50 years in Germany 7E3 Mg activated and 5E6 Mg contaminated concrete must be released from facilities under decommisioning. Before the ground and the building structures of the facility can be conventionally pulled down, the remaining radioactivity must be determined in order to check the radiological relevance of the concerned part of the plant and to decide the possible path of material release. The basis of assessment may be the so-called "10 μSv-concept" [2].

Today the mainly used strategies are analysing samples, taken from the surface or measuring the surface activity via large proportional counters. In both cases we meet severe systematic problems.

1) A commercial contamination monitor meets a minimum detectable activity of 0,1 Bq/cm² Co 60 or Cs 137 on a surface two to ten-times faster than a collimied in-situ spectrometer. On the other hand shielding layers coming from decontamination coatings or migration of the radionuclides in the surface lead to relative small errors performing in-situ measurements whereas contamination monitors are completely inappropriate in the most cases to determine the remaining activity.

2) Taking and analysing samples leads for the single sample to more exact results than in-situ measurements caused by the not completely known source geometry and the measuring time which is in the latter case normally much shorter because of the large number of measurements.

The main problem of the analysis of samples in the laboratory is the estimation of the representativity of a collective of samples. Comparing the results of in-situ measurements and laboratory analysis we noticed substancial differences which could be attributed to strong inhomogenities in the spatial distribution of activity. These were not overcome by sampling.

The importance of this problem can also be shown theoretically. Ferguson [3] determined the number of samples to meet a defined statistical significance for finding an existing contamination. As an example ~200 samples are necessary to find a circular contamination with a diameter of 35 cm on a relative small area of 10 m² with a safety of 95%.

[1] This work was financially supported by an international EC R&D-contract ("Measurement of Spatial Distribution of Contamination on Concrete Surfaces and Build-up of Dusts for the Improvement of Scenarios for Material Release" / CEC Contract FI2D-CT94-0086), which is coordinated by us.

Errors arising from this sort of problems do not appear by measuring with a collimated high-purity germanium detector directly in the regions of interest ("in-situ"). There are only two problems left.

1) The vertical distribution of a contamination may be not known well enough. This fact creates a uncertainity which can reach a factor of two. On the other hand this systematic error may be solved and is furthermore small, compared to errors which can affect the quality of the results of traditional techniques.

2) A formal problem e.g. in the german radiation protection ordinance is, that the value of a surface contamination has to be achieved by a measurement averaging over 100 cm^2. Such an averaging area is obviously not adequate for clearance measurements of thousands of square meters of probably contaminated walls, floors, roofs or soils. At the moment this subject is discussed in Germany.

Apart from this the collimated in-situ gamma spectrometry allows to get very fast informations about the radiological state of large areas and -if one takes in account the high quality of these informations- for low costs.

DEVELOPMENT OF AN COLLIMATED IN-SITU SPECTROMETER

Despite the strong worsening effect to the minimal detection limits, it was obvious that for using such a device for clearance measurements it had to be collimated to reduce the field of view. Reasons therefore are that

- possible contaminations must be locateable,
- there must be a well defined averaging area (e.g. some m^2), and
- contaminations outside the measured area must be effectively suppressed.

So we designed and constructed a prototype of a high-resolution in-situ spectrometer on the basis of 41% p-type HPGe-detector, surrounded by an optimised collimator made out of brass and a low natural activity lead-bismuth-tin-alloy. As a compromise between portability and shielding effect a attenuation mass per area of ~53 g/cm^2 was realised for radiation which reaches the collimator in an angle of incidence parallel to the investigated surface. Furthermore the detector is shielded for radiation from the rear hemisphere. Depending on the collimator used and the height of the detector above the surface our spectrometer averages over areas between 0.4 and more than 10 m^2 in one measurement.

To get quantitative values of contaminations the system had to be calibrated. This needs much more work than for gamma spectrometers in laboratory, where reference sources with relative small volumes for a reduced set of geometries and chemical compositions exist. Performing in-situ gamma spectrometry one has to deal with large sources and any horizontal and vertical activity distributions. They can only be simulated numerically or experimentally. We use three approaches to verify the calibration factors between the full-energy-peak count rate and the specific activity level;

- a modified standard method for uncollimated systems suggested by Beck et al [4] in 1972,
- an experimental calibration by recording a great number of spectra with small sources in different positions and superimposing them as a function of the source geometry and
- a complete numerical simulation modelling the source and the entire detector-collimator-system

All commonly used yield curves with calibration factors based on the standard method have already been verified experimentally. According to our current experience, the two yield curves generally do not deviate from one another by more than 10% in the region of higher photon energy (662 keV and above). Differences of more than 30% have not occurred below 186 keV. These tests make the individual calibrations very reliable, as the two methods are in no way related to one another and the occurrence of identical systematic errors is thus ruled out.

PERFORMANCE OF THE SPECTROMETER

It could be shown that our calibrated prototype is able to meet the maximal activity limits for unrestricted release in Germany in some minutes of measuring time, if the proportionate activity of a strong gamma emitter like Co 60 or Cs 137 is sufficiently high in the nuclide vector. Examples for minimal detectable activities are:

- Under rigid conditions (maximum collimation, some migration of the nuclides in concrete) and normal dose rates due to natural nuclides detection limits of 800 Bq/m^2 for Co 60 and 1400 Bq/m^2 for Cs 137 in three minutes are reached.

==> A contamination with a nuclide vector containing only 20-30% of these nuclides can be detected with the requirements of the german radiation protection ordiance.

- For the relatively difficult to measure activity of U 238 (via Pa 234m) a detection limit of 150 Bq/kg is reached after 15 minutes.

  ==> Uranium can be measured on piles with any enrichment.

  ==> The observance of exemption values for deposition of waste contaminated with the more difficult to measure nuclides of the nuclear fuel cycle can be proved (f.e. [5])

- In the same time a surface contamination of 500 Bq/m² U 235 is detectable and can be discriminated from the natural Ra 226 and U 235 186 keV background in concrete containing 1 Bq/kg U 235.

Furthermore disturbing radiation from outside the interesting field of view is strongly suppressed. For low and medium energy gamma radiation (<700 keV) from nuclides distributed vertically and horizontally homogeneous in soil 90% of the total photon flux appear from an angle of ±48° relative to the detector axis in the collimated case. Without collimator this value would rise to 82°, so it would be impossible to localise a source.

APPLICATION OF THE SPECTROMETER

Aside from the development of such a spectrometer we made experiencies with our prototype inside the restricted area and outside the buildings of several nuclear power plants unter decommissioning like KKN, KRB-A and VAK in Germany and G 3 and RAPSODIE in France by painting a grid over the area of interest and measuring at each grid point. Measurement campaigns in facilities of the nuclear fuel cycle were performed in NUKEM-A, HOBEG and WAK in Germany.

Up to now with this campaigns new knowledge about the lateral distribution of contaminations, the variability of its composition or -by comparing with other measuring techniques- their systematic uncertainities could be quantified. The description of the results of these campaigns would exceed the scope of this compact. As examples we

- mapped the short distance variability of the Chernobyl Cs 137 fall-out by scanning 1500 m² grassland completely with 3 m² surface area per single measurement,
- showed for a contaminated floor for one room in the restricted area of a nuclear power plant that the Co 60 / Cs 137 ratio was completly indefinite (variation over more than three orders of magnitude) and
- investigated large deviations in the measured surface activity via proportional counter and in-situ for several surfaces with different histories and contamination scenarios.

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DELICENSING OF NUCLEAR SITES IN THE UK

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INTRODUCTION

In the UK, the operation of nuclear facilities is regulated under the Nuclear Installations Act 1965. This Act requires the issue of a licence for a number of prescribed activities. Once a licence is issued the licensee's period of responsibility continues until the Health and Safety Executive (HSE) gives notice in writing that there has ceased to be any danger from ionising radiations. This requirement applies whether the licensee wants to remove part or all of the site from his responsibility. Both of these processes are often referred to as delicensing. As with any other licensing function under the NIA65, the Nuclear Installations Inspectorate (NII) carries out this process on behalf of HSE.

Since the inception of the original Act in 1959 there have been a number of occasions where licensees have sought to end their period of responsibility or exclude parts of the site. As current facilities age and enter the decommissioning phase of their operations it is likely that there will be an increase in the number of requests NII receive to end the period of responsibility. It is therefore important that the process under which delicensing takes place is understood and that there is a suitable mechanism on which a judgement can be made regarding the suitability for the period of responsibility to be ended. This paper sets out some views on this process from a regulatory standpoint and discusses some practical aspects of how suitability may be demonstrated.

REGULATORY PROCESS

In the UK the licensee is responsible for safety until the period of responsibility ends. In seeking to end the period of responsibility similar procedures to those undertaken for an operational plant are expected. The work will have to be planned and managed safely and the regulator would expect that appropriate safety cases for the various stages leading to this end point are prepared. Before delicensing can proceed a safety case would need to be submitted for NII's consideration. Such a case will need to address a number of aspects, which could typically include:

History and use of the land

Type of work undertaken

Arrangements for preserving records, appropriate to the licensee's obligation under the licence, i.e. records relating to site operations which may be used as evidence in any claim resulting from such operations.

The state and maintenance of the plant in the shutdown condition.

The necessity for any dismantling of the plant or buildings and for any subsequent cleaning-up operations.

The management and disposal of radioactive waste.

Reason for excluding from a nuclear site licence.

Evidence that no leakage of radioactive matter has occurred into the land, or if such has occurred how it was dealt with.

Evidence that no radioactive contamination has been deliberately sealed or fixed into structures that will remain after the site is declared free from danger.

Information concerning natural background radiation in the vicinity.

Site survey information that compares measured levels with local background.

NII will be looking to establish that, to the degree necessary for the purposes of the Act, remaining levels of radioactivity are indistinguishable from those which would naturally occur in the area.

TECHNICAL EVALUATION

The previous discussion sets out the type of information that NII would expect the licensee to provide for its consideration. It is clear that there are a number of aspects against which such a safety case can be assessed and there will be site specific features. There will therefore be a mixture of qualitative and quantitative judgements depending on the site. An important aspect is to be able to interpret the monitoring undertaken and to form a judgement on whether it is appropriate for the circumstances and whether further monitoring is reasonably practicable. Monitoring approaches may include direct radiation monitoring and soil sample analysis. To help NII's judgement some technical investigation of the available techniques for the former and the detection capabilities has been undertaken in conjunction with NRPB. All currently available instrumentation has an inherent instrument background. This limits the level of radioactivity which can be measured with any degree of certainty and sets a detection threshold. In addition, there is a limit to the discrimination between a response due to natural background and any contaminant found. These are generic aspects which will apply to all delicensing cases. There is a very real practical need to establish whether it is possible, with current instrumentation, to easily monitor low levels of environmental radioactivity and to know the limitations of the measurements.

RESULTS

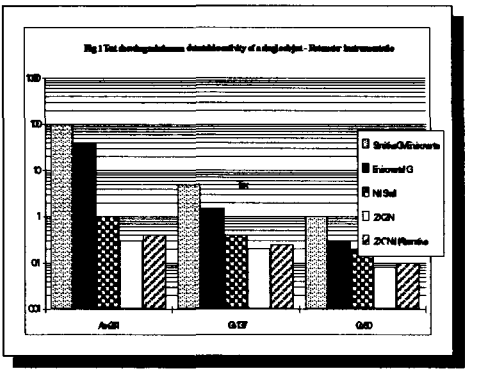
A range of possible monitoring instrumentation is available in the UK. Some commonly available instruments currently in use for directly monitoring levels of radiation were selected with the view to performing the following tasks;

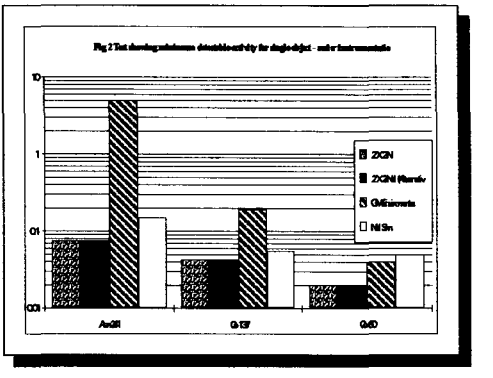
Determine the minimum detectable activity assuming a single object in the ground.

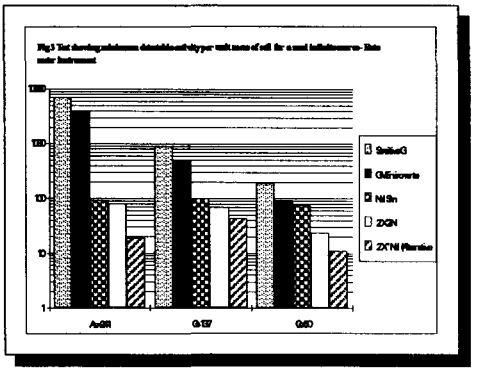
Using instrument in a scaler mode with a 30 second counting time assess the minimum detectable activity.

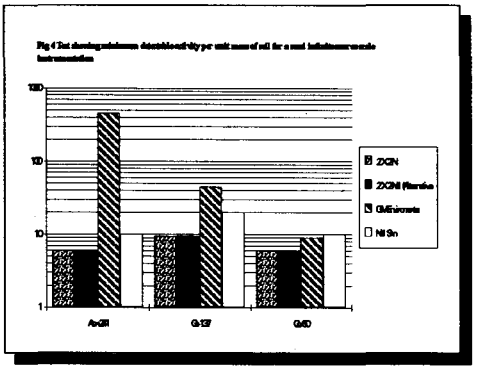
Relate each minimum detectable activity to a concentration in soil.

For each task a range of single nuclides was assumed (Co-60, Cs-137 and Am -241)









The results show that in general a lower level of detection can be achieved with instrumentation in a scaler mode. Figures 1&2 indicate that some instrumentation has the capability to measure levels at 10 M Bq in ratemeter mode and 1 MBq in scaler mode for a number of nuclides. These results illustrate the level of activity, in a single object, above which there is a high degree of confidence that the measurement is due to presence of radioactive material. (Full details of the arrangement and tests are provided in ref 1.) It should be pointed out that this is an implied level of activity and could be due to natural radioactivity or the presence of artificial radionuclides. To clarify this would require further determination, probably using gamma spectroscopy equipment to compare with the natural background spectrum. The initial monitoring is a first step and provides valuable information to enable regulatory decisions about the need for and type of more detailed monitoring programmes.

Turning to figures 3 and 4, they provide information which relates to what the instrumentation response means in terms of a minimum detectable activity in the ground. This is useful in terms of comparison with criteria under other UK legislation and again for comparison with natural background levels for the area. The results illustrate that it appears to be possible to monitor for levels of 1 Bq per gram for a range of nuclides. The investigations have shown that it is possible to carry out such measurements in reasonable timescales using standard equipment.

Some scoping work was done to extend this technical evaluation to examine the effects of shielding due to normal building materials and depth of buried objects. The difficulties of measuring for Plutonium was also investigated as was the use of portable gamma spectrometry equipment as part of the overall task. The latter aspect can be important due to the variation in the natural radioactivity content in building material. The basic conclusion of this scoping work was that for artificial isotopes it is reasonable practicable to achieve low limits of detection.

CONCLUSIONS

The process by which a site can be delicensed requires a range of information to be provided to the regulator. The experimental work undertaken has shown that it appears to be technically possible to monitor low levels of radioactive material using portable instrumentation which can subsequently be compared with current regulatory levels for below regulatory concern and natural background. The paper has also illustrated that judgements on delicensing need to be made in empirical and quantitative terms.

The views expressed in this paper are those of the authors and do not necessarily represent those of their respective organisations.

The authors wish to gratefully acknowledge the assistance of Mr M Bacon and Mr I Robinson of NSD for their assistance in preparing and reviewing the paper.

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RADIATION MONITORING AND QUALITY ASSURANCE IN DECOMMISSIONING OF NUCLEAR FACILITIES

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1. Introduction

Radiation monitoring in decommissioning of nuclear facilities is a major part of the entire decommissioning engineering. Quality of radiation monitoring has a direct influence on the quality of decommissioned work. Decommissioning due to poor engineering quality will result in not only great economic loss but also poor social impact. Therefore, it is not difficult to understand in decommissioning of nuclear facilities the radiation monitoring quality is of special importance. The paper is intended to present the characteristics of radiation monitoring in nuclear facility decommissioning, quality assurance system, quality assurance requirements and the related issues.

2. Characteristics of radiation monitoring in nuclear facility decommissioning

Radiation monitoring in decommissioning of nuclear facility involves the dismantlement, decontamination and cleaning of equipment, facilities and buildings as well as radioactive waste disposal, including source term survey, decommissioning monitoring, termination survey, and verification inspection survey[1-4], etc. The work amount of various monitoring categories varies greatly with the different stages of decommissioning. Decommissioning monitoring requires the large amount of work to be done, lasting long time throughout the full process of the decommissioning. the termination and verification inspections are to check and certificate the quality of decommissioning monitoring. So issues on radiation monitoring quality discussed in the paper is mainly focused on decommissioning monitoring.

3. Radiation monitoring quality assurance system

Radiation monitoring quality assurance system is shown in Fig. 1. The decommissioning project is contracted by licensee or contractor(s). The licensee sets up Decommissioning Office (DO) and Quality Supervision Office (QSO). Radiation Monitoring Leading Group (RMLG), led and supervised by decommissioning office, include Radiation Monitoring Group (RMG) and Quality Assurance Group (QAS), which is also guided and supervised by the QSO. The RMG consists of decommissioning operating monitoring, radiation protection monitoring, and environmental monitoring, which are also guided and supervised by the QAG according to decommissioning plan. The QAG is responsible for preparation of monitoring program, preparation and provision of quality assurance documents, organization of personnel training, selection of monitoring instruments, determination of measuring and analytical methods, determination of sampling and sample preparation methods, review of monitoring data and report. The QAG is also responsible for organization of post-decommissioning termination radiation survey and providing of termination survey report. These monitoring groups separately exert interior and exterior quality control measures according to quality control requirements. Termination verification inspection survey for decommissioned nuclear facilities should be undertaken and carried out by the inspection survey group entrusted by competent authority.

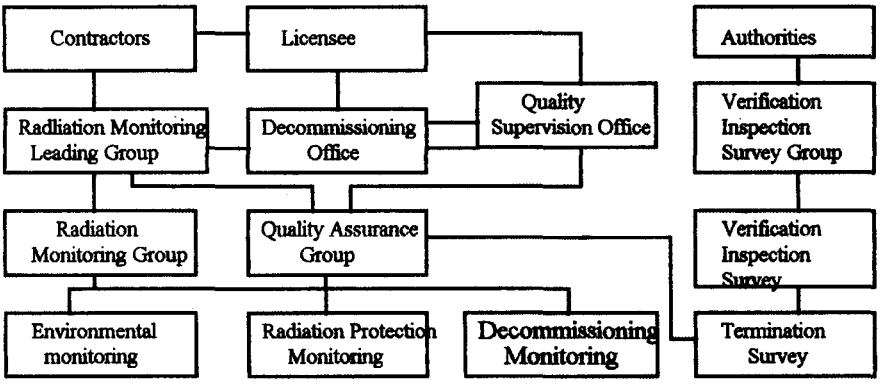

Fig.1. Radiation monitoring Quality Assurance System

4. Quality requirements for radiation monitoring

It is the final quality target of radiation monitoring during nuclear facility decommissioning to enable the residual radioactive contamination level of decontaminated equipment, facility, building, field, and site to meet the standards required by decommissioning project, and to identify these items can be open for the public to use on a restricted or unrestricted basis. After decommissioning, radioactivity inventory and desposal must be clear. Thus, in preparation of nuclear facility decommissioning programm, the quality requirements for the process of monitoring should taken into account.

4.1 Quality requirements for source terms survey

Source terms survey at the preparatory stage of nuclear facility decommissioning mainly involves two factors: residual radioactivity and radioactive contamination. The former means the amount of radioactive materials left within the decommissioned nuclear facility while the latter means the extent to which the decommissioned object is contaminated and the related contamination amount. In principle, the quality requirement for source term investigation is to give accurate or comparatively precise magnitude for the large while contamination source survey is to discriminate contaminated objects, types, and extent, giving the estimates of contamination.

4.2 Quality requirements for decommissioning monitoring

4.2.1 Training

Scope of training includes basic features and properties of decommissioning project; types, distribution and magnitude of contaminating nuclides; underlying know-how and regulations of radiation protection; contents and requirements of decommissioning monitoring; control limits of nuclides of concern; monitoring methods; performance of monitoring instruments; operating procedures; sampling requirements; analytical methods; target and requirements of quality assurance; format of records and monitoring report.

4.2.2 Selection and use of instruments

Detectable lower limit of instrument should meet the specified decommissioning standards. Probe surface area of surface contamination monitor should perferly be same with that specified by decommissioning standards.

Reference standards used for calibration of monitoring instrument and sample analysis should be traceable to the national standards. All of monitoring and analytical instruments should be confirmed to be steady and reliable.

4.2.3 Quality control in decommissioning monitoring

Quality control in decommissioning monitoring requires conducting full-range scanning mohitoring with view to not omitting hot points and above-standard points. So the extent of decommissioning monitoring should properly larger than the contamination extent indicated by source term investigation. In the process of monitoring, decontaminated area and volume of items should be given, and magnitude of radioactivity decontaminated should be estimated in a timely manner, so as to sort contaminated parts and determine where the decontaminated contaminants are located. Parameters, such as types of nuclides and inventory of radioactivity, should be presented during packaging of radioactive waste into drums.

5. Implementation and control of standards

In decommissioning of nuclear facility, the objective of radiation monitoring is to identify whether or not the residual radioactivity level on the decontaminated or cleaned equipment, facility, building and field meet the specified standards or limits. In implementation and control of standards, there would be an issue that is: what is above standard or above standard point. Because any of specified standards or limits is a absolute value. This can be reflected in instrument reading, or measuring data or analytical results of sample, where there will exist a variety of errors. There should be a confidential interval in implementing standards: $A \pm k\sigma$, where A is a number value specified by standards, σ is total error of measurement and k is a coefficient related to confidential level. In decommissioning radiation monitoring, standard can be implemented in terms of $A \pm k\sigma$, $k=1$, with relative error 30%. This means that standards is implemented in terms of 2/3. However, for termination survey and verification inspection survey, standard should be implemeted in terms of $A \pm k\sigma$.

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GENERAL PROBLEMS OF RADIATION MEASUREMENTS IN DECOMMISSIONING OF NUCLEAR FACILITIES

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1. Introduction

Radiation measurement in decommissioning of nuclear facilities is a major part in the entire decommissioning work, being involved in source term survey prior to decommissioning, decommissioning monitoring, termination survey and verification inspection survey. These measurements are dependent upon the nature of decommissioning engineering at different stages. They have their own respective specialty and complexity in spite of much similarity with each other[1]. Additionally, radiation measurements in decommissioning of nuclear facilities include radiation protection monitoring of personnel and environmental monitoring. This paper is confined to monitoring of the decommissioned objects of nuclear facility.

2. Source term survey prior to decommissioning

2.1 Aims and contents of source term survey

Source term survey before decommissioning belongs to preparatory work of decommissioning, and is to provide basis of preparation and implementation of decommissioning programme by determining the state of a nuclear facility to be decommissioned, such as source location, nuclide type, contaminated object, scope of contamination and determining radioactivity inventory. To achieve these objectives, the contents of source term survey include (1) major workshop to be decommissioned, accountability, classification and registration; (2) investigation and estimation of residual radioactivity magnitude; (3) level of contaminated workshop, equipment, parts, place, and scope, and their estimation; (4) radiation field distribution in nuclear facility: (5) accounting and registration of licensee-own radiation sources, (6) determination of decommissioned objects and decommissioning work area.

2.2 Source term survey method

There are two methods for source term survey: (1) estimation according to technological process and (2) estimation according to radiation measurements. The magnitude of residual radioactivity within nuclear facility, which is related to technological process, should be given by operating department, but in some circumstances, accurate data are difficult to be obtained from the operating department. Thus the needed data can be acquired through radiation measurements.

Contamination source term survey is much larger than residual radiation investigation in area. In addition to installations, equipment, parts, and building plant, and site, the former involves environmental issue, varying with type of nuclear facility. If there exist many types of contamination nuclides and complex equipment, and even more all of currently developed measurement methods are employed, there still would be some problems difficult to be tackled with that are needed to be studied according to decommissioning object and given the optimal investigation programme.

2.3 Estimation of radioactivity magnitude

As indicated previously, source term investigation involves two magnitudes: the residual radioactivity and radioactive contamination within facility. Magnitude of residual radioactivity is larger generally than that of radioactive contamination. Both are different in order of magnitude.

Source term survey should firstly focus on the accurate estimation of residual radioactivity and secondly on radioactive contamination. Currently it has not been specified what accuracy is needed in estimating radioactivity magnitude in source term survey.

3. Decommissioning monitoring

3.1 Objects and tasks of decommissioning monitoring

Decommissioning monitoring is the continuation of radiation measurements in source term survey. The primary objects of decommissioning monitoring are equipment, parts, building plant, field, and other location and position contaminated during dismantling. The task is to indicate decontaminating and cleanup position and extent to ensure them kept below control level after decontamination. Heavily contaminated parts should be estimated for their radioactivity level, together with their disposal method and site. Of course, decommissioning monitoring should incorporate radiation protection monitoring of personnel. Dose to individuals should be controlled below the specified limits and as low as reasonably achievable.

3.2 Decommissioning monitoring method.

Decommissioning monitoring must be made with attendance of decommissioning personnel. If possible, Decommission personnel may be equipped with monitoring meters for self-monitoring. Owing to non-homogenous contamination, decommissioning objects should be given full-scanning measurement, without any leak of hot point. samples taken for cleanup should be of representative, and number of sample should be estimated by stratified sampling method. Prior to packaging of wastes into drum, radioactivity should be measured on drum by drum basis to provide radiological data.

4. Termination survey

4.1 Aim of termination survey

Termination survey is conducted made after completion of dismantlement, cleanup, disposal. It is conducted by licensee and contractor for decontaminated equipment, parts, field for the purpose of identifying whether termination survey is qualified. Verification inspection survey can be made only after termination survey is confirmed to be qualified and decommissioning monitoring report and termination survey report are submitted.

4.2 Contents and requirements of termination survey

Termination survey is firstly to measure the terminal state of decommissioning work, aiming at providing statistic data for termination survey programme. Termination survey objects are (1) decontaminated and cleaned up facility, equipment, parts, plant building, etc., for which surface contamination and radiation field are to be measured; (2) cleaned up soils and fields, for which sampling and analysis of soils, water, sediments are necessary in addition to surface contamination an radiation field measurements. Termination survey should give reliable data because they have direct relevance with standard and control level. Meanwhile, It should be noted that background, or close-to-background, data given by termination survey is also important to same extent, that must be completely recorded.

5. Verification inspection radiation survey

Verification inspection radiation survey is a important part of decommissioning quality inspection, independent on licensee and radiation monitoring personnel participating in decommissioning and not being affected by administrative management. Verification inspection is only based on the national radiation protection regulations and specified decontamination and cleanup criteria.

Verification inspection group staff should be designated or commissioned by regulatory authorities. Verification inspection organizations should be established with consultation with the licensee and local authorities, and the inspection staff should be from the various relevant departments under the approval of the higher competent authorities. If there is no need to sponsor national-level verification inspection, then verification inspection is only technical acceptance of decommissioning work. If national-level verification is necessary, then, such verification inspection is awaiting for the verification inspection by the national authorities. The termination verification inspection radiation survey report is required to be submitted for review by national authorities.

The differing parts between verification inspection radiation survey and termination survey are that the data provided by licensee shall be audited at the stage of verification inspection radiation survey, such as decommissioning procedures, methods, quality assurance documents, measures, remedial actions, records and reports of decommissioning monitoring and termination survey, data processing and evaluation.

6. Conclusion

Radiation measurements in decommissioning of nuclear facilities may involve many issues and can be reflected in decommissioning work at different stages. The quality of decommissioning work can be ensured only each of them is carefully tackled with. High attention should be paid to quality of radiation measurements in decommissioning work at different stages. Integrated quality assurance system should be established, checked and perfected on step by step basis. On the other hand, special radiation measuring apparatus or devices should be developed according to the types of nuclear facilities in our country to meet the needs of nuclear facility decommissioning. This is a task to be undertaken jointly by research units and nuclear facilities.

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BASIC PRINCIPLES OF FEDERAL POLICY FOR REHABILITATION OF TERRITORIES AND POPULATION EXPOSED TO RADIATION (ILLUSTRATED BY THE SITUATION IN THE URALS, RUSSIA)

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The Urals region is known to be one of the most unsafe territories of the Russian Federation both in terms of radiational and general ecological conditions. An emergency radiational situation arose mainly due to the activities of the Mayak Industrial Association (MIA) subordinated to the MinAtom of Russia. The facility's operation has become a source of radioactive contamination of the territory, radiation effects on the population. Besides, it may potentially lead to major radiation accidents.

Radiation exposures of large populations in the Urals resulted from uncontrolled gas-aerosol routine releases which were going on from 1948, discharges of radioactive wastes into the river Techa in 1949-1956, an explosion of a storage tank with high-level wastes in 1957 which led to the formation of the East-Urals Radiation Trace (EURT), downwind transfer of activity from the shores of the drying-up lake Karachay (a depot of medium-level wastes) in 1967, on-going discharges into the bypassing canals from which the activity enters the Techa due to seepage through dams constructed on the river. As a result of only the first three incidents large territories were contaminated in three oblasts (provinces): Chelyabinsk, Sverdlovsk, Kurgan. About 23,000 sq km were contaminated with long-lived radionuclides and 437,000 people were exposed to radiation. Within first years of exposure the maximum absorbed effective doses received by the population due to different radiation accidents were as follows: 170 cSv for the residents of the upper reaches of the Techa, 60 cSv for the EURT residents. In a proportion of exposed population and Mayak workers cases of chronic radiation sickness (CRS) were observed. Over 2000 cases of CRS were diagnosed in Mayak personnel and over 900 cases in off-site population.

The radiation incidents in the Urals inflicted considerable damages on the population's health and the socio-economic status of the entire region. On the whole, economic losses were estimated as 9.5 billion (calculated on the basis of the prices operant in 1990) for Chelyabinsk Oblast alone. The destruction of the social-production infrastructure on these territories, exclusion of the contaminated lands from agricultural use resulted in deformed demographic, migrational and other social processes. The majority of the population of the region developed a persistent radiophobia and anxiety which resulted from living in the locality characterized by increased radiation risk.

In spite of the steps taken before 1992 to minimize the impacts of the radiation exposure on the population and the damage inflicted by the territory contamination the results of such steps proved to be inadequate. The health measures did not encompass the entire exposed population, measures aimed at radiation protection, economic and ecological rehabilitation proved to be ineffective too.

In 1992 the Federal Program for radiation rehabilitation of the population and environment restoration in the Urals region was adopted. It specified the economic, engineering and social tasks which had to be achieved in the period from 1992 through 1995. The dose from accidental exposure was assumed to be the main criterion for decision-making about the necessity of carrying out protection measures on the contaminated territories, the character and scope of such measures, as well as compensation of damages suffered by the population. The effects of radiation exposure on human health were considered in combination with other confounding factors, such as chemical pollution, endemic and biogenic factors characteristic of different rayons.

The principal objectives of the Program were as follows:

- localizing the sources of radiation contamination and preventing further spread of radiation contamination in the Urals region and adjacent territories which may result from further waste accumulation and re-distribution of radionuclides currently deposited in the environment;
- nature restoration in the areas designated as sanitary-restricted zones;

- further development of monitoring the radiation conditions on the territory contaminated with radionuclides;

- improving the standards of medical assistance to the exposed people, especially to the critical population groups, safeguarding the population's health, minimizing the exposure to non-radiation polluting factors;

- development of social and industrial infrastructure, diminution of the socio-psychological tension.

The achievement of each of the tasks listed was based on the implementation of a complex of corresponding measures.

The period of the implementation of the Program coincided with a very complicated phase of the country's economic development. Because of inadequate funding many important tasks of the Program related to social, medical and radiation rehabilitation of the population could not be realized in full measure.

The social-economic and radiation conditions determined the necessity of the development of a follow-on comprehensive Federal Program for social and radiational rehabilitation of the Urals population exposed due to MIA operation for the period 1996-2000.

BUILDING STRATEGIES FOR RESTORATION OF CONTAMINATED AREAS AFTER A NUCLEAR ACCIDENT : IMPLEMENTATION OF RELATED STUDIES AND FIRST ORIENTATIONS

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INTRODUCTION

The widespread contamination resulting from the Chernobyl accident has lead IPSN to implement a program, aiming at building strategies for environmental restoration after an accident potentially occurring in a reactor or a fuel cycle facility. Establishing optimized strategies requires to consider the incidence of all remediation operations, from the collection of the contaminated material to the final storage of the wastes issued from decontamination. Most of the decontamination strategies actually consist in shifting the polluting substances from the environment to a controlled and safer system, i.e. a storage or a repository, which result in the overall decrease of hazards undergone by the various groups of the population exposed. The usefulness of the treatment of contaminated material against the direct storage of the latter will therefore strongly depend on the ability of applied technologies to transform the material in an innocuous product or reduce significantly the volumes of wastes to store, so that doses and costs are globally spared. The incidence of storage will therefore be of great importance in determining optimized strategies. Another important aspect of remediation relates to the feasibility of the industrial implementation of technologies. Some considerations on the above issues are given in the present paper. As a starting point, only rural environments contaminated after a major Nuclear Power Plant accident were considered.

HYPOTHESES ON ACCIDENTAL SITUATION

Making prognosis on the radiological situations resulting from a potential accident is a difficult task considering the numerous parameters involved, such as the nature of radioactive releases, the meteorological conditions and the environmental components that may be touched. A good example of the variability of contamination distribution was given in the Chernobyl area where rainfalls during the period of the accident occasioned a deposition of radionuclides in spots and where the levels of activity were found to vary strongly with the distance according to the physico-chemical nature of the nuclides released, i.e. released in gaseous forms or included in fuel particles (1). The hypotheses presented below must therefore be considered only as rough estimations of what could be an « homogenous » radiological situation in rural environments after a severe accident in a French Nuclear Plant.

First estimations were made through a simulation exercise considering reasonable assumptions on a major accident that could occur on a French 1300 Mwe PWR plant (a S3/10 scenario was chosen for this exercise). In such case, deposition of released activity would lead to levels of the range of 10^4 Bq.m^{-2} for ^{90}Sr, 10^5 Bq.m^{-2} for ^{134}Cs and ^{137}Cs and 10 Bq.m^{-2} for ^{239}Pu at 2 km from the plant. If the site remediation objective was to bring the levels of public exposure down to recommended limits of 1mSv/year, this would lead to remove soils within a zone of 2 km radius around the plant during the first year after the accident, according to effective dose calculations carried out by mean of the ABRICOT code (developed for the modelling of nuclide transfers into the biosphere (2)). Considering a penetration depth in soils of about 5 cm for the major radionuclides, as could be observed in most sites of the Chernobyl area (3), a volume of about 7 10^5 m^3 would have to be removed. This is considerably less than for the Chernobyl case for which the fulfilment of a less stringent remediation criteria (2,2 mSv/year after 4 years) would lead to remove about 10^9 m^3 of soil (4). As a base for studies on the feasibility of site remediation, a contaminated area of 10 km radius around the plant was considered to be conservative (scenario S3 for which public exposures are estimated in most cases to be negligible outside the 10 km zone). Such scenario implies the treatment of about 10^7 m^3 of soil and 3 10^3 to 3 10^5 t of dry vegetation depending on seasonal conditions. If the treatment was to be completed within 2 years, which we propose to be a working assumption , the removal and the treatment of 2 ha/h (1500 t/h) of soil as well as the treatment of 1 to 100 t/h of fresh plants would be necessary to ensure 24 hours a day. Such figures illustrate that besides the decontamination yields necessary to obtain, the industrial constraints of remediation are of key importance in determining strategies for site restoration.

FEASIBILITY OF INDUSTRIAL TREATMENT

A review of existing standard techniques commonly used in clean up industry was made, in addition to already published information (5). The techniques of heap leaching were at first examined. Preliminary investigations show that heap leaching does probably not allow efficient treatment of large volumes of soils contaminated after an accident of Chernobyl type. Besides the very large number of heaps and volumes of reagent needed for treatment, it is very doubtful that a substantial amount of Cs (which is predominant in

contaminations resulting from NPP accidents) may be removed within the time recommended for treatment. Simulations have been performed in order to evaluate the efficiency of leaching for ^{137}Cs. The results are given in figure 1.

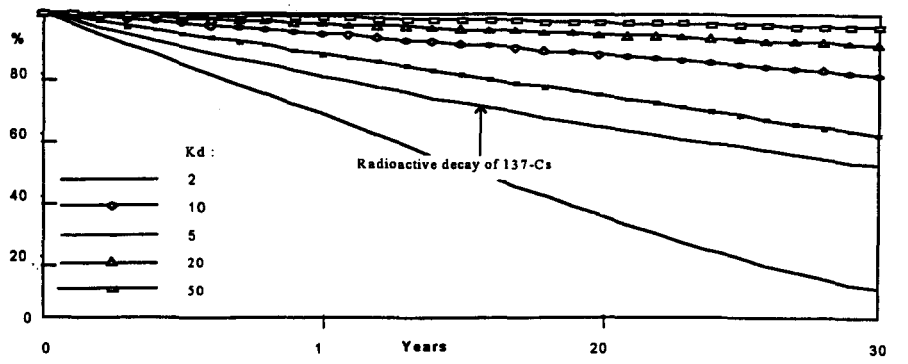

Figure 1. Simulation of the fraction of ^{137}Cs removed (in %) by leaching at a rate of percolation of 0.1 $l.h^{-1}.m^{-2}$ save for radioactive decay

It appears that for a percolation rate of 0.1 $l.h^{-1}.m^{-2}$, which seems a maximum for heaps of about 10 m height, no significant fraction can be removed within two years even for a distribution coefficient (Kd) of 2 ml/g. For smaller heaps of 3m height where larger percolation rates may be obtained (1 to 5 $l.h^{-1}.m^{-2}$), better results can be achieved but still, a very low Kd value is needed. The laboratory tests made on samples from the Chernobyl area show a very strong sorption of Cs on soils which does not allow significant removal by leaching with standard acid reagents. Even if a very efficient reagent was available, the industrial constraints would still remain since enormous amounts of reagent would be needed for efficient leaching (up to 16500 m3/h). It appears therefore that leaching techniques cannot be applied to the treatment of contaminated soils in large volumes. It may nevertheless be remembered for smaller amounts of material contaminated with soluble species.

Other standard techniques used in clean up industry follow principles of separation and sorting of material according to their physico-chemical nature so that adapted treatments can be applied to each component of the contaminated media. In rural environments, one would therefore recommend the separate collection and treatment of vegetal cover and soils.

The collection of vegetal cover at the rates indicated in the previous section appears feasible. Several types of engines adapted to the collection of many kinds of cultures and fields at rates ranging from 20 to 40 ha/day are already available at moderate costs. The economical incidence of radioprotection depends on the choice of designing special engines or « nuclearize » existing ones. This choice and the level of protection required for workers are highly related to the time passed before collection since it has been shown that exposures from atmospheric pathways decrease by several orders of magnitude after a few months. The protection required after such period would therefore concern only external exposure from soils. The methods of treatment of the collected vegetation are based on the potential of the latter for volume reduction. A reduction of a factor 10 can easily be expected when using standard techniques of drying and incineration of collected fresh material. Operations can be conducted in units of moderate sizes, easy to set up in a short time scale, or even in mobile units able to treat 60 to 300 m^3/h of material. The techniques of treatment of secondary products, such as steam and smoke are well proven and can probably easily match safety requirements through the use of high efficiency filters. Drying and incineration techniques are however demanding in energy. An appropriate supply of fuel or electricity on site must therefore be anticipated. The investments and operational costs (without manpower) were roughly estimated to be of the range of 600 FF (French francs) per ton of treated material. Above operations would lead to the production of a maximum of 30000 m^3 of ashes, which is close to the annual volumes of wastes received by the « Aube » repository (low level waste repository operating in France at present). According to predicted levels of activity deposition presented in previous section, the storage of such material (as well as the direct storage of contaminated soils) should be compatible with the overall amount of activity allowed to be received in LLW repositories (« radiological capacity »). These wastes may however have to match other requirements (such as massic activity limits, immobilization and chemical passivity of product) in order to be accepted for disposal in such facilities.

As for vegetal products, the collection of contaminated soils is feasible at moderate costs and submitted to the constraints that were mentioned above. The principles of treatment of collected soils is based on the separation of mineral and organic particles in a first stage, followed by the removal of polluting substances driven in solution by use of a reagent. The resulting effluents are then decontaminated by standard techniques (concentration by evaporation, co-precipitation, elution in ion-exchange resins, etc.). The separation techniques

are mostly based on gravitational sorting. « Dry » and « wet » processes of sorting can be applied. The dry techniques can consist of a sedimentation of particles in a column blown by an upward air stream, or by sieving on rotating meshes. Nevertheless these techniques have been seldom used for industrial purposes and their ability to treat large volumes of material still needs to be assessed. They are furthermore probably extremely demanding in power supply. The wet techniques are much more frequently used in clean up industry and consist in the decantation of particles in a water stream or the flotation of the small fraction driven to surface by injection of compressed air in water. These operations are often simultaneously completed by the addition of reagents for removal of polluting species. As mentioned before, processes of decontamination using the chemical properties of the polluting species are not promising for contaminated soils of Chernobyl type, since the major radionuclides they contain have shown a very strong affinity to the substrate. On the other hand, it has been shown that the major part of the deposited radioactivity was always contained by the small grain size fraction of soil. The separation processes based on size fractioning could therefore act as decontamination techniques for soils contaminated by a PWR accident. However, the efficiency of former techniques for decontamination purposes still needs to be assessed and must allow the separation of sufficiently fine fractions to provide significant reduction of the volumes of material to be stored. Furthermore, the wet separation techniques need heavy infrastructure and are extremely demanding in water supply (up to 8 m^3/s) so that their implementation requires delay and appropriate siting. An optimistic estimation of investments and operational costs for the treatment of the overall 10 km zone (based on the assumption that decantation and flotation can treat a load of 100 g/l of suspended material which is probably a maximum) would be of the order of 3 10^8 FF, but the manpower needed for building the facilities and the transport of material to the treatment plant may considerably increase this cost.

CONCLUSIONS

The studies carried out on the ability of standard industrial techniques to treat very large volumes of contaminated material from rural environments have shown that it is certainly possible to treat a contaminated vegetal cover within a short time without undergoing major industrial and economical constraints. The drying and incineration of collected plants may allow a reduction of the volumes of wastes to store by at least a factor of 10. Some investigations on the nature of the wastes produced are nevertheless necessary in order to assess if additional conditioning is needed to match safety requirements for disposal. The optimization of the treatment of vegetal cover raise the question of the time necessary to leave before beginning operations. An early collection may potentially allow the removal of most of the deposited activity but present higher risks of exposure for workers. The question of finding a balance is, in other respects, also raised for the implementation of some early counter-measures of agricultural type (ploughing, planting) which enable an immediate reduction of exposures by dilution of deposited activity but involve additional difficulties in achieving total decontamination (increase of the volumes to treat). The treatment of soils in large volumes is much less promising, because of heavy constraints of industrial implementation (large infrastructures and very important needs for water and power supply). Furthermore, all the reviewed techniques have to be calibrated with respect to their ability to achieve significant decontamination for soils contaminated after a NPP accident. The technologies based on size fractionating seem the most interesting to test since they may allow good decontamination and a reduction of the volumes of wastes to store. But their efficiency must be assessed at industrial scale and the constraints of implementation still remain for the treatment of large volumes. From above considerations, we would recommend to address the following issues : treat vegetal covers and soils separately ; look for a reduction of the volumes of soil to treat by methods of radiological sorting ; calibrate and improve technologies based on size fractionating ; examine the ability of wastes produced to meet safety requirements for disposal. Finally, optimization should be made by considering, in particular : the balance between the doses spared for people living on decontaminated sites against the doses received by workers as well as by people exposed to the secondary products of decontamination ; the balance between the benefit of early counter measures against total and long term decontamination ; the balance between the radiological and economical costs of the reduction of the volumes of wastes to store against direct disposal of all collected materials.

These conclusions concern of course only rural environments contaminated after an accident of Chernobyl type (NPP accident). We believe that a real optimization must consider together all the categories of environments touched (rural, forests, aquatic, urban). Other types of accidents must also be addressed.

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EFFECTIVE DOSES FROM DIAGNOSTIC MEDICAL EXPOSURE IN ROMANIA

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INTRODUCTION

It is well known that medical applications represent the most important man-made source of radiation exposure for population. Certainly, the diagnostic radiology is by far the largest contributor to the annual collective dose arising from all medical exposures. A relatively small contribution comes from diagnostic nuclear medicine procedures that involved a much smaller fraction of the population than X-ray examinations.

The purpose of this paper is to reassess in terms of effective dose(1) the magnitude of patient exposure from both diagnostic X-ray examinations and diagnostic nuclear medicine procedures,performed in Romania.

Knowledge of the real level of patient dose is an essential component of quality assurance programs in diagnostic radiology and in diagnostic nuclear medicine.

METHODS

Effective doses to about 5900 patients undergoing 20 most important type of X-ray examinations were derive from entrance the surface dose or dose-area product measurements carried out in 42 X-ray departaments,selected by their averall annual workload throughout the country (2), and the appropriate conversion factors calculated for a mathematical phantom by Monte Carlo techniques (3).

Effective doses to patients receiving diagnostic nuclear medicine examinations were calculated from the mean administered activity of the most frequently used radionuclides (4) and the dose factors (for adult patient) published by ICRP (5).

The collective doses, necessary to estimate the average annual effective doses per patient and per capita from all diagnostic medical exposures, were calculated using the frequency data provided by our 1990 national study, included in UNSCEAR 1993 Report (6).

RESULTS AND DISCUSSIONS

Table 1 summarises our results regarding diagnostic X-ray examinations performed in Romania : the entrance surface doses as patient-wieghted averages and the corresponding individual and annual collective effective doses.
Barium enema was associated with the highest effective dose (8.12 mSv) and the annual average effective dose per patient was estimated at 1.55 m Sv. The annual collective effective dose resulting from all X-ray diagnostic examinations was evaluated at 14300 man x Sv. Unfortunately, fluoroscopic examinations have a contribution which accounts for more than 60% of this collective dose. The high values of the effective doses due to traditional fluoroscopy might be reduced by improvements in the techniques and equipment, as well as by lowering the fluoroscopic time.

The annual effective dose per capita was of 620 μSv representing about 24% from the natural radiation background.

The average administered activity and the resulting effective doses both

MEDICAL EXPOSURE IN ROMANIA

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INTRODUCTION

Medical exposure and in particular diagnostic radiology represents in Romania the largest manmade source of public exposure (about 14,200 man Sv/y).

METHODOLOGY

The annual frequencies of 32 different types of X-ray examination by age and sex distribution were estimated from radiological registers of 45 medical units (25 hospitals and 20 ambulatory care units) from 20 districts. As representative for country they were selected on the reported annual workload of their X-ray departments.

The sample of X-ray exams consisted of all radiological examinations carried out in a specified week (2^{nd}) of the middle month of each quarter in 1990 in every medical center taking part in this survey.

The details recorded included age and sex of patients,type of examinations as well as technical factors such as the number of films or the fluoroscopic screening time.

Sample's data (213,675 patients representing 2.86 % from the annual total) were extended to the annual number of X-ray exposures reported to the Ministery of Health by local district Health Statistic Departments [1].

The grand annual total for all types of medical and dental X-ray examination was 7,471,235. This coresponded to 495 exams per thousand head of population [2].

The size of population concret by the medical services was 15,084,146 people. The population of Romania in 1990 was 23,2 million [3]. Local population details such as age and sex distribution (1990) were taken from local Offices of Population Censuses.

In order to assess the tendency of medical exposure in Romania, present results were compared with those obtained in the earlier surveys (1970, 1980) [4].

RESULTS

The annual numbers of diagnostic X-ray examinations decreased between 1970 and 1990, as shown in Table 1,from 1010 exams per thousand inhabitants in 1970, to 495 exams in 1990.

Romania, like other few countries of health-care level I, showed downward trend [2]. The calculated annualy decrease rate for the last ten years was 2.2 per cent. This tendency was manifestly in all age groups as the following data show (Table 2). Also, these data indicate the 16-30 years old as the most frequently examined.

Although particular examinations such as fluoroscopy or mass miniature chest radiography were dropping between 1980-1990, as showed previously, they still have a frequent utilisation. For example, comparing the frequency of fluoroscopic examinations it

was evident that 63 - 74 per cent of all 1990's fluoroscopic examinations were performed for chest (lung, heart) (Table 3).

Table 1. The annual average numbers of radiological examinations per thousand population in Romania (1970-1990)

Type of examination	1970		1980		1990	
	number	%	number	%	number	%
Fluoroscopy	320	31.68	264	42.17	178	35.95
Radiography	240	23.77	150	23.96	182	36.77
Photofluorography	450	44.55	212	33.87	135	27.28
TOTAL	1010	100	626	100	495	100

Table 2. The average ratio of total number of X-ray exams by age group (1990/1980)

Age group(years)	0 - 15	16 - 30	over 30
Ratio 1990/1980	0.48	0.52	0.46

Table 3. The frequency of fluoroscopic examinations between 1980 and 1990

Examination	Age group (years)							
	0 - 15		16 - 30		31 - 45		over 45	
	I*	II**	I	II	I	II	I	II
Chest	87.4	73.3	75.7	70.7	69.4	64.4	68.2	63.4
G.I. tract	12.5	26.6	24.2	29.2	30.5	35.6	31.8	36.5
Stomach,duoden	9.1	15.7	19.1	25.0	24.2	29.1	24.6	28.9
Small+large intestine	3.4	10.9	5.1	4.2	6.3	6.5	7.3	7.6

*) = 1980 **) = 1990

The values of table 3 point out that the children under 15 , are the population group where this X-ray procedure has been most frequently carried out.
In the last survey (1990) also digestive examinations were more common in older patients, the children had an unexpected high frequency of bowel examinations (10.9 per cent).
Detailed quantitative data on the frequencies of different types of X-ray examinations conducted in 1990 are given in Table 4.
As table data show, the thorax is the most irradiated anatomical region, either by fluoroscopic or by mass miniature chest radiography; their cumulated annual frequencies ranged between 37.9 (for 0-15 years old) and 65.9 (for 16-30 years old) from all X-ray examination carried out in a specified age group [5].Digestive barium examinations increase with age,males having a small higher frequency than females (4.71 v 4.45).

Table 4. The frequency of the radiological examination by age and sex (1990)

Type of examination	Age group (years)				Sex	
	0-15	16-30	31-45	over45	M	F
FLUOROSCOPY (total)	35.92	23.16	33.94	45.14		
Lung	26.52	16.18	21.88	28.70	9.54	8.30
G.I. tract :						
- Ba swallow	1.03	1.41	2.27	2.75	0.84	0.78
- Ba meal	4.61	4.57	7.59	10.26	2.90	2.67
- Ba enema	3.76	1.00	2.20	3.43	0.97	1.00
RADIOGRAPHY (total)	64.08	76.84	66.06	54.86		
Lung - PA	12.53	2.63	3.68	4.24	2.60	1.99
-tomography	0.35	0.53	1.10	1.27	0.59	0.28
-photofluorography	11.60	49.72	26.53	13.27	19.43	15.89
Spine(+lombo-sacral joint)	1.51	2.48	5.45	6.32	2.42	2.50
Skull,facial bones (+dental)	9.82	9.38	10.54	8.54	4.92	4.85
Arm,hand,leg,foot	18.98	6.79	9.49	10.07	5.44	3.83
Pelvis	3.51	0.49	0.82	1.11	0.56	0.58
Abdomen: - plane	1.53	0.71	1.44	1.65	0.68	0.60
- colecistography	0.05	0.44	0.99	1.27	0.28	0.50
- urography	0.65	0.56	1.22	1.58	0.76	0.83
-hysterosalpingo-graphy	-	0.37	0.31	0.09	-	0.21
Angiography	0.92	0.60	1.33	1.75	0.20	0.15
Mammography	0.01	0.30	0.59	0.35	-	0.34
Others	2.62	1.84	2.57	3.35	1.21	1.32
TOTAL	100	100	100	100	53.38	46.62

Some specific X-ray examinations as computed tomography have had a very low annual frequency and were reported as "others exams". Mammography was used only for diagnostic, not in screening programmes.
It is possible that spectrum of medical exposure in Romania will be changed for the next years due to extensive use of ultrasounds, after 1990.
So, it will be interesting to compare this last "clasic-radiological picture" with that of the end of century.

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POPULATION EXPOSURE PRODUCED BY UNNECESSARY EXAMINATIONS

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INTRODUCTION

The unnecessary radiological examinations and their implications referring to medical exposure and benefit against detriment problems represent an important subject of a great part of the literature. The evaluation of these is very important in the ex-communist countries, where such publications are almost completely missing. Our study follows to establish the number of unnecessary examinations in three roentgendiagnostic departments and their contribution to medical exposure.

MATERIAL AND METHOD

The study has been done during three months in three different radiological services: Unit A - a hospital of 315 beds, Unit B - a hospital of 147 beds and Unit C - an outpatient's department with 30 000 persons belonging to it. All the three X-ray installation has been in average 20 years old, without image amplifier.

The number of unnecessary examinations varied among 6 - 25 %.

TABLE 1

Unit	Nr. fluoroscopic examinations	Nr. radio-graphs	Nr. all examina-tions	Nr. unnecessary fl. ex.	Nr. unnecessary rad.	Nr. unnecessary at all
A	760	1 125	1 885	28 (3,6 %)	85 (7,55 %)	113 (5,99 %)
B	1 035	791	1 826	43 (4,1 %)	104 (13,1 %)	147 (8,05 %)
C	842	758	1 600	190 (22,5 %)	210 (27,7 %)	400 (25 %)

The causes of the unnecessary examinations were:

1. Factors exterior the X-ray department:

- the general practitioner, who recommended the X-ray examinations (insufficiency or lack of clinical examinations; the X-ray examination is not relevant for the prognosis; unknowing or carelessness of some examinations algorithm; unjustified controls; lack of utilisation the information's of the X-ray examination by the clinician)
- subjective recommendation elements (the patient's pressure; some financial interest of the physician; medico-legal or insurance purposes)
- patients insufficiently prepared for the examination
- the circuit of the examination's results and the films

2. Factors belonging to the X-ray department:
- professional training (radiologist, radiographer)
- the performance of the X-ray installations
- the quality of the materials used (films, developer, cassettes, follies)
- the way of the editing of the results

The number of the repeated radiographs is variable depending on pretension the radiologist about the film's quality. The most frequent repeated radiographs in the studied units were: chest, lumbar spine, cervical spine, skull, extremities, intravenous urography.

The mostly repeated fluoroscopic procedures were: chest, barium meal and barium enema. The reasons: unnecessary recommendation and improper p[reparation of the patients.

To appreciate the patient's exposure during the most frequently repeated procedures we could not apply any direct measurements: entrance surface dose per radiograph and dose-area product per examination because of lack of correspondent instruments.

We measured the exposure dose in the air, using a spherical ionisation chamber and keeping the same conditions of the real X-ray procedures. We investigated in this way 4 radiographs and 2 fluoroscopic procedures.

TABLE 2

RADIOGRAPHS	**REFERENCE DOSE/FILM mGy (average values)**	**COMMENTS**
Lumbar spine AP LL	24,59 57,63	Very different values depending from the X-ray installation
Cervical spine AP LL	16,3 17,3	Similarly.
Pelvis AP	23,37	Similarly.
Urography	97,49	Differences from number of films required
FLUOROSCOPY	**REFERENCE DOSE/EXAMINATION mGy (average values)**	**COMMENTS**
Chest (average duration of examination: 1,5 min.)	24,14	Different values depending from the X-ray installation and radiologist
Barium meal (average duration of examination: 6 min.)	96,54	Similarly.

CONCLUSIONS

The study of the reasons of the unnecessary X-ray examinations have a great importance in such countries, where the use of the old X-ray installations, the frequent use of fluoroscopic procedures, the lack of systematically organised quality assurance in X-ray departments, the lack clinical leaflets and diagnostic algorithms contribute to the growth of the medical exposure of the population and it is only partial controlled. As the replacing of the X-ray installations on national level is a slowly and very expensive process, it seems to be for great importance to put aside or improve the other reasons that lead to unnecessary examinations and irradiation, as follows:

1. Introducing a personal card for the X-ray examinations evidence
2. The responsible involvement of the clinician or general practitioner in recommendation of the radiological examination, with good knowledge of benefit against detriment of this
3. Transforming the radiologist from a simple executive into a decision taking factor; introduction of examination algorithms
4. Informing the patient about the implications of the X-ray examinations
5. Establishing of any criteria in order to editing of the results
6. Improving the education prócess and the training of the radiologists and radiographers
7. Implementing an organised form of the quality assurance in radiological departments.

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RADIATION EXPOSURE OF THE GERMAN POPULATION FROM X-RAY DIAGNOSTIC PROCEDURES

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INTRODUCTION

The following provides an overview of the frequency of X-ray diagnostic procedures in Germany, the effective dose per examination type, the collective and per caput effective doses, trends in frequency and effective dose and the risk evaluation, based on age-dependent risk coefficients.

FREQUENCY OF X-RAY DIAGNOSTIC PROCEDURES

During the years 1990 to 1992, an average of nearly 100 million X-ray examinations per year were carried out in West Germany with 65 million inhabitants, resulting in an average of 1,500 examinations per 1,000 inhabitants per year. The most frequent X-ray examinations were those of the extremities (302/1000), followed by chest radiography (275/1000) and dental examinations (270/1000) (1). All numbers represent complete examinations, i.e., partially consisting of several single radiographs.

More than half of the X-ray examinations are performed by practicing physicians on out-patients. Nearly 25 % of all X-ray examinations are performed by dentists and only about 20 % of all examinations on in-patients in hospitals (2).

At present no relevant data are available from the new federal states (former GDR).

DOSE ASSESSMENT AND EVALUATION

Mean doses per examination type

In order to determine the average radiation exposure of the patient for the various examination types in diagnostic radiology, more than 5000 measurements of the dose-area-product (DAP) have been performed in the years 1992-1994 by our institute and medical phycisists in other parts of Germany (3). Large variations of the DAP up to 2 orders of magnitude have been found for the same examination type due to the great variability of the patients (age, size, weight etc.) and the equipment and technique used (number of exposures, fluoroscopy time, film/screen combination, grid, kV, filter etc.).

From these data mean values of DAP were calculated for more than 30 examination types. Mean values of effective dose E were determined by using examination type-specific conversion factors obtained from Monte-Carlo simulation of the relevant examination with anthropomorphic phantoms (4). Effective doses for computed tomography (CT), mammography, dental examinations and examinations of the extremities were estimated seperately. The mean value of E vary between 0.01 mSv for dental examinations and nearly 30 mSv for CT of the abdomen.

Collective effective dose

The annual collective effective dose from X-ray diagnosis for West Germany (1992) of approximately 114000 manSv was obtained by multiplication of the estimated mean E per examination type with the corresponding annual frequency and summation over all types of examination. This rather high value, equivalent to an annual per-caput value of ca. 1.8 mSv, is mainly caused by the rapid increase of angiography including interventional radiology (ca. 9% of the collective effective dose) and even more of CT, which alone is responsible for about 35% of the collective effective dose.

TRENDS

In spite of the introduction of alternative examination techniques such as sonography, mainly for parenchymal organs, and endoscopy, mainly for esophagus, stomach and large intestine, the number of X-ray examinations has altogether increased,

namely by about 10 % between 1988 and 1992. The frequency of some types of X-ray examinations is very strongly increasing, i.e computed tomography by +80 %, angiography by +54 % and mammographie by +42 %. Other examination types are markedly decreasing, i.e. mainly those of the GI-tract by -32 % (2).

Since dose intensive examinations such as CT and angiography, including in particular interventional radiology, significantly increased and their absolute number is larger than that of GI-tract examinations of decreasing frequency, the collective effective dose increased between 1988 and 1992 by about 14,000 manSv or 14 % from 100,000 manSv in 1992. The annual per caput effective dose increased by only 8.5 % during this period, since the population also increased from about 61.5 Million 1988 to 65 Million 1992.

However, the per caput value is purely computational, since it includes also those fractions of the population where no X-ray diagnosis was performed. The per caput dose may not be used for calculating individual risks. The value is only suitable for comparison of countries with different medical care systems.

QUALITY ASSURANCE AND QUALITY CONTROL

The legal provisions concerning diagnostic radiology prescribe that all radiological equipment must be thoroughly inspected by the manufacturer and an expert prior to starting up and afterwards every 5 years (5).

Another quality assurance measure concerns the qualification of the staff. Only physicians with a special qualification in both the medical field and radiological protection are permitted to use X-rays and are eligible for establishing the indication. All other physicians as well as radiographers and auxiliary medical staff are allowed to take action only upon instruction and under supervision. However, they must also possess medical knowledge and are obliged to attend a course in radiological protection, albeit on a lower level.

Quality control of all X-ray machines is achieved with so called " constancy tests" performing phantom exposures once a month. The performance of the film-processing machines has to be controlled daily.

Important institutional panels for acceptance auditing are committees which request X-ray films of patients subjected to all types of diagnostic procedures from all physicians and hospitals and are offering advice on the improvement of image quality (1).

The Federal Medical Board of Germany has published guidelines for quality assurance describing medical and technical requirements how to conduct standard examinations of conventional radiology and computed tomography (6-7).

By all these measures it is intended to optimize the radiation exposure of the patient in diagnostic radiology.

ASSESSMENT OF RISK AND BENEFIT

It is shown in annex C of ICRP 60 (8) that the lifetime probability of radiation induced fatal cancer varies with sex and predominantly with the age at the time of exposure. For the multiplicative risk projection model, which is prefered at present by the ICRP, the lifetime mortality risk decreases with increasing age at exposure.

If the collective risk from diagnostic radiology needs to be assessed, one must use an adequate average risk coefficient for patients to be applied to the collective dose. Since the age distribution of patients is generally very different from that of the whole population, the ICRP average risk coefficient of 5.2 % per Sv (8, Annex C) is not applicable to patients.

Based on a representative trial in West-German hospitals of 1990 the age distribution of in-patients representing about 20% of all patients was determined for all X-ray diagnostic procedures and classified for 7 types of examinations, making about 86 % of the collective effective dose of in-patients. About 45% of all in-patient examinations are performed on patients older than 65 years, further 30% on patients aged between 41 and 64 years. From the age distribution per examination type mean risk modifying factors for in-patients per examination type were derived. For the 7 examination types the risk modifying factors vary between 0.38 and 0.51. The weighted mean, finally gives a mean risk modifying factor for in-patients of 0.47. In other words: the stochastic radiation risk of in-patients is only about half of that of the population. The mean risk modifying factor for out-patients (ca. 75 % of

all X-ray examinations) surely must be higher due to their younger mean age. Therfore a risk modifying factor of 0.6-0.7 for all patients of diagnostic radiology seems to be quite reasonable.

These factors take into account only the different age structure between patients and population, but no other possible differences in risk related to the different health status of patients. The mean life-expectancy of patients with severe diseases is shorter than for the average of the population in the respective age class and also shorter than the latency period. Since a considerable part of the collective dose is used in the diagnostics of such patients the overall radiation risk coefficient is reduced.

However, assessments of the radiation risk in diagnostic radiology are of no value if the risk is considered separately from disease and therapy related risks and if the benefit for the patient from the radiological examination is not taken into account. The difficulty is, of course, to quantify the benefit because it is not easily definable. The benefit could, for example, be measured in years of prolonged life expectancy, which again are difficult to assess. According to other conceptions, the benefit is identified by those portions of radiological diagnoses that lead to - positive or negative - therapeutic decisions because they are the only ones of importance for the patient. Therefore the most important question before performing an examination should be: Will I get an information which really influences the therapy of the primary disease?

When ionizing radiation is applied in medicine, the benefit to the patient should always be the main priority. This benefit is an integral part of the risk-benefit-evaluation together with the individual radiation risk for the patient and other individual risks from the examination. For the assessment of the individual radiation risk from a specific examination, the age-dependent cancer mortality risk coefficient of that patient may be used together with the average effective dose of the respective examination type. Additionally must be taken into account that the age-dependency of cancer mortality varies for different tumors and also that a very different individual presdisposition exists for the development and survival of a cancer disease, which, in part, could be of genetic origin. Added to this, for the exposure of young patients, the genetic risk must be considered which can largely be disregarded for older patients.

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A SURVEY OF CHEST MEDICAL X-RAY DOSES

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INTRODUCTION

The medical X-ray exposures due to radiological examinations are responsible for the largest contribution to the population collective dose as result of the normal use of artificial sources of radiation.. The relative impact of the medical exposures to the total dose received by the population from all kinds of radiation sources varies from country to country and in some cases within the same country (1). The dose variations observed for a specific type of examination are in general associated to several factors i.e. the type of film-ecran combination, the choice of the appropriate physical parameters of the x-ray generator (Kvp, filament current, exposure time) and finally to the film processing conditions.

At the present moment the data available in Brazil are scarce and scanty to allow a complete analysis of this question so needed by the health authorities to justify the implementation of a quality assurance and dose reduction programs. In addition, it is desirable to establish a cost effective operation based on simple administrative concepts in order to reduce the number of films retake, then increasing the life expectancy of the equipment and the number of radiological procedures.

The aim of this work was to assess the typical doses of an PA and LAT chest wall X-ray examinations in five different public hospitals (a University Hospital, a Cancer Hospital, a Navy Hospital, an Emergency Hospital and a State General Hospital), as representative of the city of Rio de Janeiro, and compare the results with the international data and recommendations available.

MATERIALS and METHODS

Chest X-ray examination was selected for this study since by far is the most frequent type of procedure requested as part of a normal medical routine in a hospital or as part of the admittance procedure for a new job.

The experimental methodology adopted for this work is very similar to the one developed by the Institute of Physical Sciences in Medicine (2) and accepted by the Common Wealth as well as by several countries in the European Community.

In-vivo measurements were conducted using chips of Lithium Fluoride (TLD-100) thermoluminescente dosimeters, manufactured by Harshaw. A conventional thermal treatment was applied to the TLD's (400°C/1h + 100°C/2h) and then they were exposed several times to a reference Co-60 gamma beam in order to group them in accordance with its relative response. A calibration factor adjusted for its well known energy dependence was assigned for each group. As result of this procedure, the typical overall random uncertainty for a dose of 1 mGy is 4% (1 σ).

For each measurement, the dosimeter was placed on the patient skin at the center of either the PA or the Lateral radiation fields, and immediately after stored in a lead box for subsequent evaluation. A minimum of 10 patients were selected in each hospital (average height of 1.6 m and 65 kg of weight) and further identified by age and sex. No changes or suggestions were made related to the selection of the radiographic parameters during the radiological procedure in each place.

RESULTS and DISCUSSION

The Fig.1 gives an overall view of the average entrance dose measured at the surface of the patients for the PA and LAT projections for all the examination rooms evaluated. By looking at the results individually by machine or institution, one is tempted to believe that the homogeneous distributions are an indicator of consistency of the operational procedures. However, when the individual patient records were analyzed, severe inconsistencies were clearly shown related to the choice of the machine parameters(kVp and current) for a

particular dimensions of the patient (height and corporal mass). This observation might be strongly related to the general level of training of the radiographers or his lack of attention to a specific machine. It is a normal practice in our Country to have the radiographer working on shifts, and as result of that, they tend to work on several different places and different machines, not being able to develop familiarity and a professional feeling about the performance of a machine.

The Table 1 shows the average dose values for this type of examination published by national (2) and international organizations (1,3) together with the values obtained by this study. It is clear that the measurement results obtained at the hospitals # 1, #3 and #5 are in good agreement with the recommended values also shown in the same table.

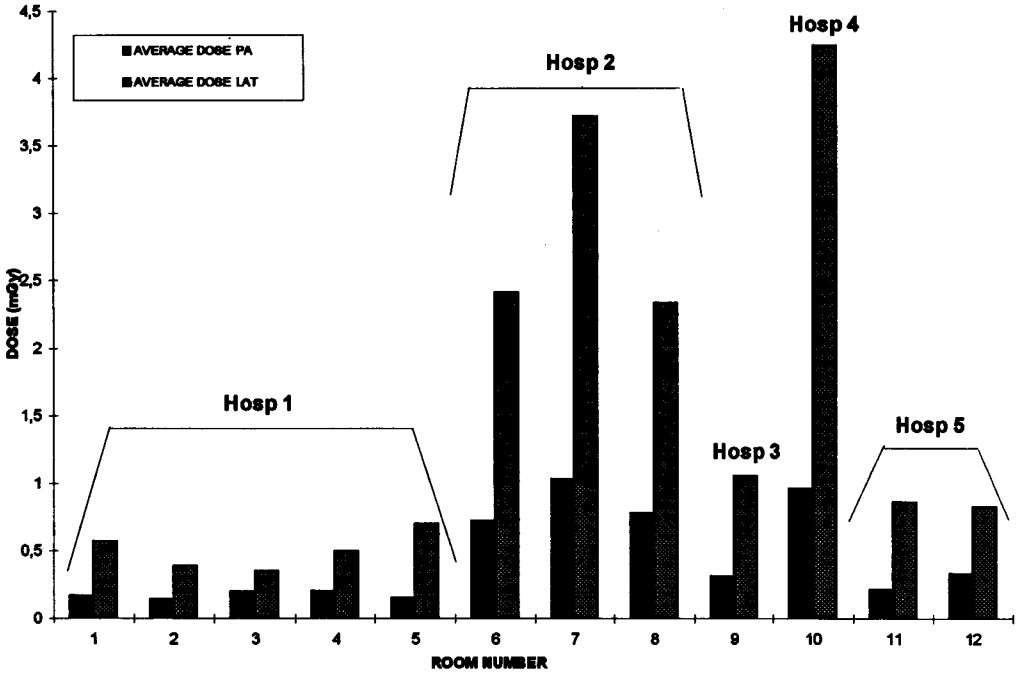

Figure 1: Average entrance dose values for the entire set of data.

However, the same positive considerations cannot be made for the results presented by the hospitals # 2 and # 4. The measured values are two times greater than the reference values presented in Table 1, as result of problems observed in the operational procedures presently adopted i.e. consistently wrong choice of physical parameters for a particular patient size, old ecrans still in use, filament current indicator and timer both running out of the specifications, dark room with light leakage without any quality control and last but no least insufficient training of the radiographers

Table 1: Comparison between the values obtained in the present work with the recommended values published by the UNSCEAR, IPSM-UK and IAEA.

INSTITUTION	AVERAGE DOSE- PA mGy	AVERAGE DOSE- LAT mGy
IPSM-UK	0.18	0.99
UNSCEAR	0.4	1.5
IAEA	0.4	1.5
PRESENT WORK	0.43	1.49

CONCLUSIONS

1 - The use of thermoluminescent dosimeters, TLD-100 to measure the entrance doses of chest X-ray examination has shown to be adequate if the necessary care is taken in its selection and a good and consistent methodology is followed for the storage, irradiation, thermal treatment and the read-out of each chip.

2 -The results tends to show that the average entrance dose for the chest x-ray examinations measured in five typical hospitals in Rio de Janeiro are very similar to the world average value published by UNSCEAR and the IAEA reference value as well. However, the results obtained in two of those hospitals strongly indicates the immediate need of a quality assurance program in order to optimize the radiological procedures generating a positive impact in the dose reduction for the population, in the operational costs and set a example for other institutions as well.

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PAPER TITLE Analysis of diagnostic X-ray units in Poland concering the exposure of patients to ionizing radiation

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MAJOR SCIENTIFIC TOPIC NUMBER 6.1. (see page 7)

ABSTRACT (See instructions overleaf)

The diagnostic X-ray units in Poland were subjected to the analysis. The air-kerma on the X-ray table was measured for the X-ray parameters used routinely during lumbo-sacral spine radiography of standard patient. The measurements of air-kerma were performed using TLD made of lithium-fluoride.

The dose received by patients in different X-ray departments can be more then two orders of magnitude from 0,14 mGy up to 72,11 mGy, for the same X-ray examination. The dose can be significantly reduced if the cassets with amplyfying foils made from components of rare earth are used.

EVALUATION OF RADIOLOGICAL DETRIMENT FROM NEGATIVE RADIOLOGICAL EXAMINATIONS

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ABCTRACT

The individual doses of radiation due to diagnostic radiology are usually low, though their contribution to the collective dose is very important given the large numbers of people exposed to these. This paper presents an analysis of the number of negative radiological examinations in a major Cuban Hospital, and their contribution to the collective dose of radiation.
The reports of each individual radiological examination are classified according to the type of examination, and recorded as positive or negative according to the criterion of clinical indications. The absorbed dose by irradiated organ and tissue, effective dose equivalent, collective dose, and radiation risk (expressed as severe hereditary effect and the occurrence of fatal and non fatal cancers) are all evaluated.
The total contribution of negative examinations to the collective dose are found to make up 52.9 % : 11.35 Sv-man in the studied population.

INTRODUCTION

The rapid development reached in medical radiology and increase medical irradiation of patients during the last years, has made possible that the diagnostic radiology and radiotherapy are now considered the biggest contribution to the collective dose, because large numbers of people are exposed; consequently the majority radiation exposure from artificial sources.

This situation has given place to the fact that the expert scientific groups in radiation protection consider the necessity for the protection of the patient and promote the aplication of the concept of rational use of diagnostic radiology and the justification of its practice, if it really produces sufficient health benefit to the exposed individuals (1-2).

In our country during the last years it has been reported significant increment of radiological examinations and using the criterium that a large percent of these examinations, in its clinical indications were not justified, we decide to know the incidence of negative examinations, evaluate its contribution to the collective dose and estimate the health detriment associated to these expositions in a big hospital in Havana.

MATERIAL AND METHOD

This study was carried out in the important Hospital "Manuel Fajardo" in Havana City. All patients with radiological examinations performed, in the emergency services, hospital admission and out-patients services, during four months in 1993, were recorded. We individiualized information for each patient, and examinations were classified by type of X-ray examination and number of exposures.

Based on the justification criterium of clinical indication of the radiological examination, Its were classified into positive or negative examinations. The positive examinations were considered those in which the

clinical indication was confirmed with the diagnostic information obtained or by diagnoses where a new disease was unexpectedly found. The negative examinations were those radiografic studies without clinical diagnostic or no useful radiografic because of use of unsuitable radiografic technique: improper film and defect in the film pocessing card.

An analysis of frecuency distribution for each examination type, according to classification, was carried out. The absorbed dose has been obtained according to the organs and tissues irradiated in correspondence with the radiografic techniques taking into account the technical factors employed per examimation (3).

The equivalent dose, the effective dose and the collective effective dose for each study group in the exposed population. were estimated. The contribution to the collective dose of each examinations, was evaluated. The comparative analysis of the contribution dose of the negative and positive examinations. was presented. The radiological risk was expressed through the probability of ocurrence of radiation induced fatal cancers, non fatal cancers and serious heriditary effects as a result of medical X-ray exposure (4).

RESULT AND DISCUSSION

In this paper 5486 patients with X-ray examinations, were recorded. From them 2249 (41%) resulted negative examinations and 3287 (59%) positive.
The distribution and percentage of each type of examinations according to classification is presented in Table I.

TABLE I
DISTRIBUTION OF RADIOLOGICAL EXAMINATIONS

Examination Type	Positives No	%	Negatives No	%	Totals
Barium enema	101	38.9	158	61.0	259
Urography	133	46.6	152	53.3	285
Single Urinary t.	150	43.8	192	56.4	342
Digestive tract	399	57.4	295	42.5	694
Spine	435	59.5	295	40.4	730
Chest	2019	63.5	1157	36.4	3176
Totals	3237	59.00	2249	41.00	5486

Chest examinations represented a high average in relation to the remainder of the X-ray examinations reported. The high average for negative examinations in three examination types: barium enema, single urinary tract and urography, was obserbed, though these examinations require the specific clinical criterium for indication and they are the most complex studies in conventional radiography.

The contribution to the collective dose according to the examination type is shown in table II. The collective effective dose estimated for all examinations was 21,58 Sv/man. The contribution of negative examinations to the total collective dose constituted 52.9%. The contribution to the collective dose due to negative examinations such as barium for enema, simple urinary tract and urography examinations, is significative because they

represent 86,1% of collective dose resulting from all negative radiological examinations, although these examinations represent only 16,1% of all examinations. These results are similar to those reported in the literature (5).

TABLE II
CONTRIBUTION TO THE COLLECTIVE DOSE

EXAMINATION TYPE	COLLECTIVE DOSE (mSv)	
	Positives Exs.	Negatives Exs.
Barium enema	1,72	2,69
Urography	5,14	5,87
S urinary tract	0,91	1,17
Digestive tract	0,39	0,29
Spine	1,29	0,87
Chest	0,76	0,43
Totals	10,26	11,32

The risk estimate for the studied population showed the probability of occurrence of one case for fatal cancer induced by X-ray examinations. Probability of ocurrence of non fatal cancer and hereditary disorders was not estimated. The mathematical expectancy of probability of occurrence of radiation induced cancer in patients with negative examinations was not estimated, because the influence of low level doses reported in organs, sample size and recent values of risks factors recommended in the ICRP 60(4).

CONCLUSIONS

The significant average obtained of negative examinations in this study showed that it is necesary to establish the justification criterium for the clinical decision of radiological study.
Contribution of the collective doses associated to the negative radiological examinations gave significative values to the collective effective dose.
The matematic expectancy due to negative radiological studies discarded the possibility of estimate quantitatively radiological risk, however we is not discard occurrency of health effects in exposed individuals.

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DOSE DISTRIBUTIONS OF PATIENTS FROM CHEST FLUOROSCOPY, UPPER GI-TRACT RADIOGRAPHY AND CINEMATOGRAPHY IN JAPAN

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INTRODUCTION

The per caput dose from medical exposure in Japan is several times higher than in other developed countries. There are no dose limitations for medical exposure. Then, the appropriate applications of radiation diagnosis/treatments (justification of practices) and the quality control of diagnosis/treatments(optimization of protection) are needed to reduce the doses from medical exposure. It is well documented that patient doses from a X-ray diagnosis are distributed in the broad range. Recently, the IAEA introduced guidance levels for some typical X-ray diagnosis and in vivo nuclear medicines. We carried out the investigation of dose distribution of patients from the X-ray examinations of chest, cardiovascular cinematograph and upper GI-tract X-ray examination in order to give the basic information on the quality control of each X-ray diagnosis. These X-ray diagnoses are performed frequently in Japan, and especially chest X-ray examinations are carried out periodically to all population more than 18 years old as legal health check and GI-tract X-ray examinations to the persons more than 35 years old. The cardiovascular cinematography and the upper GI-tract X-ray examination bring higher effective dose for patients. More information is, therefore, needed for the reduction and quality control of medical exposure in Japan.

MATERIALS AND METHODS

We estimated the student doses from the chest fluorography as legal health examination, the adult doses from the upper GI-tract X-ray examination as periodical health check from a part of the occupational health service and the child doses from cinematography of cardiovascular. The doses from the chest fluorograph were estimated by a phantom experiment on the condition of the specific voltage and current of X-ray tube. From these result, each dose was calculated on other condition, which were given in our nationwide survey for all national and prefectural universities and colleges. The doses from GI-tract X-ray examination were estimated from both phantom experiments and the direct measurement of 105 patients with TLDs (Mg_2SiO_4 and $CaSO_4$). The time of fluoroscopy and the numbers of exposure during GI-tract X-ray examination were counted for 417 subjects. The doses from the cardiovascular cinematograph were measured with TLDs (Mg_2SiO_4) worn on the body surface of 9 children.

RESULTS AND DISCUSSION

The dose distributions of lung from chest X-ray examination are shown in **Fig.1**. The doses of lung from chest X-ray examination were broadly distributed from 0.1 to 1.5 mGy in spite of one exposure in all the examination. Some of the doses from chest examination were over the guidance levels of typical X-ray examinations recently proposed by the IAEA. In chest fluorography, many types of X-ray machine, old and new one, were used. The higher doses were brought by old X-ray machines in which lens cameras were frequently used as fluorescence screen. In the optimization of chest fluorography, it is necessary to consider an appropriate X-ray machine. The range of the skin dose from the upper GI-tract X-ray examination was from 8.9 to 210 mGy (**Fig.2**). This is due to the variability of time of fluoroscopy and the number of exposure. According to our survey, the time of fluoroscopy in upper GI-tract examination depended on radiation technicians as shown in **Fig.3**. The time per one examination for technician was broadly distributed from 32 to 331 seconds. In addition, **Fig.4** shows that how to increase of the time per exposure with the number of exposure is different between fluorography and radiography. It was evident that the optimization in GI-tract examination depended on the technical control of a technician. In this case, education for the technician is essential for optimization in upper GI-tract X-ray examination. The skin doses of children treated with cardiovascular cinematography are shown in **Fig.5**. The doses were ranged in 206 to 995 mGy. The difference of doses in cinematography depended on the clinical

condition of each patient. In medical exposure, it is suggested that there are high possibilities of the dose reduction from technical aspects of machines and technicians.

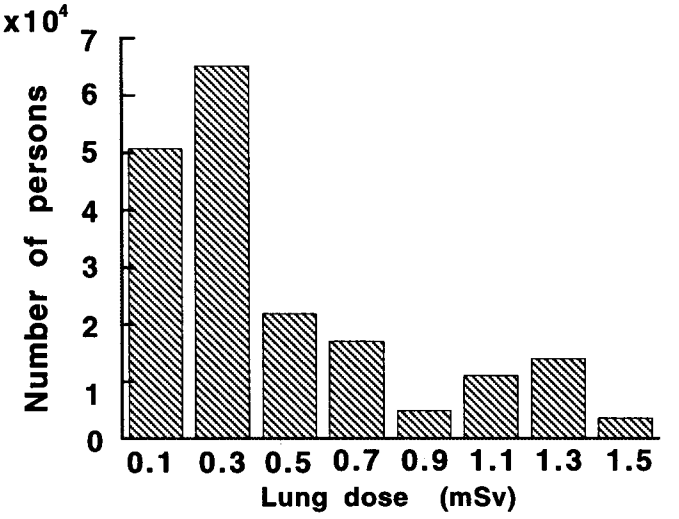

Fig. 1 Dose distribution of lung dose from chest X-ray examination

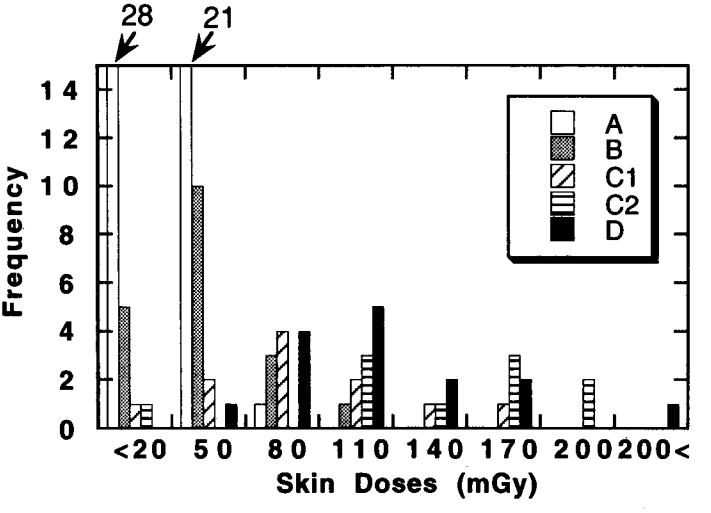

Fig.2 Skin dose distribution of upper GI-tract examination

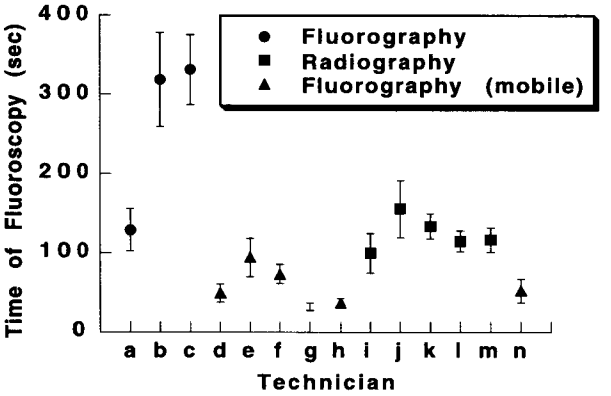

Fig. 3 Distribution of the time of fluoroscopy among technicians

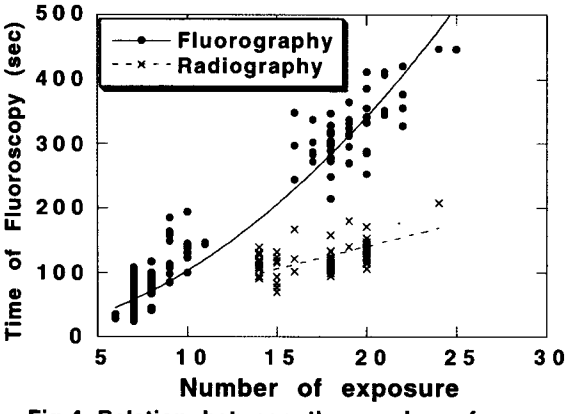

Fig.4 Relation between the number of exposure and the time of fluoroscopy

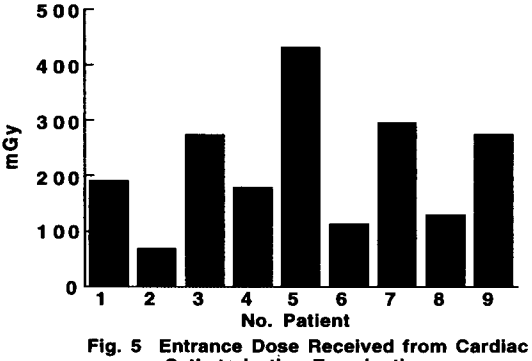

Fig. 5 Entrance Dose Received from Cardiac Catheterization Examination

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PAPER TITLE A National Survey of Canadian Mammography Facilities

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MAJOR SCIENTIFIC TOPIC NUMBER 6.1 (see page 7)

ABSTRACT (See instructions overleaf)

During 1994-1995, a national survey was conducted in Canada to assess equipment imaging performance and radiation doses of Canadian mammography facilities. The survey was a cooperative effort of the federal Radiation Protection Bureau, all ten provinces and both territories. The Bureau provided the survey instrumentation, data assessment and overall coordination, while the provinces and territories carried out the actual site surveys.

A random sample consisting of approximately 200 facilities and stratified by province was initially selected. As the survey progressed, however, several provinces used the opportunity to conduct census surveys of all of their facilities. As a result, the final total number of facilities surveyed was over 300, which represents approximately one-half of all Canadian mammography installations.

The survey followed the Nationwide Evaluation of X-ray Trends (NEXT) protocol, which was developed and used by the US Food and Drug Administration for a similar 1992 survey of 400 US installations. Since one aim of the Canadian survey was a comparison with these earlier US results, care was taken to ensure comparability of the data. Accordingly, survey instrumentation was calibrated and film scoring was done using identical methodologies and, where possible, was carried out by the same individuals.

Participating provinces and territories will use their individual results as a baseline for the comparison of the performance of mammography facilities within their jurisdiction with those of Canada and North America as a whole.

The aggregate national results, including a comparison with the US survey of 1992, will be presented.

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PAPER TITLE Evaluation of radiation load of pediatric patients from diagnostic X-ray examinations in Slovakia

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

A survey has been performed to investigate typical radiation dose levels for children undergoing the most frequently radiodiagnostic examinations. Data analysis of radiographic techniques and measured entrance surface dose using TLDs were performed. Determination of effective doses and collective effective doses from different types of radiodiagnostic examinations were calculated. The result of this study emphasize the necessity for the improvement of pediatric radiologists training and urgent need for quality assurance guidelines.

SLOVENIAN NATIONAL SURVEY OF DOSES TO PATIENTS UNDERGOING X-RAY EXAMINATIONS

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ABSTRACT: In 1994 Slovenian national survey of doses to patients undergoing X-ray examinations begun. Results of the survey of lumbar spine examinations in both projections are presented. Entrance skin doses per radiograph were measured with TL dosemeters. Measurements were peformed on adult patients of both sexes. The median of entrance skin doses is calculated and compared to guidance levels from Basic Safety Standars. Effective dose is also estimated.

INTRODUCTION

The population of Slovenia is estimated to about 2.000.000 inhabitants. There is about 300 X-ray machines in medical use. In 1991 the x-ray diagnostics was covered with 87 radiologists and 318 radiological engineers. The number of x-ray examinations in 1991 was estimated to about 1.200.000 including a little more than 430.000 sceletal examinations (1).

However, till 1994 when national survey of doses to patients begun, little was known about doses to examined patients. Due to relatively high frequency of sceletal examinations, the first phase of the survey was focused to X-ray examinations of lumbar spine.

MATERIALS AND METHODS

According to Slovenian regulations each X-ray machine has to be controled at least once a year by a qualified expert. The collection of patient doses data was carried out during these regular checks. $Li_2B_4O_7$ TL dosimeters (Panasonic UD-807HA2) sticked on the patient skin were used for entrance skin dose (ESD) measurements. Dosimeters were calibrated to measure entrance skin dose with Co-60 source Capintec ionisation chamber PR 06 and Capintec electrometer WK 92.

For X-ray examinations correction factor for energy dependance was used. Aditional data concerning exposure technique (tube potential, total filtration, FFD, image receptor size) and examined patient (sex, age, height, weight, patient thickness in the centre of the beam) was also collected.

Effective dose calculations were carried out using X-DOSE software, developed by John Le Heron from National Radiation Laboratory of New Zeland and is based on normalised conversion factors from ESD to organ doses calculated in NRPB using Monte Carlo techniques (3).

RESULTS

From the total number of our X-ray nachines only on about 10 % X-ray examinations of LS spine is performed and about one half of them were checked during the survey.

Statistical parametres caracterising patient sample are listed in Table as well as the corrensponding measured and calculated values for doses.

Median values of entrance skin dose measurements per radiohraph are well belove the guidance levels (4).

CONCLUSIONS

1.The number of "counts" was small. (See Table!)
2.Estimated mean effective doses for X-ray examination of LS spine for both prejections were for males 1,9 mSv and for females 1,5 mSv.
3.Higher doses were due, (mainly) , to improper film processing.

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Table : Statistical parameters caracterising patient samples

LS - AP	male			female		
	age	*weight (kg)*	*ESD (mGy)*	*age*	*weight (kg)*	*ESD (mGy)*
Mean	53	85,0	8,0	50	67,4	5,9
Standard Error	3	3,1	1,0	5	2,9	1,1
Median	51	87	7,4	46	64	4,6
Standard Deviation	13	11,9	3,9	19	10,7	4,1
Range	42	51	11,0	63	36	12,4
Minimum	31	52	3,3	25	54	1,1
Maximum	73	103	14,3	88	90	13,5
Count	15	15	15	14	14	14
Confidence Level (95%)	7	6,0	2,0	10	5,6	2,2

LS - LAT	male			female		
	age	*weight (kg)*	*ESD (mGy)*	*age*	*weight (kg)*	*ESD (mGy)*
Mean	52	82,7	18,0	52	65,1	15,5
Standard Error	3	2,8	3,1	4	1,9	2,2
Median	50	85	11,3	53	62	10,3
Standard Deviation	14	12,4	13,5	19	9,6	10,9
Range	49	51	42,1	64	38	34,6
Minimum	31	52	5,1	25	52	3,05
Maximum	80	103	47,3	89	90	37,6
Count	19	19	19	25	25	25
Confidence Level (95%)	6	5,6	6,1	7	3,8	4,3

AGE AS A FACTOR IN ASSESSING RISKS TO PATIENTS FROM THE USE OF X-RAYS

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INTRODUCTION

Any use of ionising radiation carries with it a risk of harm, and it is often necessary to estimate this risk for groups of irradiated persons, or for individuals. Such persons include patients in diagnostic radiology. In diagnostic radiology effective dose has become widely used as the preferred index of patient dose. This is primarily for two reasons - effective dose is linked to a useful definition of radiation detriment and, because it converts an actual irradiation into an equivalent uniform whole body dose (in terms of detriment), it is able to cope with the partial body irradiations characteristic of diagnostic radiology. But while effective dose has these attractions, it is based on assumptions that could impose limitations on its use for risk assessment. Effective dose is an index that has been averaged over populations and risk projection models, and its application to individuals or specialised groups of the population may not be appropriate.

In this paper the collective risk from the use of diagnostic radiology in a large New Zealand teaching hospital is assessed using an age-specific approach, and this is compared with the collective risk assessed using the coefficients given in ICRP Publication 60 (1) for the general public.

METHOD AND RESULTS

Age- and sex-specific risks of stochastic radiation effects have been calculated by using the health effects model developed at NRPB for a UK population (2) . In applying the model, ICRP 60 values were used for cancer lethality fractions and for weighting factors associated with hereditary effects. These age- and sex-specific lifetime risks of total radiation detriment were then used to derive age- and sex-specific tissue weighting factors. The age- and sex-specific tissue weighting factors vary considerably since the relative contributions of the tissues to total detriment strongly depend on age at exposure and to a lesser extent sex. For example, the gonads, while dominant early in life, assume little significance later on. Conversely the lungs' relative importance increases dramatically in the middle years, while the red bone marrow's importance increases throughout.

The age- and sex-specific tissue weighting factors were used to derive age- and sex-specific "effective doses" for x-ray examinations that were considered significant in terms of contributing to collective dose. These "effective doses" were derived from the Monte Carlo based normalised organ dose data of the NRPB, UK (3,4). "Effective doses" for most x-ray examinations were smaller than the ICRP values, generally reflecting the reduced significance of the gonads at older ages. Notable exceptions were examinations of the chest region, where "effective doses" exceeding twice the ICRP value were obtained for some adult age bands. Differences between males and females were small compared with the differences due to age at exposure, hence a sex averaged age-specific "effective dose" was assigned to each x-ray examination.

Assessment of lifetime risk from an x-ray procedure was performed by multiplying the age-specific "effective dose" by the age-specific total risk coefficient. When these results were compared with the estimate of risk obtained by using effective dose and the risk coefficient for the general public significant differences were obtained. For example, with skull x-rays, the age-specific risk was about two to three times greater than the ICRP estimate for children, but fell to below the ICRP estimate for older adults. For chest examinations the age-specific risk was above the ICRP value for ages at exposure up to about 60 years. And for the abdomen, the age-specific risk was about double the ICRP risk for children, but less than half for most of adulthood.

X-ray examination statistics were obtained for a large New Zealand hospital for a six month period, giving patient numbers for males and females in 5 year age bands. To illustrate the very different make up of the patient population compared with the New Zealand population, a comparison of proportions of patients in age bands versus the general population is presented in Figure 1 for three x-ray examinations.

Age-specific collective risk to the patient population was determined using age-specific "effective dose", age-specific risk coefficients and the age-specific frequencies of the x-ray examinations. For comparison the ICRP estimate was obtained using effective dose, the risk coefficient for the public and total frequencies for the x-ray examinations. The relative results are presented in Figure 2 for simple radiographic examinations, barium studies, IVUs and CT examinations. The age-specific collective risk assessment for each of these examinations is about one-half that of the ICRP estimate, with the exception of chest and skull examinations (both conventional and CT). In the case of the chest age-specific risk estimates, this is a result of the relative importance of lung doses in the older age groups, for the skull examinations, it reflects the relatively large number of these examinations performed in the younger age groups and the higher risks at these younger ages.

The estimate of total collective risk to the patients from the practice of diagnostic radiology in this hospital, using the age-specific risk approach, was approximately three-quarters of the ICRP estimate. The large proportion of CT examinations being performed in the hospital meant that the contribution from CT chest and CT head examinations had the effect of bringing the age-specific total collective risk estimate closer to the ICRP estimate.

DISCUSSION

The estimate of collective risk using the age-specific approach was less than the simple ICRP estimate. However the age-specific approach necessitates many sets of tissue weighting factors, and the potential confusion of age-specific "effective doses" for all the x-ray examinations. In addition, given the large uncertainties in the nominal risk coefficients for radiation detriment, it must be concluded that the additional "accuracy" of the age-specific approach to collective risk is not justified.

For estimating individual risks or risks to particular age groups it would be desirable, from a practical perspective, to retain the single ICRP set of tissue weighting factors but to obtain a more specific estimate of risk. NRPB have looked at this in the context of simple radiographic projections, and concluded that sufficient accuracy was obtained by adhering to the ICRP approach, but if a particular age group was of interest, then applying scalars of 1.8, 0.8, & 0.15 for paediatrics, adults and geriatrics respectively to the ICRP risk estimates would suffice (5). Applying this approach to the data in this study (which included CT examinations and fluoroscopic examinations) gave a collective risk estimate within 10% of the age-specific estimate, thus giving support to this simple solution.

CONCLUSION

While patients in diagnostic radiology may be unrepresentative of the general population, assessment of their collective risk can be assessed with sufficient accuracy using effective dose and the risk coefficients for the public given in ICRP 60.

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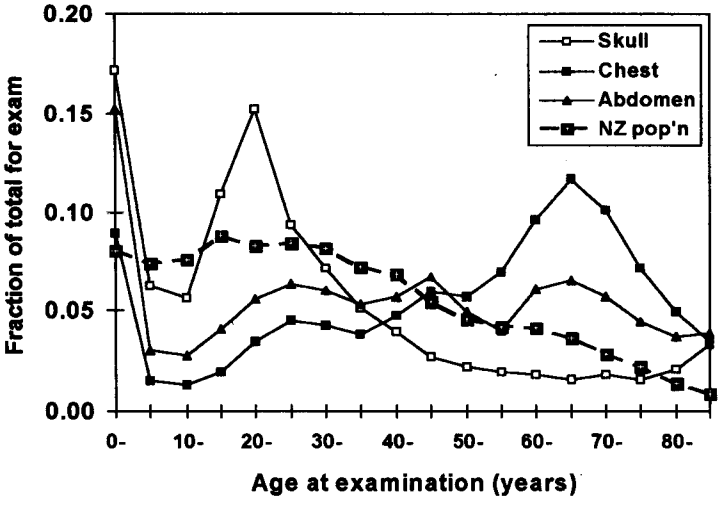

Figure 1. Relative proportions of patients for three x-ray examinations in age bands compared with the New Zealand population.

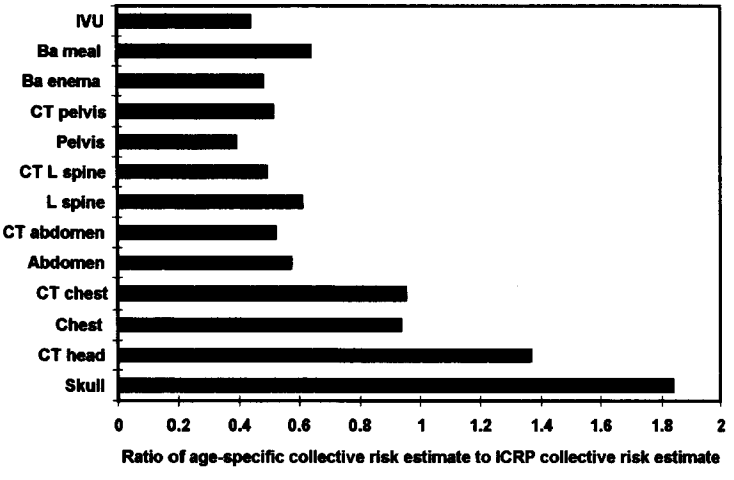

Figure 2. The ratio of age-specific collective risk estimate versus the ICRP estimate for specific x-ray examinations at a teaching hospital.

QUALITY ASSURANCE PROGRAM IN DIAGNOSTIC RADIOLOGY

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INTRODUCTION

One hundred years after the discovery of X-rays, they continue being nowadays part of physicians' daily activities, and the diagnosis throughthe use of X-ray equipments is one of the most important fields in clinical medicine, thus becoming the most important cause of human exposure to artificial sources.

For this reason, in the last twenty years, most of the developed countries did the utmost to establish programs which could warrant the quality of the radiographic image. Aiming the protection of human being against the harmful effects of ionizing radiations, in December 1980, World Health Organization decided to form a group of professionals highly experienced in medical radiology, and initiate an inspection and quality control program. In September 1988, the Group of Studies of the Program Related to Radiological Protection of the Commission in the European Communities, prepared a working paper in which guidelines were set up regarding quality of images, dosage to patient, and associated radiographic factors, necessary to obtain acceptable radiologic performance.

In Brazil, efforts driven in this direction, guided by some equipment testing, starting in 1990, began to be more known. When the Director and the Head of Radiology in the Military Police Hospital of the State of Rio de Janeiro (HPM) learned about these efforts, they decided to contact the Institute for Radioprotection and Dosimetry (IRD) of Comissão Nacional de Energia Nuclear and submit the problem of low radiologic performance and increasing rates of rejection.

Thus, with the coincidence of interests and needs, along with a proposal from the Commission of the European Communities (CEC), IRD decided to offer the author laboratory support to elaborate a Quality Assurance Program (QAP) to be implemented in HPM.

METHODOLOGY

The first step for implementing the QAP was setting the objective, that is, setting a methodology which maximizes the probability of obtaining good diagnostic quality images, allowing to reach high performance level of the radiology service.

In order to reach this objective, some administrative goals were fixed, such as improving patient's care, improving qualification of professionals, and creating a working environment based on shared responsability and solidary participation; and inspection goals, such as obtaining and maintaining high quality standards in each step of the radiographic process. All this aiming improvement of the quality of the image, optimizing dosage of each examination and reducing costs.

After establishing the objective and fixing the goals, there was still the problem of how to determine the more adequate way of implementing the QAP. Obviously, everything could not be changed overnight, nor could changes be imposed, without misorganizing in force proceedings. Thus, in view of the need to face all steps of the radiographic process, and considering that QAP should aim also patient's satisfaction, from the clinical standpoint, where the first issue is care, a methodology was worked out in such a way that it can be seen from an administrative standpoint (set up conditions to implement QAP; motivation and adoption of a working system aiming changes in values and attitudes; creating and bringing into practice flow control tables and tables for analysing film losses), and from a technical standpoint (set up of procedures to interpretate lost radiographies, and to get quality inspection and administration; data collection regarding radiographic techniques, size of patient, dosage on surface entrance and quality of image; refresher course and set up of image criteria). Finally, it was considered the adoption of corrective measures for each step of the program, analysis of images before and after set up of image criteria, and establishment of economical advantages of the QAP in diagnostic radiology.

RESULTS

Reduction of Rejection Rates

Rejection rate of radiographies ocurred in a radiology service is an indication of the constancy with which the quality of the approved image is obtained, and therefore, is related to efficiency. The first step when QAP was implemented was the control of this rate that suffered successive reductions as shown in figure 1.

Causes of Rejection

The advantages of QAP are highly dependent of the correct interpretation of the causes that lead to repeating examinations. In order to identify these causes, the technique of repeated exploitations was adopted, which can be described as a critical evaluation of radiographies which are produced by the radiology service during the activity of image production, but that, however, do not contribute for the process of diagnosis.

This analysis produced a large range of information about the efficiency of image production activity of the radiology service, and demonstrated when a certain area served by a particular process of the film had an elevated rejection rate. As shown in figure 2, one should first identify whether the cause of rejection was due to equipment, the film, the patient, or to film development, by studying in depth within each group, and doing a more detailed study of the causes of error.

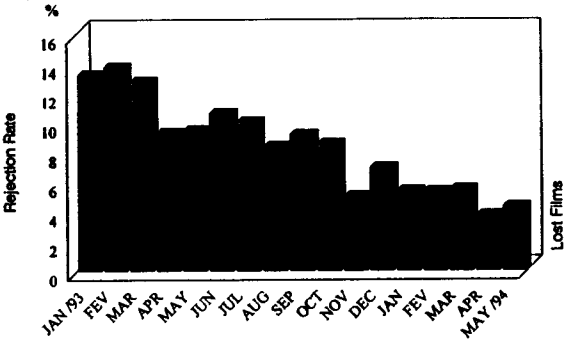

Fig. 1 - Reduction of rejection rate of radiographies

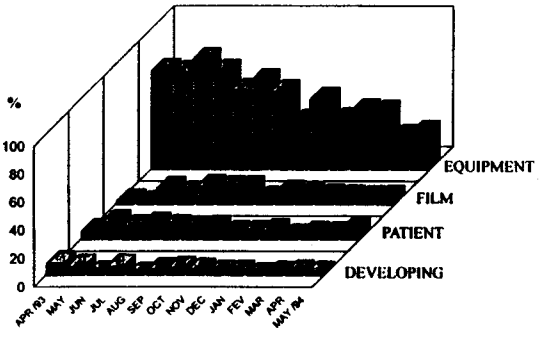

Fig.2 - Evolution of film rejection causes

Cost Reduction

To get a picture on how cost evolution was progressing due to lost films, as shown in figure 3, we calculated the average cost of each square meter of radiography produced in the department, and the amount of square meters lost every month.

Optimizing of Dosage

The dosage absorbed by the tissue in a certain examination depends mainly of the technical factors employed in the radiography, of the characteristics of the equipment, of the X-ray beam, of the number of exposures, and of the delay of exposure. Comparing dosage obtained before and after applying quality criteria, as shown in figure 4, one can notice a significant reduction of dosage in all examinations. Amongst the factors that contributed to this reduction, is the utilization of a radiation field of smaller dimensions, and the type of radiographic technique that was used.

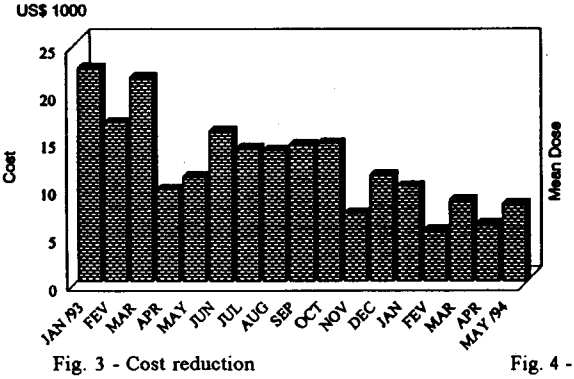

Fig. 3 - Cost reduction

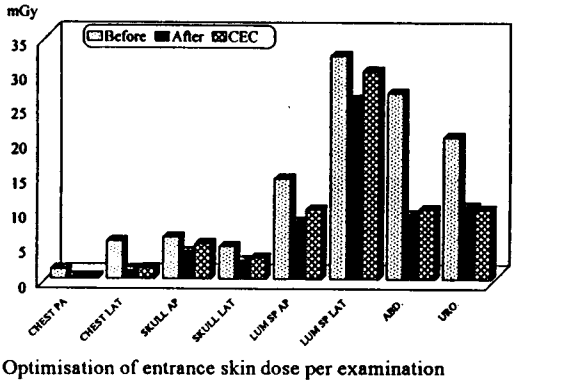

Fig. 4 - Optimisation of entrance skin dose per examination

Improvement of Radiographic Image

Based on criteria made up by the CEC, and having the support of one of their specialists, specific criteria for each of the normalized projections were created. In a comparative study between images obtained before and after

QAP, radiographies were observed by radiologists from HPM itself, acting as judges, attesting and quantifying the visualization of anatomic details of interest, according image criteria. As an example, the result of this evaluation in the PA chest examination is shown in figure 5 and 6. Generally, the results show that little distances focus-film, low V and high mA.s lead to contrast defficiency, lack of light and medium tones in important area, details mildly expressed, no clearness, images of low density, and excessive magnification of the object in the radiographies, all this before utilization of QAP.

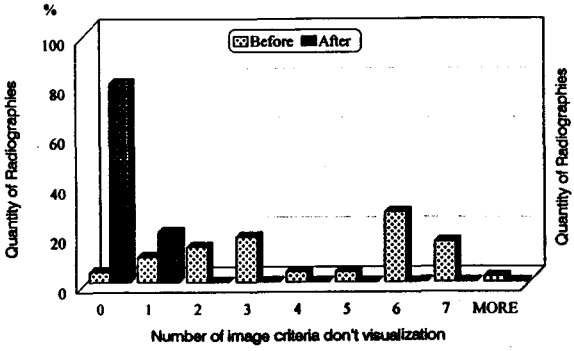

Fig. 5 - Visualization of imge criteria in the PA Chest examination, before and after QAP

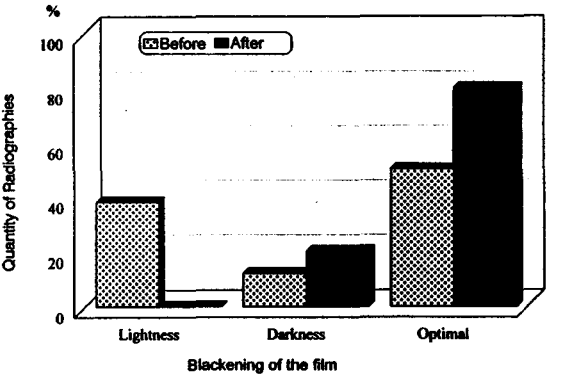

Fig. 6 - Evaluation of blackening of the film in the PA Chest examination

CONCLUSION

It is pertinent to mention that during the period of implementation of the QAP, there was always the intent to exercise a kind of leadership which aimed motivation of professionals, through friendly methods of consciousness, orientation and capability. To solve problems, new ideas were stimulated and took advantage of creativeness, resulting in a positive relation between the degree of participation and the feelings of satisfaction, responsability and compromise. The analysis of films rejection, besides allowing an evaluation of problems which lead to a poor quality image, also was useful as a tool to improve technicians' performance and as a basis for overall guidelines for the improvement of image quality.

The lessons that were given fully achieved their objectives, motivating people and contributing in a decisive way to the quality of the image, to optimized dosage, and reduced rejection rates and costs. They were also useful to incline professionals to like the changes, as much as they rejected them in the past, starting to think of the need of training, self-improvement and constant changes.

Settling quality criteria, besides allowing a uniform evaluation of radiographies, also forced radiology doctors to make a more complete and detailed evaluation of final images.

A significant reduction in dosage was found in all evaluated examinations, reaching 80% in the chest examinations, and 30 to 60% in the other examinations. It was also found a reduction of 70% - from initial 14% to below 5% - in the rejection of films, and a cost reduction of about 75%.

In the evaluation of the gotten images, before and after of QAP, all established criteria could be seen in a high percentage of radiographies obtained after the use of the new techniques, demonstrating an effective improvement in the quality of images.

It was confirmed that, with serious and persistent work, one can offer to patients services and products within their expectation, even in a public hospital. Therefore, such programs should have the support of related authorities, not only because of its economical and technical aspects, but also because of the social ones.

As a final comment, it must be mentioned that many things were made correctly, but some errors were made. But, above all, there was persistence. And, if good results were achieved, these are due, with no doubt, to hard work.

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PROPOSAL OF DOSE CONSTRAINT VALUES TO THE PATIENT IN DIAGNOSTICC RADIOLOGY

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INTRODUCTION

A dose constraint is the value of an individual dose not to be exceeded in the individual dose distribution considered in an optimization process. The objective of a dose constraints is to set a ceiling to the doses to individual from a source, practice or task which are considered acceptable in the optimization process at the design stage. Implicitly, as C. Zuur (1) states, dose constraints are below the relevant dose limit and usually should be established as local or national levels. Exposures for medical purposes are not subject to dose limits and hence dose constraints were recommended by the ICRP just for occupational and public exposures. However, as an effective tool for optimization for medical exposures, ICRP-60 in §180 (2) recognizes the value of applying this concept to patient diagnostic radiology with some peculiarities: "Considerations should be given to the use of dose constraints, or investigation levels, selected by the appropriate professional or regulatory agency, for application in some common diagnostic procedures. They should be applied with flexibility to allow higher doses where indicated by sound clinical judgement".

As B. Wall indicates (3), there is some interesting features in this recommendation: it suggests that for diagnostic medical exposures the dose constraints are to be used retrospectively as investigation levels for checking existing practices an not at the planning stage, as is the case for occupational and public exposures. In this practical application of ICRP concept of dose constraints to medical exposures of patients, the term reference level (4) or guidance level (5) have been indistinctly used. Unless justified by clinical judgements, appropriate action should be taken by the centres to improve practice; this could involve changes in technique or equipment to reduce doses to values below the reference level without compromising the quality of diagnostic information. When used in this manner, this has the same function as the "dose constraints".

National and sometimes local reference dose levels usually are set towards the top end of the observed range of typical practice, i.e. defined at a certain percentile such as the 75, to identify where corrective action is most urgently required. This concept is usually used at the first stages of optimization in diagnostic radiology as a level that should be obtained, however, measured doses at any other optimization stages should be clearly below, so that, new *constraints* should be imposed.

This paper analyses retrospectively the dose levels imparted to patient in some common examinations (chest, lumbar spine and mammography) at different optimization stages of different facilities to propose some local constraints for diagnostic examinations. Dose values have been obtained under routine working conditions. Centres included in the survey have been chosen all over Spain, classifying them with particular attention to the following aspects:

- Organizational aspects of the diagnostic radiology service, i.e., operational, technical and clinical criteria, as well as quality requirements.
- Evaluation and revision of routine medical protocols.
- Quality control of the radiological equipment.
- Quality criteria for the surveillance of the working procedures, with requirements of proper training of the technical staff.

MATERIAL AND METHOD

The examinations included in the survey were of the chest (postero-anterior and lateral projections), lumbar spine (AP and lateral) and mammography, since they are representative of simple examinartions (4) and hence coul be considered as *common* diagnostic procedures. Patient entrance surface doses were mainly estimated

from DAP (dose per area product) measurements carried out with transmission ionizing chambers. Alternatively, some data were obtained with thermoluminiscent dosimeters placed at the centre and on the surface of the patient field. A previous chambers and electrometers intercomparation among the participating centres were carried out to ensure that all measurements were comparable. Data are obtained from samples higher than 10 patients in each facility. Paediatric patients were not included in the survey. Third quartile values are shown in tables.

RESULTS

Tables summarize dose levels of the aforementioned examinations and the characterisctics or degree of implementation of quality assurance programmes in each facility. First level (which correlates with the highest dose reference values) corresponds to data obtained by the Medical Physics Group of the Complutense of Madrid (6) at an early optimization stage of radiation protection in diagnostic radiology. This group conducted a research proyect in the community of Madrid during the period 1986-1992, partly supported by the Eropean Community, and carried out in several health centres. Those first data could be representative of the dose reference values without any implimentation of quality control programmes (most data were obtained before 1989). Rest of data included in table came from services with different degrees of quality assurance implementation. Lowest local reference values could be considered representative of a facility in which all organizing aspects of the radiological services have been taken into account, also staff training on radiological protection topics and equipment quality control.

OPTIMIZATION LEVELS	L1	L2	L3	L4
1. Quality criteria for the surveillance of the working procedures, with requirements of proper training of the technical staff		X	X	X
2. Quality control of the radiological equipment.		X	X	X
3. Organizational aspects of the radiodiagnostic service i.e. operational, technical and clinical criteria, as well as quality requirements			X	X
4. Evaluation and revision of routine medical protocols				X

MEAN VALUES OF PATIENT SURFACE ENTRANCE DOSE (mGy) CLASSIFIED ACCORDING TO LEVELS OF OPTIMIZATION

Examinations	L1	L2	L3	L4	EU [1]	UK [2]
Chest PA	0.50	0.35	0.18	0.12	0.30	0.20
Chest LAT	1.60	1.20	0.56	0.70	1.50	1.50
Lumbar Spine AP	20.00	11.10	7.51	3.28	10.00	9.20
Lumbar Spine LAT	45.10	22.70	17.90	8.53	30.00	22.80
Breast	13.90	5.60	4.48	-	7.00	10.00

[1] CEC reference values [2] Ref (7)

CONCLUSIONS

The proposals of the introduction of dose constraints in diagnostic radiology could be a solution to ensure that patient doses do not exceed the level considered has optimum for each examination.

Each facility according to the different optimization levesl that could be achieve, should be adopt their own constraints values as presented in this paper, however, it is important to be account that the imposed levels could be improved while getting new optimization procedures.

Until now the best level of optimization (L4 in table) achieves was due to the good practice of the complete radiological process, including all from the moment an examination is asked to the final clinical report is issued.

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ASSESSMENT OF RADIATION DOSE AND QUALITY ASSURANCE IN MEDICAL X-RAY DIAGNOSIS.

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ABSTRACT. The intensive employment of X-rays in diagnostic radiology,represent so far the largest man-made source of public exposure to ionizing radiation."The aim of radiation protection of the patient has gradually shifted from a concern about population exposures and hereditary effects, to the ambition of limiting the risk to the individual patient" (1).

Key words:patient doses,radiation protection, quality assurance.

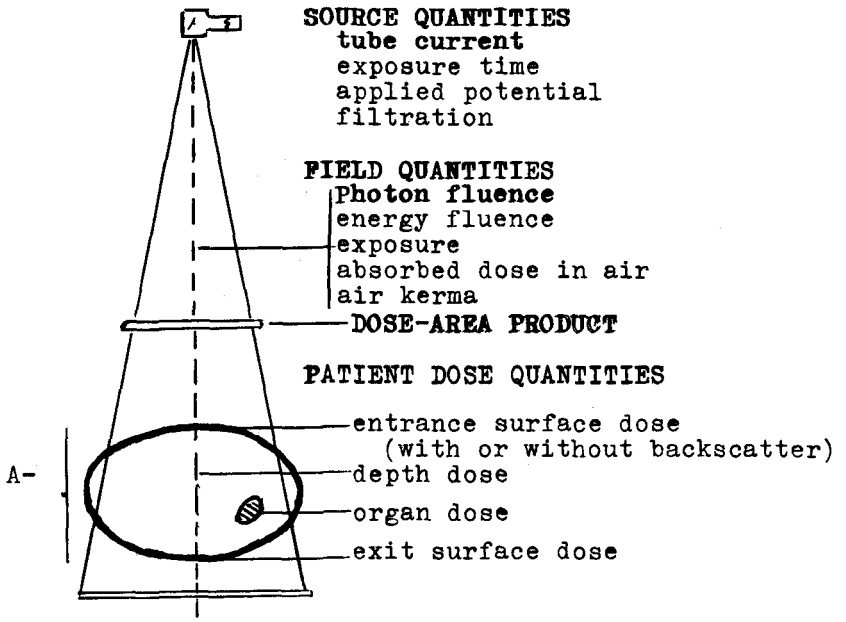
INTRODUCTION

By measurement of radiation is known the cause ionization within matter.Are used the general units of mass and volum. The concept of a give dose of radiation producing a particular effect is no different from that which applies in the medical administration of drugs.The radiation dose is an electrical phenomenum and is measurable using appropriate instrumentation e.a. various types of detectors available such as film badges,geiger counters,scintillation counters etc.Readings from these instruments may be translated into a dose of radiation absorbed by a unit mass of tissue (2).

MATERIALS AND METHODS

In order to be able to assess patient doses are necessary the equipment characteristic of X-ray examinations, dosemeters,tissue-equivalent phantom,water phantom and the competence technical personnel.

Include details of technical medical procedures by the EUROPEAN GUIDELINES of radiation protection of the patient. The biological damage of ionizing radiation,stochastic effect,which implies a single "hit" or repeatedly doses, effect which can be visible tardy,for all technical practical procedures in X-ray examinations impose respect of potential exposure by the justification of a practice, the optimisation of protection -ALARA principle-,and used of individual dose and risk limit for reducing doses without loss of diagnostic information (3).For each patient, individual -specific sex,age group- the radiation dose should be applied with flexibility and combined with dosimetric quantities,data on field size,field position,



A-**energy imparted**

focus distance and beam quality -KVp,filtration=the radiation quality- .

Patient dosimetry (4)

EXPOSURE contents ions which are generated in air,as a consequence of radiation.Ions can be mesured with an ionization chamber,which is an air space between two conducting plates coupled to the positive and negative poles of a voltage source.The exposure,means,the number of ions with negative or positive charges divided by the mass of air in the ionization chamber.

SOURCE QUANTITIES-that are selectable and often indicated on the equipment.

FIELD QUANTITIES -some of which can be measured with suitable instruments.

PATIENT DOSE QUANTITIES -which are mostly incapable of direct measurement,but which are the quantities that we realy want to know to assess the level of harm.

The quality and quantity of the radiation emerging from an X-ray tube,is a function of the tube current,exposure time,applied potential and filtration which give rise to a radiation field propagating through air that can be specified in term of the photon or energy fluence.

Radiation intensity,as well as light intensity,decreases in inverse proportion to the square of the distance from focus.Dose decreases also without matter,even in air.

The photons interact with the air to produce measurable values of air kerma or dose absorbed in air,at any point in the X-ray beam or integrated over the beam area -dose area product-.These "free-in-air" doses can be measured with ionisation chambers of typically between 3cm and 60cm volume for "point" dose measurements and of a large area,typically 400cm,for dose area product measurements.

PATIENT DOSES

By attenuation the interaction of X-ray beam with patient produces a non-uniform distribution of absorbed dose within the body,since the beam is localised to the area of interest and is attenuated as it passes through the body.In conventional film-screen radiography,the size of the patient will be influenced patient doses,since larger patients require higher entrance doses for the same dose to the image receptor.Measurable quantities during X-ray examinations are (5):
-entrance surface dose per radiograph,with termoluminescent TLD stuk directily to the patient's skin,in which case the measurement will include radiation back-scattered from the body.Entrance surface dose is called incident dose.
-dose area product per examination
-effective dose,concept which describes the probability of damage to different organs with their weighted sum,which are the radiosensitive,such as gonads,bone marrow,lungs,colon, brest etc.The sum of the weighting factors equals to 1.
The relative depth dose values strongly depend on radiation quality and geometric conditions.Parameter influencing organ doses are:patient size,tube voltage,filtration,field size, field position,focus-skin distance,scatter-gruid,sensivity of the film-screen combination.Dose calculations for organs and tissues are normally carried out in two steps:
1.calculation of dose free in air by dose measurements,dose calculations can be base on source data -source concept,or on the known or estimated dose at the image receptor -receptor concept,which is preferably used in those cases where the mAs prodoct is not known.
2.conversion of dose free in air to the dose in organs or tissue.

QUALITY ASSURANCE PROGRAMME IN DIAGNOSTIC RADIOLOGY

The quality assurance is a strategie for increase quality of diagnostic images and keeping patient exposure at adequately low level (6),after German regulations QA programme means the acceptance tests for X-ray machine and film processor, the acceptance constancy tests -the result of consecutive constancy checks can be compared,and supervised by committees established by professional medical organizations.

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UNNECESSARY X-RAY EXPOSURES:SPOILT FILMS AND THEIR CONTRIBUTION TO ANNUAL COLLECTIVE DOSE

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INTRODUCTION

Medical X-ray examinations in Romania give rise to a large annual collective dose equivalent of about 14,200 man Sv (90% of all manmade sources) [1]. The extent to which the unnecessary medical radiation contributes to this collective dose was not investigated and on a national scale the response is very difficult to obtain.This work is an attempt to estimate, even rough, the current extent of unnecessary exposures due to repeated examinations and of those with poor image quality.

METHODOLOGY

Twelve X-ray diagnostic departments from seven medical centers were selected on their annual workload basis for the audit. Because the rejected radiographs are not recorded separately, we have chosen a four month period (the middle month of every quarter) during 1993, to make an analysis of these spoilt films. The annual repeat rates of radiographic examinations were calculated on this basis, for each hospital. A random sample of 800 films from all X-ray departments was separately viewed by three radiologists. According to their own opinion, the examinations divided between the following categories of a three step scale: very good for films having no errors of exposure,positioning or processing; good or diagnostic acceptable for films with some errors of exposure, positioning or processing but which do not detract from the diagnostic utility of the radiograph and unacceptable for films having such errors that made than unsuitable to diagnostic. We used this last category of radiographs or spoilt films to make an evidence of clinically unhelpful X-ray examinations because the probability of obtaining an useful information for patient management is extremely low.

RESULTS

The four month sample involved 11819 patients which had a X-ray examination; the overall number of films used was 21330 and the percentage of patients with a repeated exposure ranged from 0.53 to 5.70. The main causes of a repeated film which form part of a X-ray examination are divided into the following categories (A-D) :

A - faults due to personnel (radiographer) : positioning or processing ;
B - machine faults : overexposure, underexposure (errors of exposure) ;
C - films with faults of manufacture : stains and traces revealed by development ;

D - consultant's request.
The annual frequency of repeated examinations with their causes in each medical center is presented in Table 1.

Table 1. Annual frequencies of repeats with their causes

Hospital	Main causes for a repeated film (%)			
	A	B	C	D
1	6.2	79.3	6.2	8.3
2	41.6	41.6	-	16.8
3	100.0	-	-	-
4	5.9	29.4	64.7	-
5	14.8	59.3	25.9	-
6	16.6	16.6	66.8	-
7	23.1	61.5	15.4	-

The exposure errors seem to be the major fault accomplishing the higher percentage of repeated films (up to 79%) in almost all hospitals. Radiographic films with faults such as stains and traces which appeared by processing the film are another cause for a repeated examination.
The faults in positioning included also the selection of wrong beam limiting cone or diaphragm. Except one hospital (for lung diseases) where all repeated examinations were due to this kind of error, the radiographers'faults had a lesser effect on repeat rates. From the eighteen commom X-ray examinations observed, only five had relevant repeated rates : lung, heart (21.2%); lumbosacral spine (16.9%); i.v. urography (21.1%); gastric series (21.2%) and extrimities (8.5%). The frequency of rejects by cause and type of examination is present in Table 2.

Table 2. Frequencies of repeats for some examinations

Examination	Categories of repeats causes (%)			
	A	B	C	D
Lung,heart	22.9	54.2	20.0	2.9
Lumbosacral spine	7.1	78.6	14.3	0
I.V.Urography	30.0	25.0	35.0	10.0
Gastric series	0	80.0	0	20.0
Extremities	14.3	35.7	50.0	0

These data reveal that the examinations with higher patient exposure such as lumbosacral spine or urographies , having an important contribution to collective dose, have been repeated mainly due to machine faults, a kind of systematic error which cannot be avoided, because the radiological equipment is very old. On the other hand, the repeated examinations by cause C are easiest to eliminate. But apart the repeats, the most unnecessary medical irradiation arises from clinically unhelpful examinations [2,3]. In this category we have included the examinations appreciated by radiologists as unacceptable having such errors that cannot contribute to diagnostic decisions. Radiologist's impressions are summarized in Table 3.

Table 3. The radiologists'impressions on the radiographs (%)

	Image quality		
	Very good	Good	Unacceptable
% of the original sample	19.7±5.4	49.6± 3.2	30.7± 2.7
Type of examination			
Lung+heart	24.2±7.2	48.0± 5.5	27.8± 5.4
Lumbosacral spine	16.0±5.4	35.8±10.2	48.2±13.4
I.V.Urography	12.5±4.0	47.8±13.8	39.7±13.0
Gastric series	20.0±5.1	50.4± 7.8	29.6± 5.2
Extremities	28.9±8.3	53.4± 9.3	17.7± 5.9

There is an evident discrepancy between the percentage of repeats,as nowadays registered in radiology and the percentage of daily performed radiographs of poor quality,never recorded like that,but in fact as unhelpful to diagnosis as the former ones. Our results make an evidence for the need and the importance of a register of reject radiographs, a very simple and useful mean to control the radioprotection standards [4]. Estimates of the extent of the unnecessary component of the collective dose due solely to clinicaly unhelpful X-ray examinations and the repeated ones, are given in Table 4.

Table 4. Annual collective dose and potential dose savings

Type of examination	Effective dose/ film	Annual collective dose	Potential annual collective dose savings from :		
			Poor quality exams	Repeats	
	mSv	manSv	manSv	manSv	%
Lung,Heart	0.25	17.0	5	0.13	30.2
Lumbosacral spine	2.0	50.0	24	0.90	49.8
I.V.Urography	2.5	40.3	16	0.35	40.6
Gastric series	4.8	80.0	27	1.01	35.0
Extremities	0.1	6.1	1	0.02	16.7

Our results testify the potential for a significant reduction in the collective dose to patients (up to 49%) and a significant benefit to society as a whole by bettering the diagnostic quality and value of radiographs.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE: REDUCTION OF MALE GONADAL EXPOSURE DOSES DUE TO DIFFERENT DIAGNOSTIC X-RAY PROCEDURES IN BANGLADESH

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MAJOR SCIENTIFIC TOPIC NUMBER: 6.1. ..... (see page 7)

ABSTRACT (See instructions overleaf)

A series of measurements have been performed on the reduction of male gonadal exposure doses in various X-ray examinations of diagnostic radiology using a whole body human phantom. The exposure doses in the gonadal area have been measured by employing Thermoluminescent Dosimeters (LiF-100). The attenuation of exposure doses to gonad have been studied as a function of different thicknesses of lead plates, viz., 0.60 mm, 1.00 mm, 1.50 mm, 1.85 mm and 3.00 mm of sizes 20 cm X 20 cm for various X-ray examinations, namely, Oral Cholecystography (OCG)X-ray, Pelvis X-ray, Hip-joint X-ray, Intravenous Pyelography (IVP) X-ray and Barium-meal X-ray. It has been observed that the basic nature of all the exposure dose attenuation curves of lead (shielding material) are similar and there are no significant variations of exposure doses beyond the lead plate of 1.00 mm thickness. From this experimental study, a lead plate of thickness 1.00 mm has been selected to reduce the exposure-to-gonad doses adequately for different diagnostic X-ray examinations. In each case of the X-ray examination, the exposure reduction factor and the exposure attenuation efficiency for 1.00 mm thick lead shield have been determined. It is recommended that a 1.00 mm thick lead plate should be used as a shield to minimize the exposure doses to the gonads of the patients so as to improve the benefit-to-risk ratio from the uses of ionizing radiation.

PRACTICAL POSSIBILITIES FOR PATIENT DOSE REDUCTION IN DIAGNOSTIC RADIOLOGY

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INTRODUCTION

According to the United Nations Scientific Committee on the Effects of Atomic Radiation (1), the radiation doses from diagnostic radiology are the largest contribution to the collective dose from all man-made sources of radiation. In Romania, for a population of 23 million inhabitants, about 660 X-ray examinations were reported in 1990 per 1,000 persons (1), a value which represents a decrease in comparison with 1970 , when more than 1,000 exams per 1,000 were performed. Unfortunately, still about 26% of the total X-ray examinations are classical fluoroscopies (without image intensifier TV set), which give high doses to both patients and radiologists.

The general standard of the old equipment in diagnostic radiology is very far from the minimum requirements, so that great financial efforts must be done in the near future, in order to improve the present situation.

METHOD

From 1991 to 1993, the Institute of Hygiene, Public Health, Health Services and Management - Bucharest participated to the co-ordinated research programme on "Radiation doses in diagnostic radiology and methods for dose reduction", jointly organized by the International Atomic Energy Agency and the Commission of the European Communities (2).

The patient entrance surface doses (ESD) including backscatter for several X-ray projections were determined, using thermoluminescent dosemeters, before and after application of corrective actions for dose reduction (QC).

RESULTS

The main results are presented in Table 1.

CONCLUSIONS

By increasing kV and reduction of mA.s values several entrance dose reductions can be obtained, particularly for chest PA radiography, but the use of high kilovoltage technique is dependent on radiological equipment available.

An important reduction of the integral dose of the patient was determined by collimation of the beam (reduction of the field size to the investigated area).

The use of appropiate screen-film combination is another practical possibility for patient dose reduction in diagnostic radiology.

Table 1. Measurements of patients entrance surface dose.

HOSPITAL	RAY ROOM	EXAMINATION	DOSE PRIOR TO QC (mGy)	DOSE AFTER QC (mGy)	DOSE REDUCTION IF ANY (%)	CORRECTIVE ACTIONS
1	1	Chest PA	0.95	0.76	20	Increase of kV and reduction of both mA.s and field size
2	2	Chest PA	0.77	0.69	10	Increase of kV and reduction of both mA.s and field size
1	4	Urinary Tract AP	17.98	9.29	48	Increase screen-film sensitivity
1	4	Urinary Tract PA	12.87	10.48	19	
1	7	Lumbar Spine AP	33.46	24.56	27	
1	7	Lumbar Spine LAT	41.75	35.66	15	

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IMPLEMENTATION OF SOME DOSE REDUCTION METHODS IN QUALITY ASSURANCE OF DIAGNOSTIC RADIOLOGY IN SOME HOSPITALS IN IRAN

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INTRODUCTION

Although quality assurance (QA) programmes have been implemented in medicine and in particular in diagnostic radiology in Iran since a long time ago through periodic inspections, education and training, determination of genetically significant dose (1), patient and personnel monitoring, etc., a systematic national programme has been recently implemented by sending a detailed questionnaire to about 1000 diagnostic radiology departments. In 1991, the participation in an IAEA co-ordinated research programme on radiation dose in diagnostic radiology and methods for dose reduction showed up to 79% reduction in entrance skin dose (ESD) by keeping or improving the image quality (2,3). As an extension of this work, the programme was implemented on 520 patients in 13 X-ray rooms of 11 hospitals in Tehran using the image quality criteria defined in a CEC Working Document (4). The relationship of ESD to radiographic technique and to image quality was analyzed. In this paper, the results are presented and discussed.

MATERIALS AND METHODS

Based on the returned 350 questionnaires, the per caput average number of examinations, number of films per examination, the frequency of certain examinations and the technical factors applied (mA, kV and exposure time, focus to film distance or FFD, field size, etc.) as well as the departments volunteer to implement the QA programme were deduced. From the voluntary departments, ten hospitals and one mammography clinic in Tehran were selected.

The projections of chest PA and abdomen AP in 12 X-ray rooms and four mammography projections including medio-lateral left and right and cranio-caudal left and right in one X-ray room were considered. A multi-function meter (RMI, Model 240-A), a Rad-Chech PLUS (Victoreen Model 06-526), a dual color sensitometer (X-rite, Incorporated, model 334), a B/W transmission densitometer (X-rite, Incorporated, model 331), a HVL attenuator set (RMI Model 115A), TLD-100 dosimeters (LiF, Harshaw) were used. The conditions of one darkroom in each department including light tightness, fog level, safe light, processor (developer, temperature and rinsing) and cassettes (speed, cleaning, light leakage and air trapping) were tested. The ESD of 520 patients were directly measured by TLDs on the patient's skin at the center of the X-ray field. The following procedures were completely implemented in selected departments:

1- Determination of ESD, kV, mA.s, FFD and image quality scores of 260 patients weighing 60 to 70 kg prior to QA including 10 patients for each selected examination (chest PA, abdomen AP, mammography CCR, CCL, MLR and MLL). The image quality scores was given by field radiologists based on the CEC image quality criteria (4); scores of 1, 2 and 3 respectively for poor, satisfactory and good image qualities.
2- Determination of means values of ESD, kV, mA.s, FFD and image quality score of each group of 10 patients stated in part 1.
3- Based on the information obtained in parts 1 and 2, necessary corrective measures were made reduction such as reduction of mA.s by improving the film developing conditions, reducing the optical density of the film and/or increasing the speed class of film screen combination as well

as optimizing the kV, filtration and FFD by using an anthropologic phantom (Randoman phantom from Alderson Research Lab. Inc.).

4- Repeating parts 1 and 2 after QA in part 3 for another 260 patients applying the same procedure.

RESULTS AND DISCUSSION

Although many QA parameters, both related to the equipment and techniques, seemed necessary to be corrected, due to some existing problems only some parameters were considered for corrective measures including radiological techniques such as increasing kV and FFD as well as decreasing mA.s. Also in a few cases, some other possible corrective measures such as improving the darkroom conditions, selection of best film screen combinations and improving total filtration were also applied. The implementation of the above led to a significant decrease in the mean ESD values while increasing the image quality due to changes in the technical parameters, as shown in table 1. It should be mentioned that in the case of mammography, only type of film and the anti-scattered grid were changed. Under the conditions applied, the mean ESDs have been decreased up to 72%.

Table 1. Effects of applying dose reduction methods on ESD and image quality.

QA parameters			Variations of the means (%) ± SD					
			Chest PA	Abdomen AP	Mammography*			
					CC-R	CC-L	ML-R	ML-L
Dose Reduction Methods	kV	increased	61±21	37±14	0±5	0±5	0±5	0±6
	mA.s	decreased	76±14	63±19	0±12	4±23	7±8	9±12
	FFD	increased	30±16	16±6	0±0	0±0	0±0	0±0
Image quality increased			42±20	38±9	30±10	30±10	30±10	30±10
ESD decreased			72±14	54±17	12±14	28±16	31±12	28±13

* CC-R=Crano Caudal Right, CC-L=Crano Caudal Left, ML-R=Medo Lateral Right, ML-L=Medo Lateral Left.

Figures 1 and 2 show mean ESD values of patients undergoing respectively chest and abdomen examinations in different X-ray rooms. It is observed the ESD has been significantly decreased after implementation of the QA in all departments. Figures 3 and 4 show the mean scores of image quality, respectively for chest and abdomen examinations. From these figures, it can be concluded that while Figs. 1 and 2 show significant ESD reductions, Figs. 3 and 4 show significant improvement in the image quality of the radiographs. Also table 2 shows the maximum, minimum and their ratios as well as the mean values of the ESD before and after the QA. From table 2, it can be also concluded that the range of the ESDs and in turn the maximum to minimum ESD ratios have been significantly reduced after the QA. It can be concluded that by adjusting the exposure techniques for different X-ray units and using a written protocol for these techniques, the requirement for repeating the radiographs due to over or under exposures will be reduced.

Regarding the rejected films, during the first set of measurements, about 500 films were analyzed by field radiologists from which 18.8% were rejected for repetition. The main reasons for repeating of rejected films included patient movement (5.4%), over exposure (4.2%), exposed by light (3%), under exposure (2.9%), positioning (1.2%), repeated radiographs by mistakes (0.8%) and developing conditions (0.4%). It can be concluded that by using a technical protocol, about 7.1% of the overall rejected films (due to over or under exposure) may be reduced.

As shown in table 1, the limited QA programme applied has been very effective in dose reduction in the X-ray units used in this study. Based on the data collected from the questionnaires, the frequency of chest and abdomen examinations were respectively 4.3 million and 1.6 million in 1994.

So reduction of 72% ESD for chest and 54% for abdomen can lead to a significant reduction of collective effective dose in Iran.

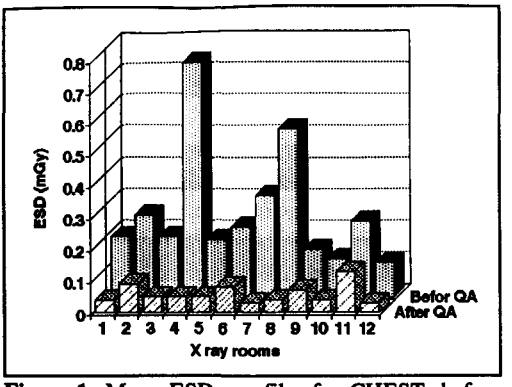

Figure 1. Mean ESD per film for CHEST, before and after QA.

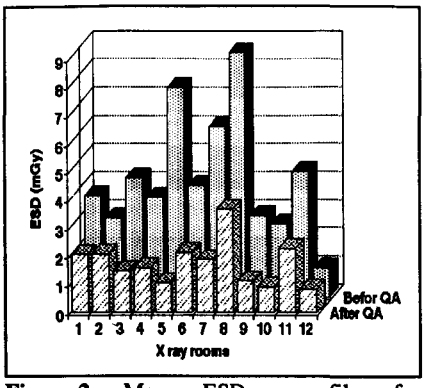

Figure 2. Mean ESD per film for ABDOMEN, before and after QA

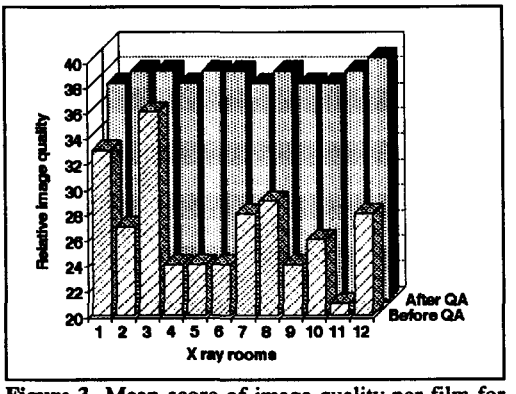

Figure 3. Mean score of image quality per film for CHEST, before and after QA.

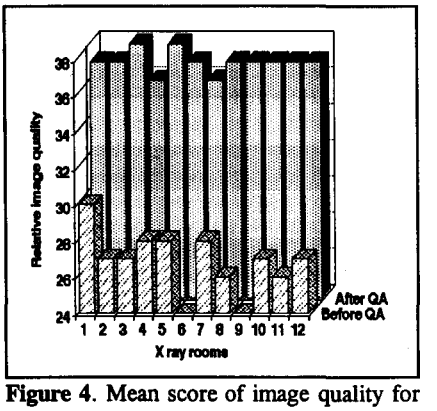

Figure 4. Mean score of image quality for ABDOMEN, before and after QA.

Table 2. ESD ranges by examination

Examination type	Min (mGy)	Max (mGy)	Mean±SD (mGy)	Max/Min
Chest PA before QA	0.05	1.69	0.27±0.23	34
Chest PA after QA	0.02	0.17	0.06±0.03	8.5
Abdomen AP before QA	0.26	13.73	4.23±2.77	52.8
Abdomen AP after QA	0.76	3.83	1.66±0.77	5

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ASSESSMENT OF PARAMETERS REQUIRED FOR OPTIMIZATION OF X-RAY DIAGNOSTIC PROCEDURES

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ABSTRACT:
The quality of diagnostic information provided by an x-ray examination depends on the quality of an x-ray image obtained on the film, intensifying systems or in through computer image reconstruction methods. The quality of the image is defined by two main parameters: degree of resolution and difference in optical densities, the latter being related to the type and thickness of a target tissue.
This study is an attempt to asses those parameters as simple and as reproducible possible. Within the framework of the study, the influence of conditions under which the image was obtained as well as the reproducibility of x-ray diagnostic procedures will be discussed. The results of this study were meant to serve as the basis for optimization of x-ray diagnostic examinations, respecting also the specifications of an available x-ray unit.

INTRODUCTION:
According to the latest survey in which we have completed our data base about the ionizing sources in the Republic Croatia, we have collected all available technical data about the medical x-ray facilities and x-ray units which are still operating.As the result was not encourageble, over 60 % of x-ray units older than 10 years, and as the financial potential of the country is not able to cover immediate replacement of units we were forced to reconsider our radiation protection situation in medical diagnostic and try to implement the CEC image quality criteria (1) and European standards in diagnostic radiology through three steps. First, we have carefully distinguished the x-ray units which are matching the minimum of requirements for good image quality from those which need permanent maintenance to be able to perform the good image quality. In the last five years we have checked every x-ray unit in the country, performing the quality control tests for radiation quality, image quality, patient dose and radiation protection (1-5). Second, we had to make a difference in methodology for the common x-ray units producing the film as the end product and the x-ray units producing the digitized image. The main group were common x-ray units. Dental units and mammography units are considered separately. Third step was to gather the data about the type of examinations performed on the units of interest.. We have decided to cover the five main types proposed by CEC (1). The pediatric situation is considered separately.
At the end we had a situation:

-over a 100 common x-ray unit which need a permanent maintenance because they are older than 7 years and/or the producer is not supplying the spare parts.
-three main manufacturers of these units sometimes not able to calibrate the generator and the whole system
-a big difference in patient dose between the units even of the same type.

There are no Health physicists in our hospitals. That's why we have tried to find the most reliable, quick and reproducible method of checking the parameters which represent the radiation quality of x- ray units, introducing the quality control programme into the daily work in the diagnostic radiology departments .
For the radiologist is the end product important, namely the image.

METHODS AND MATERIALS
The main parameters included in quality control procedures have definitely significant influence on the image quality. In order to find the method which can proper describe the status of x-ray unit we have decided to cut the influence of the film processing and intensifier screen systems. So we have made all our measurements with the film cassette with no intensifier screen at all and with the Al filtering phantom, water phantom 15 cm thick for every x-ray unit the same. Dose was measured with the Victoreen RAD CHECK +, film and TLD dosemeter badges. kVp was measured with Vicoreen kV meter, calibrated by Victoreen. For all possible quality tests we have used a set of phantoms, various step wedges, resolution patterns, etc..mostly produced by Victoreen.
The first step was always to check the actual size of the focal spot end at the same time the resolution which was always documented on the x-ray film This was done for every unit in the same geometry arrangement. After that, we have started to measure kV step by step within the interval from 60 to 120 kVp in steps of 10 kV. The measurements results where fitted with Statgraphics ver.4.2 package using simple regression method (table 1.)

TABLE 1.

kVp selected on the generator	measured kVp (kV)	Dose at 1 m from focus in mGy/mAs no add. filters	Dose at 1 m from focus in mGy/mAs 3mm Al add.filter
40	38.1	0.02307	0.00562
42	40.4	0.02562	0.00653
44	42.6	0.02831	0.00754
46	44.9	0.03114	0.00866
48	47.1	0.03412	0.00988
50	49.4	0.03724	0.01120
52	51.6	0.04051	0.01265
54	53.9	0.04393	0.01422
56	56.2	0.04749	0.01591
58	58.4	0.05120	0.01773
60	60.7	0.05507	0.01969
62	62.3	0.05908	0.02180
64	65.2	0.06324	0.02405
66	67.4	0.06756	0.02645
68	69.7	0.07203	0.02901
70	71.9	0.07665	0.03173
72	74.2	0.08142	0.03462
74	76.4	0.08635	0.03768
76	78.7	0.09144	0.04092
78	80.9	0.09666	0.04434
80	83.2	0.10207	0.04795
82	85.5	0.10763	0.05176
84	87.7	0.11334	0.05576
86	90.0	0.11921	0.05997
88	92.2	0.12523	0.06439
90	94.5	0.13142	0.06903
92	96.7	0.13776	0.07388
94	99.0	0.14427	0.07896
96	101.2	0.15093	0.08428
98	103.5	0.15776	0.08983
100	105.7	0.16475	0.09562
102	108.2	0.17190	0.10166
104	110.2	0.17921	0.10795
106	112.5	0.18669	0.11450
108	114.8	0.19432	0.12132
110	117.0	0.20213	0.12840
112	119.3	0.21009	0.13576
114	121.5	0.21822	0.14340
116	123.8	0.22652	0.15132
118	126.0	0.23498	0.15954
120	128.3	0.24361	0.16805
122	130.5	0.25240	0.17687
124	132.8	0.26136	0.18599

Second step was to measure the dose output through the whole kVp range achievable with the generator at the fixed geometry and fixed mAs values. Two sets of dose output were measured: one without any additional filtration and second with 3 mm Al additional filter.

DISCUSSION

The measurements were made to find the kVp where the 3mm Al represents HVL. At that kVp it is recommendable to make the image in such a way that half of the image was shoot once without any additional filter and the other half should be shoot twice using 3mm additional Al filter. The mAs value should be chosen so that the optical density of the film would be about 2.0 and identical on both halfs of the film. Any change in the generator operation parameters (kVp, mAs, etc) has direct influence to optical density mentioned above,by changing it. The whole series of measurements where done for small and broad focus size separately.

We used our own Kodak films always comparing it with the films which were used by users.:

This enables us to have a good overwiev over the once optimized radiation quality of the x-ray unit. The working staff has a simple method to verify the leak of quality theirs x-ray units.

The doses shown in table 1 are presented in mGy/mAs. So, it is always possible to evaluate the skin entrance dose multiplying the applied mAs with the dose value at given kVp in order to get the real dose applied to the patient during the diagnostic procedure. That's why those tables are attached to each x-ray unit.

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PAPER TITLE X-RAY DIAGNOSTICS: DOSE TO PATIENTS VERSUS TECHNICAL EXPOSURE PARAMETERS

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MAJOR SCIENTIFIC TOPIC NUMBER 6. (see page 7)

ABSTRACT (See instructions overleaf)

Doses to patients in X-ray examinations extremely depend on a choice of technical exposure parameters. Approximately the same visualization of anatomical details is possible to achieve using different combination of kilovoltage, charge (anode current and time product) and total filtration, but with quite different doses to patient. (Source -skin distance is designed by type of procedure.) In the newest X-ray units kilovoltage - charge combination is fixed automatically, but the older equipment needs to choice of these parameters "handly" - by technicians.

This paper presents results of evaluation how particular parameters can affect on dose to patients, taking into consideration the mode of X-ray generators. (Kilovoltage seems to be the most important from mathematical point of view.) As in Poland X-ray units with different rectifying system are in use, this analyse takes into account the following systems: half-wave, 6- and 12-rectifiers and constant potential generators. The influence of total filtration is significant as too low filtration was very frequently found error in Polish X-ray units.

As conclusion - a possibility of patient dose reduction is discussed versus an appropriate choice of X-ray equipment and their exposure parameters.

ASSESSMENT OF ORGAN AND TISSUE DOSES FROM COMPUTED TOMOGRAPHY EXAMINATION

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INTRODUCTION

Computed tomography examinations greatly contribute to the collective dose from diagnostic radiology. This is caused by increase in CT examination frequency and relatively high level of patient irradiation. Advances in design of CT scanners give better quality images but at the cost of increased patient exposure. Organ and tissue doses from CT examinations are usually assessed by Monte Carlo method and mathematical phantom (1). In this work anthropomorphic Alderson phantom and thermoluminescent detectors were used. Particular attention was paid to brain examination. In many CT installations the gantry is tilted during brain examination in order to limit eye lens exposure. M C method can not simulate this and as the result the eye lens and thyroid doses obtained by both methods differ considerably.

METHOD

The study was carried out using five different CT installations: CT/T 8800, CT 9800 and CT MAX made by General Electric; SOMATOM 2 made by Siemens and CT 1200 SX made by Picker. Five most frequent CT examinations were selected: brain, axial orbits, lung, liver and abdomen-pelvis (2). Lithium Fluoride (TLD-100) thermoluminescent dosemeters (3.2x3.2x0.9 mm) were used. They were read on Toledo 954 reader. Quality of the X-ray beam characterized by half value layer (HVL) and kVp value was measured for each scanner. The data were later used for TLD calibration procedure. Obtained values were:a.) HVL=8.5 mm Al, kV_p=125 kV for SOMATOM 2 and b.) HVL=5.5 mm Al, kV_p=120/130 kV for other scanners. Calibration was carried out in a water phantom with 0.6 cm^3 Farmer ion chamber type 2571 as a reference dosemeter. Correction factors from water to ICRU muscle and compact bone were calculated from conversion factors (3) for both X-ray qualities: a.) $f_m/f_w = 1.02$, $f_b/f_w = 3.06$; b.) $f_m/f_w = 1.26$, $f_b/f_w = 3.65$. Settings of nominal slice thickness and exposure (mAs) recommended by each manufacturer for each examination were used. Dose distribution resulting from CT exposure was measured in the phantom at the central part of irradiated area and these data were used for mean dose estimation to organs extending through many phantom slices (4). Since Alderson phantom is made close to Average Man specifications the results are expected to represent mean organ doses for patient of similar specifications.

RESULTS

Operating parameters of CT scanners for two groups of examinations are given in two tables. Organ doses are given in tables for each examination. Dose to skin is measured as local dose in the irradiated area. Red bone marrow is marked as RBM. Organs with doses less than 0.05 mGy were excluded. Radiation field limits for each examination are shown on figures adjacent to tables with organ dose results.

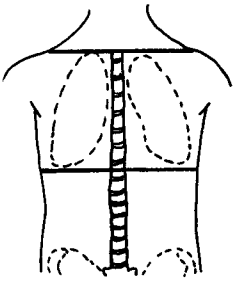
OPERATING PARAMETERS FOR LUNG AND ABDOMEN-PELVIS EXAMINATIONS

Scanner	Slice width (mm)	Lung examination		Abdomen - pelvis examination	
		Exposure settings (mAs)	Number of slices	Exposure settings (mAs)	Number of slices
CT 8800	10	288	23	368	22
CT 9800	10	420	23	510	21
CT 1200	10	240	20	330	20
CT MAX	10	365	24	365	20
SOMAT 2	8	230	31	330	24

For lung examination somatic risk is due to the dose to lungs, breast, bone surface, RBM and oesophagus lying in direct X-ray beam. Considerable scattered radiation dose is to the thyroid , liver and stomach. Genetic risk is minimal due to large distance from gonads to the radiation field.

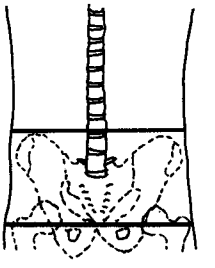
MEAN ORGAN DOSES (mGy) FROM LUNG EXAMINATION

Organ	CT 8800	CT 9800	CT 1200	CT MAX	SOMAT 2
Lung	14.4	33.5	26.6	28.6	26.0
Breast	10.4	26.0	38.7	35.1	37.4
Thyroid	1.4	10.1	4.7	7.2	4.3
Bone surface	1.8	6.3	4.9	5.3	5.3
RBM	2.9	9.9	7.8	8.4	8.4
Oesophagus	11.8	25.7	28.5	28.7	22.6
Liver	3.8	9.8	10.1	10.5	8.5
Stomach	2.7	6.9	7.1	7.4	6.0
Skin	10.9	33.4	32.6	27.0	32.5



MEAN ORGAN DOSES (mGy) ABDOMEN-PELVIS EXAMINATION

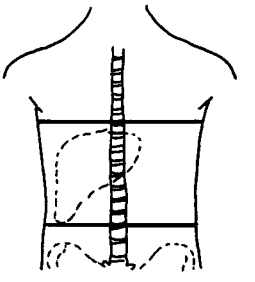
Organ	CT 8800	CT 9800	CT 1200	CT MAX	SOMAT 2
Colon	7.1	36.9	27.5	37.4	19.4
Liver	< 0.05	3.1	3.4	2.8	1.7
Stomach	< 0.05	5.4	5.8	4.9	2.8
Bladder	10.3	24.7	38.6	25.2	25.7
Uterus	10.0	24.0	36.7	25.0	25.3
Ovaries	10.3	24.7	38.6	25.2	25.7
Testes	1.6	2.7	6.5	5.9	3.5
Bone surface	2.2	7.6	9.2	6.5	5.6
RBM	3.7	13.2	16.0	11.7	9.7
Skin	12.8	41.0	56.4	30.0	44.3



Considerable genetic risk from this examination is due to high dose to ovaries which are subjected to direct X-ray beam. Testes with male patients are subjected to scattered radiation. Uterus dose was measured to assess conceptus dose in case when a pregnant patient was subjected to abdomen-pelvis examination. Dose distribution data from abdomen-pelvis exam. were used to calculate organ doses from liver examination.

MEAN ORGAN DOSES (mGy) FROM LIVER EXAMINATION

Organ	CT 8800	CT 9800	CT 1200	CT MAX	SOMAT 2
Oesophagus	1.7	5.5	5.9	4.7	4.8
Liver	12.4	39.9	43.3	34.5	35.5
Stomach	12.2	39.0	42.1	33.4	34.1
Bone surface	0.7	2.4	2.6	2.1	3.4
RBM	1.2	3.8	4.1	3.3	3.4
Skin	12.8	41.0	56.4	30.0	44.3

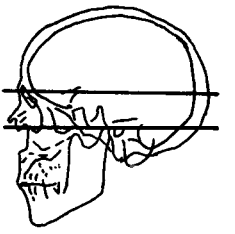


OPERATING PARAMETERS FOR BRAIN AND AXIAL ORBITS EXAMINATIONS

Scanner	Brain examination			Axial orbits examination		
	Slice width (mm)	Exposure settings (mAs)	Number of slices	Slice width (mm)	Exposure settings (mAs)	Number of slices
CT 8800	10	576	10	5	576	10
CT9800	10	600	10	5	420	10
CT 1200	10	240	11	3	300	10
CT MAX	10	365	11	2	365	12
SOMAT 2	8	518	13	2	518	14

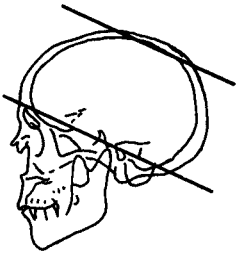
MEAN ORGAN DOSES (mGy) FROM AXIAL ORBITS EXAMINATION

Organ	CT 8800	CT 9800	CT 1200	CT MAX	SOMAT 2
Brain	0.3	0.4	0.2	0.3	0.3
Eye lens	26.2	43.5	36.1	41.0	52.6
Thyroid	0.3	0.3	0.3	0.4	0.5
Bone surface	0.7	1.3	0.7	0.6	1.0
RBM	0.8	1.5	0.8	0.7	1.1
Skin	19.1	46.3	30.7	31.7	49.7



MEAN ORGAN DOSES (mGy) FROM BRAIN EXAMINATION

Organ	CT 8800	CT 9800	CT 1200	CT MAX	SOMAT 2
Brain	25.7	47.8	30.7	36.6	40.3
Eye lens	9.4	13.0	19.0	9.9	9.2
Thyroid	0.6	0.5	0.4	0.6	0.8
Bone surface	1.9	3.6	2.3	2.7	3.3
RBM	2.1	4.3	2.6	3.0	3.6
Skin	23.0	55.5	28.8	33.7	48.2



COMPARISON OF DOSE TO THE EYE LENS (mGy) FOR THIS WORK AND (5)

	NRPB R-249		This work	
Scanner	Head	Axial orbits	Brain	Axial orbits
CT 8800	31.0	37.0	9.4	26.2
CT 9800	56.0	51.0	13.0	43.5
CT 1200	76.0	78.0	19.0	36.1
CT MAX	42.0	40.0	9.9	41.0
SOMAT 2	45.0	37.0	9.2	52.6

Comparison of dose to the eye lens for CT examinations of the head region done in this study and those obtained in (5) shows that results for axial orbits examination are reasonably close with the exception of CT 1200. The results for brain and head examinations differ considerably because of different irradiation geometry. In many CT installations the gantry is tilted during brain examination by about 20° and the eye lenses are outside the direct radiation beam. For other examinations the mean dose to organs completely covered by the primary X-ray beam measured in this work is between mean and maximum organ dose values obtained in (5) and corrected for mAs values. Dose to organs partly covered by primary X-ray beam is difficult to compare because it critically depends on radiation field limits setting by the radiographer.

CONCLUSIONS

Irradiation geometry should be specified when organ doses from CT head examination are reported to enable their comparison. Visualization of brain region in CT head examination does not have to involve eye irradiation. The eye lens dose can be kept resonably low by properly setting radiation field limits.

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PAPER TITLE DETERMINATION OF PATIENT SURFACE DOSE FROM COMPUTED TOMOGRAPHY EXAMINATIONS

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MAJOR SCIENTIFIC TOPIC NUMBER 6.1. (see page 7)

ABSTRACT (See instructions overleaf)

Computed tomography (CT) has become a major source of the population exposure to diagnostic X-rays, and the knowledge of the doses delivered by the CT equipment has become very important. Considerable efforts should be made to keep these doses to a reasonable minimum, without sacrificing the image quality. The conditions of exposure in CT are quite different from those in conventional X-ray imaging. This has required the development of specific techniques for assessing patient dose from CT. The aims of this work were to determine the doses delivered to various organs of patients undergoing CT of abdomen, thorax and head as measured on the surface of the body and to estimate the risk to the patients. Dosimetric measurements were performed at two different CT scanners (Siemens SOMATOM DR-H and Shimadzu SCT-4500TE). The doses absorbed by different organs (gonads, thyroid, chest and eye lens) and by the examined partsof the body of 90 patients of various sex and ages were measured with TLD-700.

The doses absorbed by different organs during the diagnostic CT examination of the body depended on the technical parameters, such as the number of scans, mAs, the thickness of scans, scanning times, tube voltage and other characteristics, some of which depended on the type and severity of illness. Clinical parameters, such as patient size and composition and patient cooperation with regard to control and motion, also influence the dose and the image quality.

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PAPER TITLE . Automation registration effective dose irradiation in medical radiological examinations.

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ABSTRACT

Medical radiological examinations make the greatest over background irradiation to the population. This fact results in to control and to restrict the exposure. The conception of "effective equivalent dose" (effective dose) -H is introduced into practice in order to control dose caused by irradiation of an organism during medical radiological examinations.

X-ray dose indicator(INDOR-S) is the complex of measuring and calculating equipment. X-ray dose indicator is used to define the effective dose equivalent(effective dose) during 61 types of radiological examinations.The apparatus connects to the electronic circuit of anode current definition of every radiological equipment.

Dose indicator fully automatizes the process of patient dose during any type radiological examinations. Dose indicator is conducive to essential reduction of patient irradiation facilitates the work of radiologist and his laboratory assistant. The apparatus gives an opportunity to choose the most sparing methods of patient radiological examinations.

RADIATION FIELD SIZES AND SKIN EXPOSURES IN ORAL RADIOGRAPHY

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INTRODUCTION

The increasing use of x-rays in preventive and diagnostic dentistry in Brazil has been cause of concern because dentists, in general, are not acquainted with the basic principles of radiation protection. Recently, the Brazilian Ministry of Health has urged the Departments of Health at the state level to develop actions to register dental x-ray units in their area of jurisdiction and to issue operating permits to those facilities which satisfy some basic technical requirements. On the basis of these recommendations the Instituto de Radioproteção e Dosimetria of the Brazilian Commission of Nuclear Energy has initiated a postal program to assess the performance of dental x-ray sets in the State of Rio de Janeiro (1). The postal kit used in that survey was similar to the one developed by the Bureau of Radiological Health of the U.S. Department of Health, Education, and Welfare (2). In continuation to that study, the Nuclear Energy Department of the Federal University of Pernambuco initiated a survey of dental x-ray apparatus to evaluate the operating conditions of that kind of equipment in Recife, the capital of the State of Pernambuco. The objectives of the survey were: a) to assess the degree of compliance of the equipment and procedures adopted by the dental practitioners in Recife with the accepted radiation protection standards, and b) to estimate the magnitude of the exposure to the patient resulting from a typical dental radiographic procedure.

METHODOLOGY

Data collection was made through both office visits and irradiation of dosimetric packs similar to the postal packs used at the survey carried out in the State of Rio de Janeiro (1). Each pack contains four LiF thermoluminescent chips (TLD-100), a 3-mm aluminum filter and two periapical films which are used to determine beam filtration, radiation field size, and skin dose to the patient. In each facility to be inspected the dosimetric pack was placed over a cubic water phantom and the end of the cone of the x-ray equipment was positioned over the center of the pack. The irradiation was performed by the dentist according to the procedure he (she) would use to get a standard maxillary molar radiograph. Therefore, it would be possible to compare the results of different technical procedures aiming to get the same radiographic picture. Besides, the dentists were asked to fill a form answering questions concerning the number and type of film used, the exposure time, the use of radiation protection devices, etc. The data gathered were used to determine patient skin exposure, radiation field size, and total beam filtration. This preliminary study comprised 76 x-ray sets, the majority of them made in Brazil. Most of the units operate at 50kV or 60 kV, since 70 kV x-ray sets are not common in dental offices in Recife.

RESULTS AND DISCUSSION

Figure 1 shows the range of beam diameters determined in this study. The data show that 34% of the units surveyed have beam diameters below the limit set by the Brazilian Association for Technical Standards - ABNT (4) and that 76% of the sets have beam diameters within the limit recommended by the NCRP-35 (3). It can be noticed that, despite the fact that most of the units are made in Brazil , the manufacturers adopted the NCRP limit of 7.5 cm as the standard for the beam diameter. On the other hand, 24% of the inspected units present field diameters exceeding the 7.5 cm limit. It means that a larger than necessary area of the skin face of the patient is being exposed, and that both patients and staff members receive higher radiation doses due to scattered radiation than they would receive if proper field sizes were used.

Figure 2 illustrates the distribution of skin exposures resulting from simulated maxillary molar radiographs taken in the offices surveyed. The results show that 67% of the x-ray units produce radiation exposures above the recommended value of 500mR (3). Moreover, 13.5% of the skin exposures were six times greater than the acceptable exposure for that type of radiography.

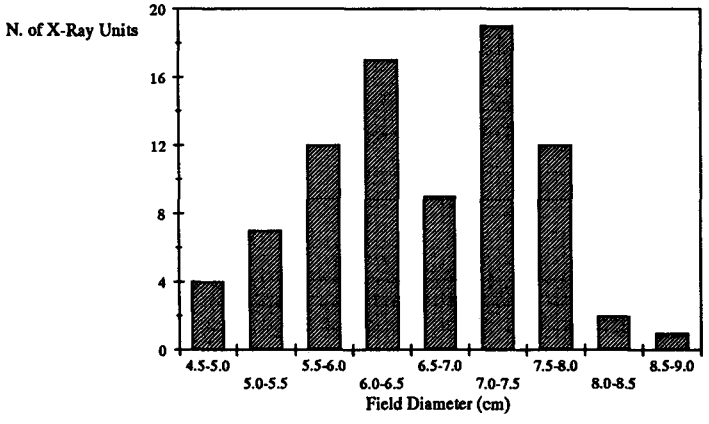

Fig. 1- Beam diameter distribution for the units surveyed.

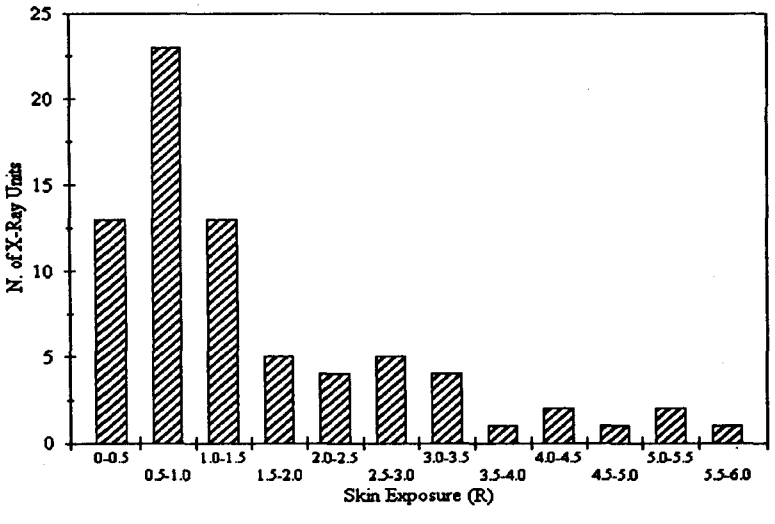

Fig. 2 - Distribution of skin exposures resulting from a standard maxillary molar radiography.

The causes for these excessive exposures are not different from those described elsewhere, specifically, insufficient filtration of the beam, too long exposures, and inadequate film processing. In fact, it was observed that dentists usually use old processing solutions and insufficient film developing times, compensating the inadequate procedure by increasing the exposure time. This is reflected in Fig. 3 that shows the distribution of exposure times used by the dentists surveyed. It can be noticed that 58.7% of the dentists adopt a exposure time around 1.0 s for the maxillary molar radiography, in spite of the fact that the majority of them use Ektaspeed film. Only 8.3% of the dentists utilize exposures of less than 0.5 s, which is considered to be adequate to get images with the Ektaspeed film without loosing radiographic quality. These results support the necessity of educating the dentist with respect to both film processing and radiation protection procedures.

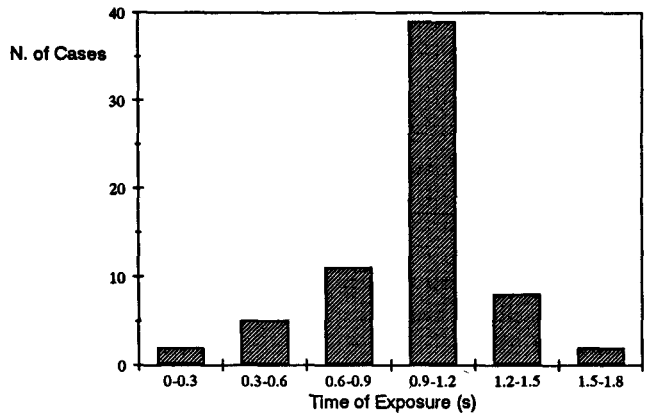

Fig. 3- Distribution of the exposure time used by the dentists surveyed.

The use of adequate beam filtration was also analyzed. The standard adopted recommends that the total filtration of the useful beam should not be less than 1.5mm Al for x-ray units operating at potentials between 50 and 70 kV [3]. However, the results show that 43.6% of the x-ray units surveyed do not meet this standard. The inadequate filtration results in an increase in the dose to the patient and to the dental staff, without contributing to the quality of the radiograph. Both the skin exposure and beam filtration distributions found in this survey are similar to those found by Peixoto and Ferreira (1) in the postal program developed by the IRD, in Rio de Janeiro. These findings reinforce the need of implementing a nationwide program to control the operating conditions of the x-ray dental units in Brazil. This program must be complemented by the training of dentists in the radiation protection procedures.

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ASSESMENT OF ENTRANCE SURFACE DOSE (ESD)VALUES IN RELATION TO THE TECHNIQUES USED IN DIAGNOSTIC RADIOLOGY

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Resumen: El uso efectivo de los rayos x pretende lograr una buena calidad de imagen con la menor dosis de radiación posible, y con unas técnicas radiológicas óptimas.

El objeto del presente documento es determinar la dosis real a la entrada del paciente en diferentes tipos de exploraciones simples llevadas a cabo en varios hospitales españoles entre los años 1993 a 1995 comparándolas con las técnicas usadas en radiodiagnóstico. En estos hospitales, que constan de un total de 4600 camas, con 258 unidades de rayos x atendiendo a una población de alrededor de 2 millones de personas, han desarrollado un programa de optimización para reducir la dosis a paciente.

Se analizan un total de 2542 examenes de pacientes reales. El único criterio de exclusión, fue la edad, no considerándose pacientes menores de 14 años. Las datos valorados en este estudio, corresponden a aquellos en los cuales hay un mínimo de 10 pacientes por tipo de examen y sala.

1.- Introduction

The Commission of the European Communities in the document " Quality Criteria for Diagnostic Radiographic Images" (1), has proposed reference values for patient entrance doses that are representative of the radiation dose for an average-size patient.

A first analysis of the obteined data, allowed us to compare our results with the reference ones (3), establishing some initial criterion for the radiodiagnostic labour optimization.

But among all criteria we are paying special attention to the methodology used by the operators. In (4) we could see the influence of the collimation affair in the obtained doses, and the proposal of one index (area index), that completes the information.

The aim of this paper, is to value the influence of two other technical parameters, which selection is only controled by operating staff.

2.- Material and Method

The radiographic examinations chosen for this study were those of the lumbar spine (A.P. and LAT.), pelvis (A.P.), skull, and abdomen.

For the estimation of the entrance surface doses presented in this document, we have only considered the data of dose x area product measured, in each exploration, with the transmission camera. The method followed in the calculations is well decribed in (2), (3) and (4)

The data were taken from 962 real adult patient examinations (excluding the ones below 14)

We have only used those measurements for wich we have had at least 10 patients per examination and room.

We calculate for all the explorations in this study, the average value of surface entrance dose considering all the patients in each one.

Unless the thorax explorations, we define the optimum values for the KV and FFD (focus-film distance) for the rest ones, admitting in all cases a margin of tolerance. In the table 1 we can see the number of the total patients considered per esploration, the optimum values and margins for the studied parameters, and also the number of patients who are within each margin or both

EXPLORATION	Total	kV opt.	Number vs. kVopt	FFD opt	Number vs. FFD opt	Number vs. kV & FFD
Skull	**126**	> 70	**102 (81%)**	[95-105]	**33 (26%)**	**23 (18%)**
Pelvis	**174**	> 70	**103 (59%)**	[90-110]	**100 (57%)**	**92 (53%)**
Abdomen	**244**	> 70	**149 (61%)**	[90-110]	**118 (48%)**	**66 (27%)**
L. Spine A.P.	**214**	> 75	**94 (44%)**	[90-110]	**214 (100%)**	**94 (44%)**
L. Spine LAT	**204**	> 85	**67 (33%)**	[90-110]	**204 (100%)**	**67 (33%)**

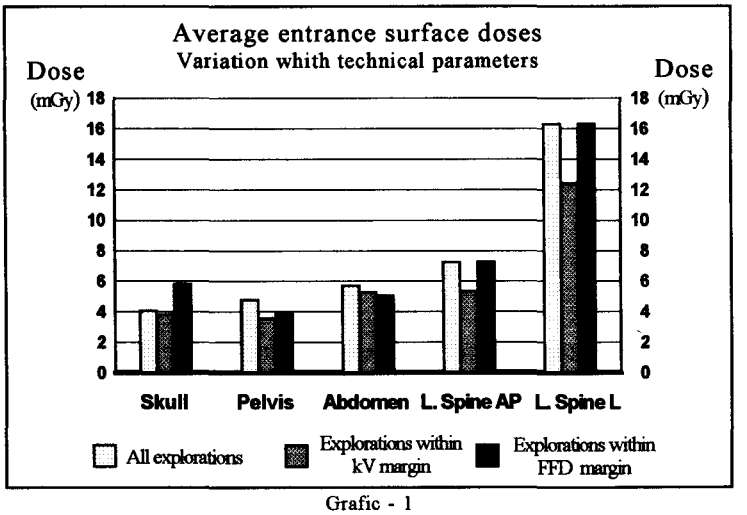
Table - 1

We can calculate the average dose value considering:
1° Those explorations within the kV margin
2° The same ones within the FFD margin
3° The same ones within both margins

The large number of the thorax explorations, and the diferent conditions of them in the diferent rooms, involves the exigency of other analysis.

For one room, we can consider diferent examinations, in some cases made by the same operator, with the same conditions except kV and FFD ones, studying their influence in the obtained doses.

3.- Results and conclusions



Grafic - 1

In the graphic we can see the large influence of the KV selection in all explorations.

This is more important in the lumbar spine and pelvis ones, reaching around 25% of dose in all the cases.

In the skull and abdomen ones, the dose reduction is 7% and 11.5% respectively

About FFD selection, only pelvis and abdomen explorations are significative. In both cases , the obtained value decreses adjusting the distance to the established range.

In the other explorations nothing can be said. In the case of the skull, because of the small number of data, and in the lumbar spine ones because all explorations are within the tolerance.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE DOSES TO PATIENTS IN INTERVENTIONAL RADIOLOGY

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MAJOR SCIENTIFIC TOPIC NUMBER 6 (see page 7)

ABSTRACT (See instructions overleaf)

Interventional and therapeutic procedures are becoming an important part of today's radiological practice. Although these procedures do not contribute significantly to the total number of X-ray examinations performed annually in Poland (below 1% of 21.4 million)they should receive particular attention as they can result in high doses as well to patient as to staff. One of the most frequently carried out interventional radiology procedure in Poland are angiographies (with possibility of additional vessel therapy). This paper presents results of patient dose evaluation for angiography of heart (coronarography), brain angiography and renal arteries catheterization. The importance of technical features of X-ray equipment is underlined. The study was performed in three hospitals, using devices made by Siemens: two of these were Polydoros (with pulsed generators), and the third -the oldest one- was Angiotron CMP (with constant potential, conventional generator). Entrance surface doses to patients were measured using thermoluminescence dosemeters, which were placed directly on the patient skin. Absorbed doses to patients were computed using Monte Carlo simulation programme (own author's code). The results allow to see an influence of generator type and fluoroscopy dose-rate on the value of effective dose to patients.

DOSES RECEIVED DURING INTERVENTIONAL PROCEDURES

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Introduction

In the United Kingdom, radiation doses from medical sources make up 97% of the total collective effective dose to the population from man-made sources [1], due mainly to the large number of x-rays performed. The contribution of interventional radiology procedures to the collective effective dose in the U.K. is unknown as these procedures are not regularly monitored. The contribution of interventional to the population dose from medical exposures could be as high as 40% in some countries. Thus there is some uncertainty in the collective effective dose from medical sources in the U.K. Furthermore, there is no mention of interventional radiology by the National Protocol. Radiation dose and risks from these procedures are therefore of interest.

It is surprising, therefore that the effective dose to patients from many interventional radiology procedures has not been assessed in large scale regional studies. Though the frequency of the procedures is low, the dose for each examination can be high. Patient exposures are high because the screening times are often long and a large number of radiographic exposures are taken. It is therefore necessary to monitor the dose patients receive during these procedures.

Interventional studies performed on twenty-two different fluoroscopy sets were monitored as part of a Regional patient dosimetry programme. The data have been collected using a computer to read and reset the dose-area product meter and also to collect patient and examination details. Data is loaded onto the regional database at quarterly intervals. All the examinations performed in the room are monitored including many interventional procedures.

Method

Diamentors, (PTW, Freiberg), were used to measure the dose-area product. A Diamentor consists of a large area ionisation chamber and control box. Data was collected for a range of examinations including angioplasty, biliary drainage, embolisation and nephrostomy. The examinations included in this study were performed on twenty-two different sets.

The dosimetry method was based on recommendations made in the National Patient Dosimetry Protocol [2]. The dose-area product is particularly useful for assessing and comparing the radiation dose from screening procedures, where the dose-area product provides a more useful indication of overall patient exposure than measurements of surface dose to particular location. Calibration of the instruments was carried out insitu, using a method traceable to a National Standard [3]. The collection of data in this dose survey was automated by use of an IBM compatible laptop computer to read and reset the Diamentor remotely and also to record the patients and the examination details as follows;

Patients: Name, Age, Sex, Height, Weight

Examination:kV, Screening Time, Number of Radiographs, Number of Spot Exposures, Radiologist

This represents a convenient and practical method of collecting data for a large scale survey of this kind.

The results from each department were copied onto floppy disc and sent to the regional centre at quarterly intervals. At the data collection centre the results were loaded onto the database. In order to make meaningful comparisons, the dose area product values were size corrected, as described elsewhere [7] , to the value the patient would have received had they been reference man size. This method uses the concept of equivalent diameter [8]. The patient is approximated by a cylinder of water having the same height and weight as the body. This reduces the variability due to patient size but does not limit the patient size sample. This correction was applied only to examinations involving the main trunk of the body.

Uncertainties

The uncertainty in the dose-area product reading as quoted by the manufactures is ±3% [8]. The calibration factor, which converts the dose-area product meter reading to mGycm^2 is slightly dependent on the tube potential but a single calibration factor is applied to all the data resulting in an overall uncertainty of ±10% in the result.

Results

The results of the dose survey are summarised in table I, giving the mean and median dose-area product values and size corrected dose-area product for each examination. The number of patients, mean screening time, mean energy imparted number of radiographs and number of spot exposures are also given for each examination. The results presented include over seven hundred interventional procedures. Figure 1 illustrates the variation in dose during these procedures, with the dose received during barium studies also shown.

Table I Summary of Results for Interventional Procedures

Examination	Number of Patients	Dose-Area Product Gycm^2		Size Corrected Dose-Area Product Gycm^2		Mean Screening Time (secs)	Mean Number Radiographs	Mean Number Spot Exposures	Mean Energy Imparted mJ
		Mean	Median	Mean	Median				
Angioplasty	337	12.9	7.8	14.3	7.4	446.8	0.6	30.1	126.0
Oesophageal Dilatation	25	12.2	7.5	16.0	8.0	278.4	0.3	2.0	511.1
PTC	83	36.0	25.6	34.3	28.6	874.9	0.7	6.1	352.4
Fluoro Guided Biopsy	65	5.5	1.5	5.4	1.7	123.8	0.3	1.2	54.3
Nephrostomy Drainage	51	16.5	11.2	16.5	13.2	568.3	0.2	4.1	160.5
Nephrostomy	20	12.3	10.3	13.2	8.7	566.3	0.4	2.6	119.9
Embolisation	92	115.1	16.4	110.1	85.0	1404.5	0.0	200.0	1202.2
Biliary Intervention	55	42.1	25.9	40.2	29.8	662.6	0.4	8.3	376.4
Biliary Drainage	42	33.4	35.2	37.0	32.3	1056.8	0.4	5.8	384.0
Dilatation	22	8.4	1.5	10.7	2.5	227.8	0.1	2.4	106.2
Stent	19	40.9	23.9	36.5	25.3	1049.3	0.0	17.6	378.5
Coronary Angiography	1738	56.8	44.5	47.7	37.0	339.7	0.2	580.0	449.1
Coronary Angioplasty	182	77.9	59.8	72.2	66.9	702.8	0.1	395.0	674.1
Radio Frequency Ablation	61	106.3	74.3	91.1	67.0	1718.3	0	75.6	826.4
Mitral Valvuoplsty	31	151.9	116.0	161.9	109.8	2058.6	0	331.9	1348.3

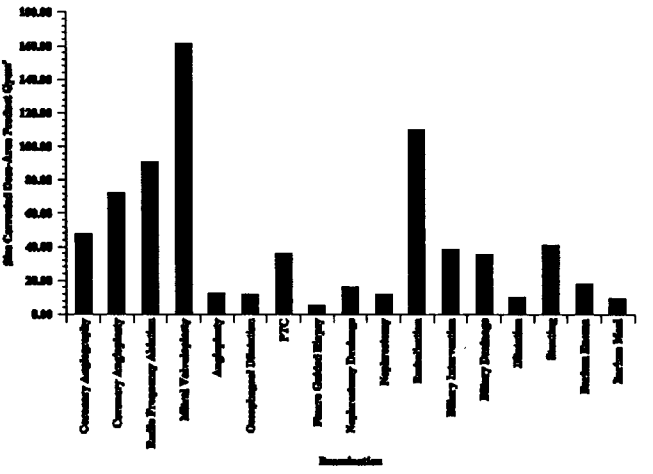

Figure 1 Mean Dose-Area Products for Interventional Procedures

Conclusion

Results are presented here from 2789 patients undergoing a range of interventional procedures. Comparing the results here with the results in [9], 47.7, 72.2,37 161.9 are the results obtained in this study compared with 66.5, 87.5, 68.9, 96.4 from [9] for coronary angiography, coronary angioplasty, biliary drainage and mitral valvuloplasty respectively. Though these examinations are infrequent, contribute significantly to the total dose from medical exposures. These results may be used to compare departments and hence optimise the dose during these procedures.

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Estimation of patient dose in mammographic screening examinations

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INTRODUCTION

Mammography is one of the most effective examinations for detecting breast carcinoma. Although the dose is usually much higher than that in other types of X-ray examination, that is accepted by the patient because for fears of suffering cancer. Benefit of relatively high doses derived from mammographic examinations is considered to well exceed the risk of cancer induction by radiation exposure.

The purpose of this study is to investigate patient dose of mammography in Japan by questionnaire sent to 531 institutions selected from whole Japan and direct measurements carried out in 28 hospitals in Aichi Prefecture. The user's guide in mammography published by NCRP(1) and Quality Assurance Program of American College of Radiology(2) were used to assess the exposure and image quality of mammogram.

METHODS

1. Dose Evaluation

At 28 hospitals in Aichi Prefecture selected randomly among those where mammographies were conducted, exposure conditions were surveyed and exposure in air of entrance surface at each mammographic equipment was measured using Victoreen Radocon 500 and ionization chamber Radocon type 30-330, 0.2cc (calibrated by using a national standard dosimeter). The pressed breast thickness of 4 cm was assumed. Average glandular dose to the whole breast per craniocaudal projection, Dg, (Dg=DgN x Xa, where DgN is f-factor (exposure-to-dose conversion factor), the average glandular dose per unit incident exposure and Xa is exposure in air) for 4cm thick compressed breast organ, 50% adipose + 50% glandular tissue breast, is estimated for each mammography unit. Beam quality was evaluated from measurements of first half-value layer (HVL) at each kV employed for mammography. Average glandular dose was computed from the radiographic techniques required to produce 1.3 Optical Density (OD) images.

2. Image Quality Evaluation

Image quality was evaluated using ACR Phantom (RMI 156) at the 28 hospitals where the exposure dose was measured. The film, intensifying screen and automatic developing machine used were the same as those used regularly at these institutions. The photographic density of the films was measured with a fuji 301RS densitometer. The film density was adjusted to 1.30 OD at the center of the image receptor. The developed films were measured, excluding artifacts (stains and flaws), for discrimination capability (complete discrimination was set as 1) and exceeded the standard by more than 10 points.

3. Estimation of Exposure Dose Based on Japan Survey

A questionnaire was sent to 1000 institutions throughout Japan where mammographic screening

examinations are produced. Based on the response, 531 institutions were selected as possible sites to estimate the average glandular dose per unit incident exposure dose. The dose estimation was made based on actual measurement data.

RESULTS

1. Fourteen of the 28 hospitals were using grids. The average glandular dose was 1.298 mGy and cases without grids were about one-half of this dose.

At nine hospitals (32.1%) image quality exceeded the standard (by more than 10 points) by using the ACR Phantom. Among these, eight hospitals used grids. Hospitals where the tube voltage exceeded 30 kV did not meet the standard. The average glandular dose for equipment exceeding the standard was 1.409 mGy; the mean for equipment not meeting the standard was one-half this figure (Table 1.).

Table 1. Average glandular dose(mGy) for QA test of ACR phantom.

QA Score	With Grid	Without Grid	Mean
≧10	1.388	1.582	1.409
<10	1.177	0.587	0.784

2. Estimation of Average Glandular Dose Based on Japan Survey

Based on the response to a questionnaire sent throughout Japan, the average glandular dose per one mammography was estimated at 1.795 ± 2.613 mGy. The average was less than 1 mGy at 240 institutions (45.2%) was less than 2 mGy at 305 institutions (76.3%). There were 365 institutions (87%) where the average was less than 3 mGy. There were 12 institutions where the average was more than 10 mGy (Fig. 1.).

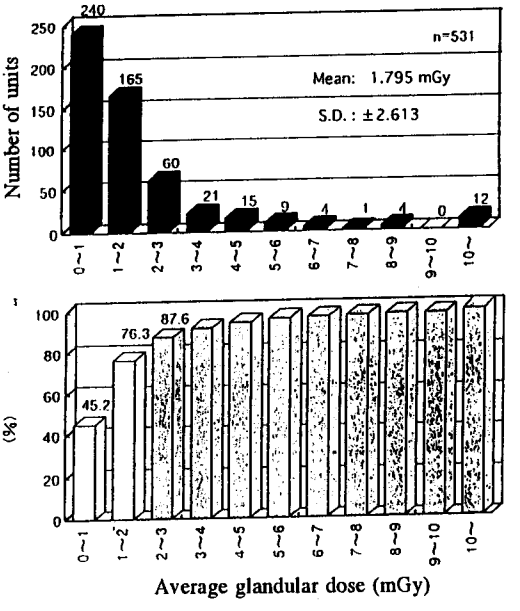

Fig. 1. Distribution of average glandular dose based on the questionnaire.

The institutions using grid for 43% of all institutions; 57% without grid. The average glandular dose was 1.667 ± 1.509 mGy for institutions using grids and 1.907 ± 3.229 mGy for institutions not using grids. For institutions where the average glandular dose was less than 2 mGy, 75% institutions were used with grids and 77% institutions were used without gird.

Among all the various equipment used, 66% combined molybdenum focusing, a molybdenum filter (0.03 mm) and a beryllium window for x-ray radiation opening for exclusive use for mammary x-ray photography. The dose of these institutions was 1.72 mGy. The dose at 54 institutions where x-ray equipment is used for general diagnosis using a tungsten tube was 1.17 mGy (Fig. 2.).

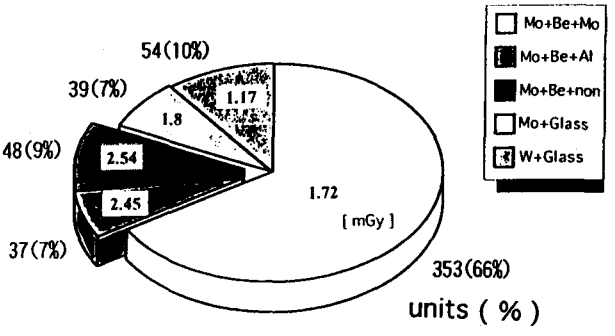

Fig. 2. Comaprison of average glandular dose based on X-ray equipment with focus,window and filter

CONCLUSION

The estimated dose for one mammography based on a nationwide survey in Japan was 1.795 mGy. The actual measurement at 28 hospitals was an average 0.992 mGy. Thus, the estimated dose was higher than that actually measured. Considering mammography accreditation program of ACR, it was suggested that the guidance level of average glandular dose delivered during mammographic screening examinations can be set 2 mGy.

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BONE MARROW AND THYROID ABSORBED DOSES FROM MAMMOGRAPHY

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INTRODUCTION

Breast dose from mammography has been estimated by various investigators, because of the established effectiveness of mammography in early screening for breast cancer and the relatively high sensitivity of the breast to radiation carcinogenesis (1,2). Nevertheless, to our knowledge, there is no available information in the literature about absorbed doses from mammography to organs other than the breast. The absorbed doses to the red bone marrow in the sternum and to the thyroid, due to scattered radiation from mammographic examinations, have been measured using a plexiglas upper-body phantom and thermoluminescent dosemeters. Their dependence on several parameters has also been examined.

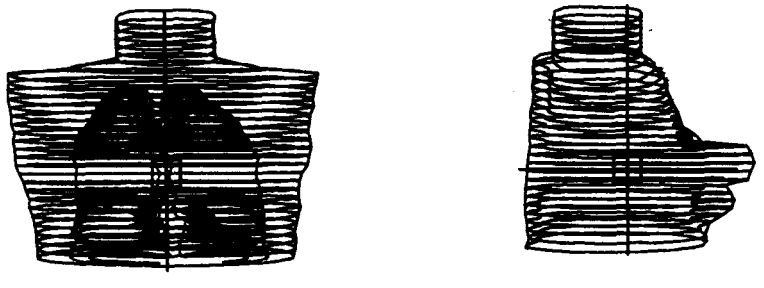
It is necessary to emphasise that this work is still in progress.

MATERIALS AND METHODS

A plexiglas upper-body phantom (figure 1) was used to simulate the female body. The phantom consisted of 30 slices, 1 cm thick, and it simulated the contour of a standard size female upper body. A spongy material inside the phantom simulated the lungs. The compressed breast was simulated by breast-shaped plexiglas slices of various sizes and thicknesses, its midline always aligned with a specific slice. Small holes capable to accommodate thermoluminescent (TL) dosemeters were drilled on each slice of the phantom. The locations of the holes were properly selected to correspond to the following organs inside the body: bone marrow in sternum, thyroid, oesophagus, left and right lung, stomach, colon and liver. In this paper measurements of the doses to the bone marrow and the thyroid are presented.

The dose measurements were performed using two LiF based TLD systems, namely TLD-100 (LiF : Mg, Ti) and GR-200 (LiF : Mg, Cu, P), on 10 sites into the phantom selected to cover the bone marrow in sternum and on 5 sites covering the thyroid.

The TL measurements were performed with a TL analyser Type 711 of the Littlemore Co. The glow-oven was first evacuated at 10^{-2} Torr and N_2 of high purity was left to flow during the readout. The heating rate was 2°C/sec. The light emission was detected by an EMI 9635 QA photomultiplier tube and the glow curves were stored in a computer via a 1024-channel ADC card operating in the MCS mode for further data analysis.



a. b.

Figure 1. The plexiglas upper body female phantom used in the present work.
a. Frontal view
b. Lateral view

Prior to any use the dosemeters were annealed; the TLD-100 at 400°C for 1 hour and at 100°C for 2 hours, whereas the GR-200 at 240°C for 10 min. Before the measurements all the dosemeters were post-irradiation annealed at 80°C for 1 hour. The crystals of both TLD-100 and GR-200 were grouped in terms of sensitivity within 2.5%.

The TL dosemeters were calibrated in beam, against a 10X5-6M ion chamber suitable for measurements in mammography, connected to a Radcal 9015 Monitor.

Most measurements were performed with GR-200 since they are much more sensitive than TLD-100 (at the energies involved by a factor of about 20).

The irradiations were performed on a IMS Giotto Mammography H.F. x-ray unit with a Mo/Mo anode-filter combination and source to image distance 60cm. A thorough quality control of the mammographic unit and the automatic film processor was done prior to the measurements with the phantom. The accuracy and reproducibility of all the exposure parameters (such as kVp, mAs, radiation output) and the automatic exposure control (AEC) system were tested and verified. The performance of the film processor was evaluated by sensitometry.

Measurements were repeated at several irradiation conditions as well as with different plexiglas thicknesses. The high voltage ranged from 26 to 32 kVp and the breast shaped plexiglas thicknesses were 2, 4.5 and 6 cm. These thicknesses correspond to 2.3, 4.9 and 6.4 cm of breast tissue, having a composition of equal parts by weight adipose and glandular tissue (3).

RESULTS AND DISCUSSION

Doses to the bone marrow in sternum and the thyroid at 10 and 5 different sites respectively inside the phantom were measured for three combinations of kVp and plexiglas thickness routinely used in mammography, namely 26 kVp - 2 cm, 28 kVp - 4.5 cm and 30 kVp - 6 cm. Table 1 presents the mean doses to the bone marrow and the thyroid in μGy/mAs and as a percentage of the corresponding entrance dose. These mean values were calculated by merely averaging the corresponding measurements at the different sites.

A general increase of the dose can be observed across the table as kVp, mAs and thickness increase.

Table 1. Mean doses to the bone marrow and the thyroid

	26 kVp, 12 mAs 2 cm plexiglas craniocaudal view		28 kVp, 32 mAs 4.5 cm plexiglas craniocaudal view		30 kVp, 64 mAs 6 cm plexiglas craniocaudal view	
Entrance dose	1197 ± 21 μGy		4550 ± 32 μGy		11934 ± 41 μGy	
	μGy/mAs	% entrance dose	μGy/mAs	% entrance dose	μGy/mAs	% entrance dose
bone marrow	0.40	0.40	0.79	0.56	1.27	0.68
thyroid	0.22	0.22	0.35	0.25	0.51	0.27

The doses measured were generally low, therefore it was decided to restrict further measurements at selected sites only, in order to avoid a prohibitive bulk of measurements. Three sites, namely BMmin, BMmax and BMmid, were chosen for the bone marrow, representing respectively the minimum dose, the maximum dose and the dose to the slice at which the breast shaped plexiglas was centred. One site, TH, was selected to represent the maximum dose to the thyroid.

The dependence of the doses on the mAs and the kVp was then examined separately. The first series of irradiations were performed at 28 kVp with 4.5 cm plexiglas at 24, 28 and 32 mAs (Table 2). These parameters resulted in mammograms with optical densities between 1.2 and 1.55. It was found that the dose rises linearly with the mAs, as it was expected. Therefore, these measurements provide some good evidence on the accuracy of our results, since the ratio dose/mAs remained constant.

Table 2. Dependence of the dose to the bone marrow and to the thyroid on the mAs

	24 mAs		28 mAs		32 mAs	
Optical density	1.21		1.40		1.55	
	μGy	μGy/mAs	μGy	μGy/mAs	μGy	μGy/mAs
BMmin	15.4 ± 1.6	0.64	18.5 ± 2.0	0.66	20.2 ± 1.8	0.63
BMmax	32.2 ± 2.7	1.34	38.9 ± 3.5	1.39	43.6 ± 3.9	1.36
TH	12.4 ± 0.9	0.52	14.6 ± 1.1	0.52	15.9 ± 1.1	0.50

The dependence of the dose to the bone marrow on the kVp was examined by a second series of irradiations which were performed at 32 mAs, with 4.5 cm plexiglas, at 26, 28, 30 and 32 kVp (Table 3). The dose again rises with kVp, being 2.5 to 3 times higher at 32 kVp compared to 26 kVp.

Table 3. Dependence of the dose to the bone marrow on the kVp

	Absorbed dose (μGy)			
Site	26 kVp	28 kVp	30 kVp	32 kVp
BMmid	23.5 ± 3.1	30.4 ± 5.2	46.6 ± 9.9	54.0 ± 12.3
BMmax	27.5 ± 3.8	38.2 ± 6.8	48.5 ± 10.1	78.5 ± 15.2

The maximum dose to the bone marrow and the thyroid after a complete breast screening examination was finally evaluated. A typical screening examination consists of two craniocaudal and two mediolateral views, one for each breast. In order for our results to be meaningful, the irradiation parameters were properly selected to closely match those of a typical examination for the x-ray system used, i.e. 26 kVp, 12 mAs for a plexiglas thickness of 2 cm, 28 kVp, 32 mAs for a plexiglas thickness of 4.5 cm and 30 kVp, 64 mAs for a plexiglas thickness of 6 cm. These conditions result in mammogram with optical densities of 1.2 to 1.6, which is the range normally required in a typical examination (Table 4).

Table 4. Total examination dose (two craniocaudal and two mediolateral views)

	Absorbed dose (μGy)		
Site	26 kVp, 12 mAs/view 2 cm plexiglas	28 kVp, 32 mAs/view 4.5 cm plexiglas	30 kVp, 64 mAs/view 6 cm plexiglas
BMmin	14.5 ± 2.5	55.2 ± 5.2	129 ± 25
BMmid	20.3 ± 3.2	93.9 ± 7.5	199 ± 54
BMmax	26.4 ± 3.5	155 ± 20	337 ± 95
TH	10.6 ± 2.1	40.3 ± 6.2	100 ± 28

Table 4 shows that the maximum measured doses to the bone marrow and the thyroid ranged from 25 to 340 μGy and from 10 to 100 μGy respectively, for a complete examination (four views), depending on breast thickness.

CONCLUSIONS

The absorbed doses to the red bone marrow in the sternum and to the thyroid due to mammographic examinations were measured and their dependence on several irradiation parameters was examined. It was found that the doses depend strongly on the kVp and rise linearly with the mAs. The measured doses for a complete screening examination were found to be generally low. However the dose to the bone marrow can become significant for large breast thicknesses.

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RADIATION DOSES AND RISKS TO NEONATES UNDERGOING COMMON RADIOGRAPHIC EXAMINATIONS IN THE NEONATAL INTENSIVE CARE UNIT

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INTRODUCTION

Neonates in the Neonatal Intensive Care Unit (NICU) can receive large numbers of radiographs owing to the clinical conditions they may present [1]. More neonatal radiation dosimetry data are required for three fundamental reasons: (1) to aid in the establishment of reference dose levels for interinstitutional comparisons; (2) to improve childhood cancer risk estimates following neonatal exposure; and (3) to indicate appropriate directions for dose reduction.

This paper describes an investigation of two different NICU radiological techniques with significantly different neonate doses. While patient-matched images taken with both techniques were assessed in a blind review, this component of the study is beyond the scope of this paper and is not discussed here.

The development of neonatal reference dose levels also requires establishing a mechanism of relating neonate size and dose. We have further investigated the variation of neonate ESD and EI with the equivalent patient diameter, d, given by [2,3],

$$d = 2\sqrt{\frac{W}{\pi L}} \qquad [1]$$

where d is in cm, W is the neonate weight in g and L is the neonate length in cm. The ESD and EI data for both techniques were fit by exponential functions in d,

$$ESD = ESD_o \exp(m_{ESD}\, d) \qquad [2a]$$
$$EI = EI_o \exp(m_{EI}\, d) \qquad [2b]$$

METHODS AND MATERIALS

The two techniques reviewed in this study were a "conventional" technique, used at this institution for a number of years, with the kVp variable between 50 and 60 kVp for a fixed 0.8 mAs, and a "low-dose" technique with the kVp increased to between 62 and 70 kVp and the mAs reduced to either 0.4 or 0.5 mAs. This latter technique is similar to that recommended by the CEC [4]. Patient dosimetry was evaluated from the technique used and previously measured dosimetric quantities. Patient dose data evaluated were the entrance skin dose (ESD), the dose-area product (DAP), the energy imparted (EI) and the mean dose (D), estimated by the ratio of the EI to the neonate weight.

All images were acquired with a General Electric AMX-4 mobile unit and a 400-speed Kodak T-Mat G/Lanex Regular film-screen combination. While others have judged such a speed class to yield less suitable images than slower speeds [5], we have found it to be clinically adequate.

The normalized in-air collision kerma ($K_{C,AIR}$/mAs) was measured for all kVp values used in the study using a 15cc thin window ionization chamber and electrometer (Keithley 96035 and 35050A), both with NIST-traceable calibrations. The ESD was determined using these measured $K_{C,AIR}$/mAs values and the technique recorded by the technologist. Backscatter and attenuation through the

incubator Perspex were accounted for and the mass energy absorption coefficient ratio averaged over the energy spectrum was evaluated for ICRU muscle and used to convert to dose. The DAP was approximated by the product of the ESD and the cross-sectional area of the imaged anatomy determined retrospectively from the film. The EI was calculated from the DAP and conversion factors evaluated by Chapple et al [6].

RESULTS

A total of 363 radiographs were acquired for 77 neonates. Of these, 262 films were acquired for the "conventional" technique (160 chest, 63 abdomen and 39 chest/abdomen combined); 101 films were obtained for the "low-dose" technique (72 chest, 13 abdomen and 16 chest/abdomen). The mean number of radiographs per neonate was 4.7, with a range of 1 to 41. Table I presents a summary of the dose quantities measured, the percentage reductions achieved by switching techniques and the levels of statistical significance in these reductions. Table II presents the log dose gradients, m_{ESD} and m_{EI}, determined by taking natural logarithms of both sides of Equations (2a) and (2b) and performing linear regression.

The major risk of concern for the irradiated neonate is childhood cancer. By using fetal risk factors to estimate the neonatal risk, the excess risk of cancer mortality during the first 15 years of life determined from the Oxford Survey of Childhood Cancers [7] is 0.0288 mGy^{-1} (95% confidence limits, 0.0171 mGy^{-1}: 0.0436 mGy^{-1}). A "worst case" estimate using the upper 95% confidence limit would put the risk per radiograph in the conventional technique to be about 1 in 5210, 1 in 4250 and 1 in 3370 for chest, abdomen and combined chest/abdominal films, respectively. The corresponding risks for the low dose technique are 1 in 6550, 1 in 5460 and 1 in 3530. Most neonates receive multiple films: switching techniques for our "worst case" baby who received 41 chest films would reduce the excess relative risk of childhood cancer death from 1 in 127 to 1 in 164.

CONCLUSIONS

A switch to a neonate radiological technique using a 400-speed film/screen combination and kVp's in the range of 62 to 70 kVp and 0.4 mAs results in considerably reduced doses. Although not discussed here, an independent and blind comparison of clinical images taken with both techniques by three radiologists showed no statistically significant difference in image quality.

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TABLE I: MEASURED NEONATE DOSES

Quantity	Site		
	Chest	Abdomen	Chest/Abdomen
Mean ESD			
Conv Tech	20.0 ± 3.2 μGy	20.0 ± 4.2 μGy	19.0 ± 2.7 μGy
Low Dose Tech	16.4 ± 3.6 μGy	14.5 ± 1.5 μGy	17.6 ± 4.4 μGy
% Change in Means	-18.0%	-27.5%	-7.4%
Significance	$p < 0.0005$	$p < 0.0005$	$p < 0.085$
Mean EI			
Conv Tech	7.9 ± 3.2 μGy	9.2 ± 4.9 μGy	11.5 ± 5.7 μGy
Low Dose Tech	7.1 ± 3.5 μGy	7.1 ± 3.9 μGy	11.9 ± 9.3 μGy
% Change in means	-10.1%	-22.8%	+3.5%
Significance	$p < 0.017$	$p < 0.013$	*
Mean Dose			
Conv Tech	4.4 ± 1.7 μGy	5.4 ± 1.3 μGy	6.8 ± 1.5 μGy
Low Dose Tech	3.5 ± 2.9 μGy	4.2 ± 2.5 μGy	6.5 ± 4.6 μGy
% Change in Means	-20.1%	-22.3%	-3/6%
Significance	$p < 0.0028$	$p < 0.0042$	*

* consistent with no change between the two techniques
- E ~ ros are 1 standard deviation

TABLE II: DOSE GRADIENTS

	Site		
	Chest	Abdomen	Chest/Abdomen
m_{ESD} (cm^{-1})			
Conv Tech	0.068 (0.051 - 0.084)	0.130 (0.105 - 0.155)	0.079 (0.50 - 0.108)
Low dose Tech	0.095 (0.054 - 0.136)	0.044 (-0.027 - 0.114)	0.090 (0.005 - 0.175)
m_{EI} (cm^{-1})			
Conv Tech	0.249 (0.233 - 0.265)	0.349 (0.308 - 0.389)	0.326 (0.255 - 0.398)
Low Dose Tech	0.244 (0.174 - 0.315)	0.178 (0.061 - 0.295)	0.310 (0.068 - 0.552)

Quantities in brackets are 95% confidence limits

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PAPER TITLE RADIATION EXPOSURE AND PROTECTION OF CHILDREN IN X-RAY DIAGNOSTICS OF RESPIRATORY TRACT

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MAJOR SCIENTIFIC TOPIC NUMBER 6.1. (see page 7)

ABSTRACT (See instructions overleaf)

The ever increasing exposure of population to irradiation because of X ray diagnostics of respiratory tract is not always followed by an adequate radiation protection measures. According to the well known radiobiological law, young organisms are more sensitive to radiation than adults. Therefore, in agreement with the recommendations of WHO, and UNSCEAR, special care has to be taken in radiation protection of children.

The basis for radiation protection is the exact knowledge of doses. The patient dose determination in X ray diagnostics is not a simple task. The dose depends from the kind of examination, the technical condition of the equipment, the way and method of exposure, the patient itself and from the accuracy and precision of dosimetric system. Even with a well calibrated dosimetric system, good statistics of measurements is needed for patient dose estimation. We measured the dose on 45 patients divided in 5 age groups. The dose was determined on the back, the breast, the thyroid, the armpit, the left eye and the gonade region. Similar measurements were also carried out in sinus examination of a group of children.

On the basis of the dose measurement results the risk of some cancerous and genetic hazards was estimated.

AUTOMATION OF TLD DOSIMETRY FOR SIMPLE RADIOGRAPHIC PROCEDURES USING THE HARSHAW 6600 MONITORING SYSTEM

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Introduction

Simple radiographic procedures are the most frequently performed x-ray examinations and as a result patient doses should be monitored. In view of the number of examinations performed, it is vital that automated methods of performing direct patient dose measurements are developed. The Harshaw 6600 extremity monitoring system has the potential for use in performing a large scale patient dose survey during radiographic procedures but has not been previously used for this purpose. The potential advantages of this system are, fast readout rate, reproducible time temperature profile, no mechanical contact with the dosemeter and barcode identification allowing individual elemental correction coefficients to be applied to each chip which improves the precision of the result.
Before this system could be used for this purpose, it was rigorously tested. The reproducibility, batch uniformity, detection limit, energy response, fading were all assessed. The overall uncertainty of the results was assessed and compared with the value recommended by the U.K. National Patient Dosimetry Protocol (1). This system has been used to measure fifteen hundred patient dose measurements to date.

Method

The examinations chosen to monitor were chest PA, abdomen AP, pelvis AP, lumbar spine AP and lumbar spine LAT as these are the most frequently performed radiographic exposures. Each department were sent a set of one hundred TLD chips (extremity monitors, x-rads) to use on the patients and ten x-rads to use as background dosemeters, a set of forms to record both patient and examination details. The radiographer placed an extremity monitor in the centre of the field of view, the exposure was taken and the extremity monitor was removed and attached to the patients form. The aim was to collect twenty patients for each category. Once the extremity monitors had been used they were returned to the regional centre where the results were processed and the patient and examination details along with the dose results were entered onto the regional database. A report was generated giving a summary of the dose information.

Results

Table I Performance Testing Results

Test	Results
Reproducibility	2.9%
Batch Uniformity	3.1%
Detection Limit	0.1mGy
Fading (pre-irradiation)	25% over 3 months
Fading (post irradiation)	12% over 3 months
Energy Response	1.6 - 1.3 rel Cs^{137} energy range 45-104keV
Overall Uncertainty at 0.1mGy	20.7%

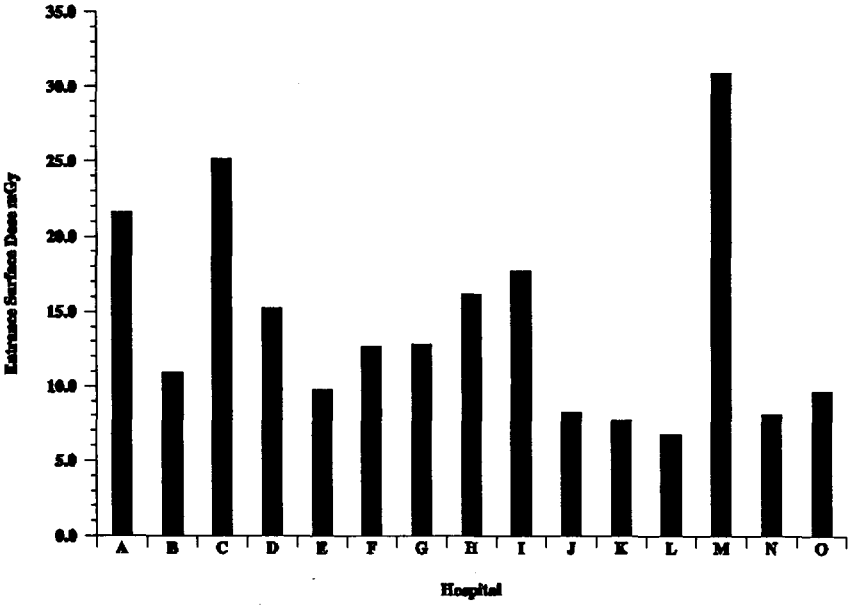

Figure 1 Entrance Surface Dose Results for Lumbar Spine Lateral Examination Throughout the North of England

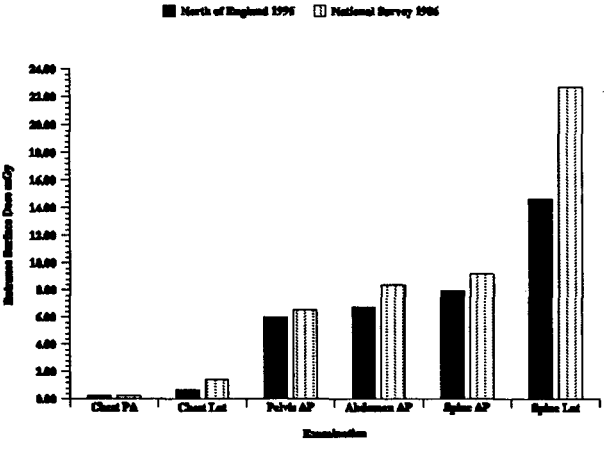

Figure II Mean Doses From the North of England Compared with the National Survey

Table II - Summary of Mean Dose Results and Parameters

Examination	Number of Patients	Entrance Surface Dose mGy			kV	mAs
		Mean	Maximum	Minimum	Mean	Mean
Chest PA	299	0.25	3.63	0.00	71	5.8
Chest Lat	32	0.74	3.59	0.20	80	11.9
Abdomen AP	178	6.77	70.58	0.38	71	35.1
Pelvis AP	291	6.08	65.65	0.23	70	31.6
Lumbar Spine AP	298	8.01	77.43	1.03	74	37.2
Lumbar Spine Lat	291	14.67	92.13	0.48	85	47.5

Figure I illustrates the range in dose values for lumbar spine lateral examination throughout the North of England. Figure II shows the variation in dose between the different examinations and compares results from the North of England with the National Survey [2]. Table I gives a summary of the performance results of the Harshaw extremity monitoring system. Table II lists the mean, maximum and minimum dose results for each examination along with the mean kV and mAs used.
The lumbar spine Lat and abdomen AP results are comparable to the results in [3] of 14 and 5.1 mGy respectively. The chest PA result presented here is higher than the result in [4] of 0.15mGy. There was a broad spread of doses between hospitals for example there was a factor of 4.5 between the minimum mean dose and the maximum mean dose for the lumbar spine lateral examinations. The mean weight of the patients included in this survey was 70.0kg with a standard deviation of 14.8kg.

Conclusion

The Harshaw 6600 personnel monitoring system can be used to monitor patient doses during simple radiographic procedures. The overall uncertainty is within the limits recommended by the National Patient Dosimetry Protocol. One thousand five hundred patient dose results have been presented from seventeen different tubes throughout the North of England. Mean entrance surface dose values are presented for the examinations of chest PA, chest LAT, abdomen AP, pelvis AP, lumbar spine AP and lumbar spine Lat. The results are lower than the results obtained in the last National Survey but are comparable to results from recent surveys.

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DIAGNOSTIC RADIOLOGY DOSIMETRY USING IONIZATION CHAMBERS

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INTRODUCTION

Due to the number of diagnostic X-rays equipments in operation in Brazil and to the fact that medical exposure to radiation is by far the major source of exposure to ionizing radiation of the population, the development of a control method to these equipments is being very important, including dose reduction techniques. It is clear that a lot of studies is being carried out in this field (1,2), but until now, in Brazil and Latin America, there is no operational system or method to calibrate dosemeters used in diagnostic radiology measurements. Since 1980 instruments are being calibrated at the Calibration Laboratory of IPEN at radiotherapy and radiation protection levels. Considering the indicated necessities, studies have been undertaken in order to improve the calibration service with tests at the diagnostic radiology level. The objectives of this work are the determination of diagnostic radiology qualities for instruments calibration using a therapy system and to test some instruments used in diagnostic radiology measurements.

MATERIALS AND METHODS

The X-rays generating system consists of a Rigaku Denki generator, model Geigerflex, coupled to a Philips tube model PW/2184/00 (tungsten target and beryllium window). This system was used to establish the diagnostic radiology qualities according to the German norm DIN 6872, part 1 (3), in the range from 30 to 50 kV. The main characteristics of this radiation system are shown in Table I.

Table I. Characteristics of the Rigaku Denki X-rays generating system.
Focus-chamber distance : 100 cm.

Qualities	Tube Voltage kV	Additional Filtration mm Al	Half Value Layer mm Al	Effective Energy keV	Exposure Rate x 10^{-4} C/(kg.min)
DN1	30	2	0.947	19.0	4.29
DN2	40	4	1.84	28.2	3.71
DN3	50	10	3.61	38.9	1.21

The exposure rates were measured with the secondary standard parallel plate ionization chamber (0.03 cm^3) that was calibrated at the National Physical Laboratory (NPL), England. This chamber was used considering its low energy dependence in this range. The maximum tube voltages (kVp) were determined by spectrometry using an Intertechnique spectrometer, with a HPGe Σurisys Mesures detector. Three ionization chambers made at IPEN(4) and four portable monitors usually used in Brazil for diagnostic radiology measurements were used to compare the results obtained with the secondary standard ionization chamber. Their main characteristics are shown in Table II.

Table II. Main characteristics of the ionization chambers tested at diagnostic radiology level.

Instrument	Type	Window Material	Volume cm^3
(A) IPEN-01-Graphite	Clinical dosemeter	Aluminized Mylar	0.6
(B) IPEN-02-Aluminium	Clinical dosemeter	Aluminized Mylar	0.6
(C) IPEN-04- Graphite	Clinical dosemeter	Aluminized Mylar	3.4
(D) VICTOREEN 660-3	Portable Monitor	Equivalent tissue plastic	4
(E) BABYLINE 81-INT	Portable Monitor	Equivalent tissue plastic	515
(F) BABYLINE 81-RATE	Portable Monitor	Equivalent tissue plastic	515
(G) RADCAL 10X5-180	Portable Monitor	Policarbonate	180
(H) RADCAL 10X5-1800	Portable Monitor	Policarbonate	1800

RESULTS

The maximum tube voltages (kVp) obtained using the HPGe spectrometer are shown in the Fig. 1. The obtained values were approximately 31.5, 41.4 and 51.8 kV respectively for the qualities called DN1, DN2 and DN3 by the German norm DIN 6872. The Fig. 2 shows the spectra measured at the same qualities.

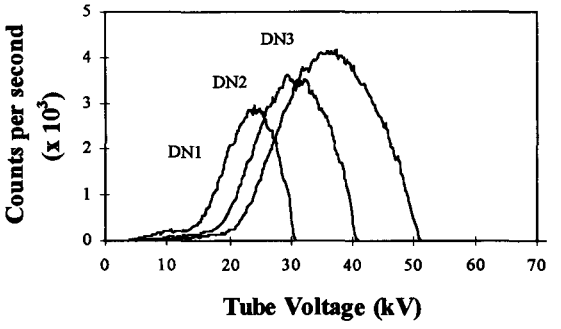

Figura 1. The maximum tube voltage (kVp) determination to DN1, DN2 and DN3 qualities.

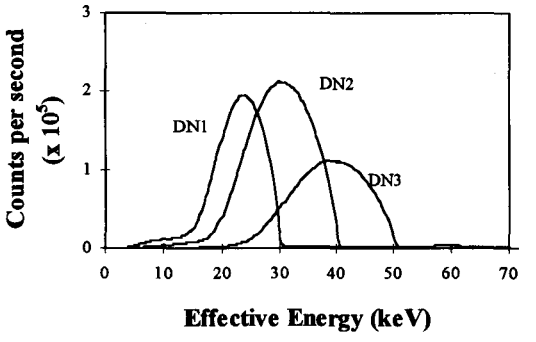

Figura 2. Spectra of the qualities DN1, DN2 and DN3.

The Fig. 3 and 4 show the energy dependence of the tested instruments.

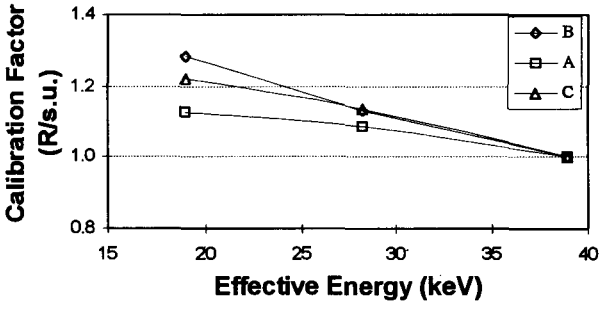

Figura 3. Energy Dependence of the IPEN parallel plate ionization chambers. All values were normalized to 50 kV to make easy the comparison.

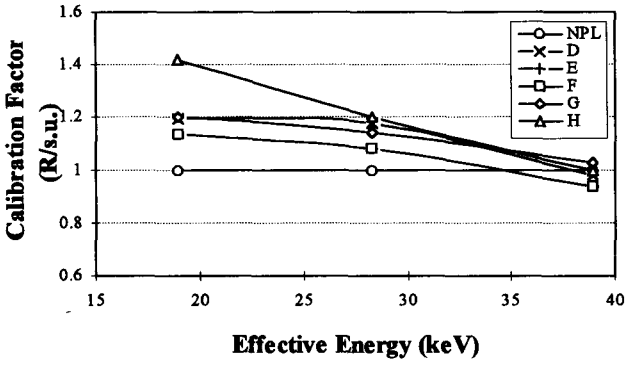

Figura 4. Energy dependence of the portable monitors. All values were normalized to measurements made with the NPL secondary standard ionization chamber measurements.

The Fig. 3 shows that among the homemade chambers the graphite electrode chamber, with a volume of 0.6 cm^3, presents the best energy dependence (13%), which could be compared with that of the secondary standard ionization chamber (8%). In the case of the portable monitors (Fig. 4) the monitors D, E, F and G show less than 20% of energy dependence related to the NPL secondary standard ionization chamber and can be recommended to be used in this range.

CONCLUSIONS

The preliminary results show the importance of the instruments tests at diagnostic radiology qualities, in the case of ionization chambers and specially for some portable monitors. This work is being extended extended for other types of instruments normally used in this kind of measurements; the establishment of the radiation qualities from 60 to 120 kV must be performed to complete the range used in diagnostic radiology, with another adequate X-rays system.

ACKNOWLEDGEMENTS

The authors acknowledge the International Atomic Energy Agency (IAEA) for the donation of the spectrometer, Dr. G. Drexler and Dr. W. Panzer for the related visits and collaboration.

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PAPER TITLE DETERMINATION OF X-RAY EFFECTIVE ENERGY USING CaF_2:Mn AND LiF:Mg,Ti THERMOLUMINESCENT DOSIMETERS

AUTHOR(S) NAME(S) SAVETA MILJANIĆ and BRANKO VEKIĆ

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ABSTRACT (See instructions overleaf)

The possibilities of determination of X-ray effective energy using the energy dependence of the response ratio of two thermoluminescent dosimeters are presented. CaF_2:Mn dosimeters with a large and LiF:Mg,Ti with much lower energy dependence, were used. The calibration factors were obtained for the energy of ^{137}Cs gamma rays and dosimeters energy dependences measured with X-ray beams with the known effective energies were normalized to ^{137}Cs gamma radiation. The results obtained from the energy calibration curves were compared with the calculations using the ratios of the mass-energy absorption coefficients and pertain uncertainties were discussed. The method was applied for effective energy determinations of diagnostic X-ray machines.

NIOBIUM FILTER EFFECT ON PATIENT EXPOSURE AND IMAGE QUALITY

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INTRODUCTION

The estimated annual dose to the population due to radiological examinations has reached by now the same level of the natural background. In order to reduce the probability of stochastic effects, it's necessary to minimise the medical exposure doses (1). In addition to the traditional techniques employed for dose reduction (intensifying screens, post-elaboration programs, beam filtration...), a niobium filter has been recently introduced with the property of having an X ray absorption peak between 19 and 30 keV (2), thus eliminating the portion of the spectrum that doesn't contribute to the radiological image but increases the patient dose. In this report, some of the image quality characteristics are analysed in order to compare the images obtained with and without filter in usual traditional radiology.

MATERIALS AND METHODS

A NIOBI - X filter, consisting of a 50 μm thick layer of niobium, was used to perform the measurements and obtain the film characteristic curves. The film employed was the Fuji HR-A coupled with a Fuji EC-A screen. The exposures were performed at 60, 80 and 100 kV using a 15 step aluminium wedge to achieve the curves. These curves report the optical densities (O.D.) corresponding to steps of increasing thickness. For each considered voltage, the following curves were obtained:

- the 1st in standard conditions of kV and mAs;
- the 2nd in the same conditions of the 1st adding the Nb filter;
- the 3rd was obtained with the Nb filter in order to achieve almost the same optical densities of the 1st varying the mAs;
- the 4th was obtained with the Nb filter in order to achieve almost the same optical densities of the 1st but increasing the kV and reducing the mAs.

For the dose measurements, the Victoreen NERO was used; this equipment also allowed to check the output parameters of the X ray tube. A further aspect analysed was the differences in spatial resolution with and without filter using the Victoreen Focal Spot Test Tool testing seven different exposure conditions (3).

RESULTS

In Fig. 1, 2 and 3, the experimental data of the characteristic curves are represented. The contrast, expressed as the gradient of the linear part of the fitted experimental data in the curve between the optical densities of 0.25 and 2 over the fog, was calculated. In table 1, the results of each exposure are displayed, including dose measurements (for practical reasons these measurements were performed at different focus - film distances). Being the dose proportional to the mAs, the doses normalised to the mAs are also reported with the respective percentage of reduction. In general, the simple addition of the Nb filter doesn't affect the film contrast but reduces its speed, expressed as the reciprocal of the exposure needed to obtain an O.D. of 1. Varying the mAs and/or the kV suitably, it's possible to obtain characteristic curves that are similar to those obtained without adding the filter. As to the 60 kV curves, the filter effect results in diminuishing the film speed more effectively with respect to the other energies. This can be explained with the absorption peak of niobium that is closer to the average energy of the X ray beam. The reported data shows that the addition of the filter with a suitable choice of exposure parameters results in a remarkable dose reduction. From the data in table 2, we can argue that the filter, used with increased kV and reduced mAs, usually doesn't affect and even improves spatial resolution.

CONCLUSIONS

Finally, we retain to argue that the employ of Nb filter reduces the patient doses without affecting the image quality, allowing a correct medical diagnosis. In the Radiology Department of the Istituti Clinici di Perfezionamento, Milan, the Nb filter is currently used on several radiological devices.

Fig. 1: Characteristic Curves (60 kV)

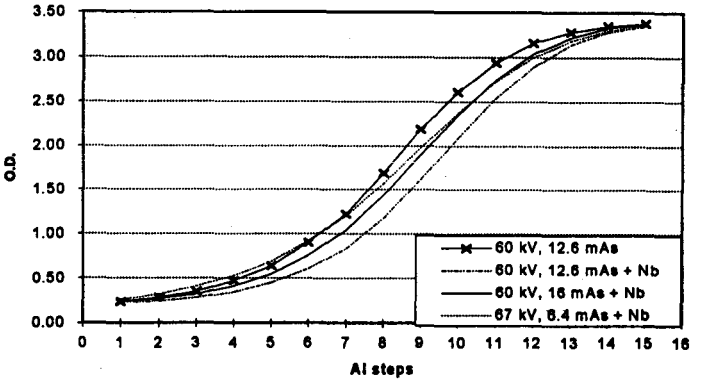


Fig. 2: Characteristic Curves (80 kV)

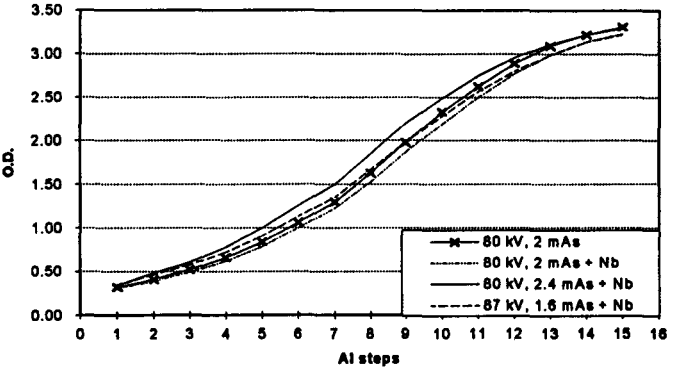
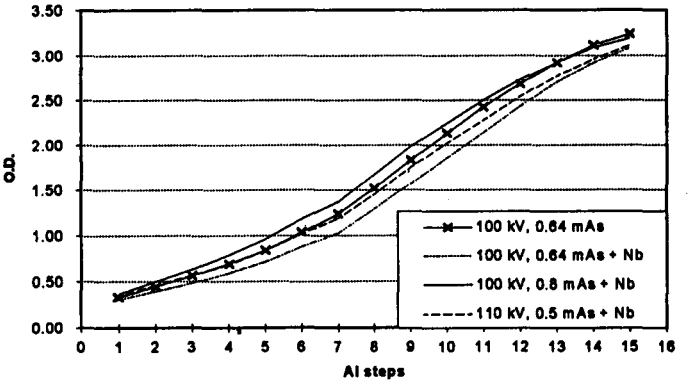


Fig. 3: Characteristic Curves (100 kV)



exposure parameters			characteristic curves parameters		measured dose		dose per mAs unit	
kV	*mAs*	*Nb filter*	*gradient*	*correlation coeff.*	*dose (µGy)*	*% red.*	*µGy/mAs*	*% red.*
60	12.6	no	0.342	0.9846	327	*	25.9	*
60	12.6	yes	0.347	0.9845	154	52.8	12.2	52.8
60	16	yes	0.332	0.9852	196	40.1	12.2	52.8
67	6.4	yes	0.286	0.9849	106	67.4	16.6	35.9
80	2	no	0.243	0.9858	94	*	47.2	*
80	2	yes	0.241	0.9880	51	45.6	25.7	45.6
80	2.4	yes	0.236	0.9876	66	30.6	27.3	42.2
87	1.6	yes	0.227	0.9881	55	42.1	34.2	27.6
100	0.64	no	0.219	0.9869	99	*	154.2	*
100	0.64	yes	0.203	0.9817	58	41.0	90.9	41.0
100	0.8	yes	0.211	0.9896	73	26.0	91.2	40.8
110	0.5	yes	0.197	0.9859	64	35.5	127.4	17.4

TABLE 1: Data obtained for each tested exposure.

X ray tube	without Nb		with Nb		resolution variation
	kV	mAs	kV	mAs	
Gem Televix 1600	60	2.00	68	1.85	+1
	70	1.00	80	0.90	+1
	80	0.56	89	0.54	+1
	90	0.26	99	0.24	+1
	100	0.16	108	0.15	+1
	110	0.40	120	0.38	0
	120	0.15	127	0.14	0

group	1	2	3	4	5	6	7	8	9	10	11
lp/mm	0.60	0.70	0.85	1.00	1.15	1.40	1.70	2.00	2.50	2.80	3.35

TABLE 2: Resolution variations as the diffference between the number of solved groups with and without filter as a function of exposure parameters and resolution related to each line-pair/mm per group.

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PERFORMANCE OF MAMMOGRAPHIC UNITS

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ABSTRACT

The performance of about 50 mammographic units used for mammographic screening in Finland was measured according to the Nordic QA recommendations. The measurements focused on radiation output and quality, focal spot size and spatial resolution, performance of the automatic exposure control (AEC) system, the sensitivity of the image receptor, and film processing. The results show substantial deviations from the recommended values in older units, especially in the focal spot size and in the performance of AEC, but most of the modern mammographic units meet the Nordic performance recommendations to a great extent.

INTRODUCTION

The frequency of mammographic examinations increased manyfold in Finland when the nationwide breast cancer screening programme started in 1987. About 150 000 women are screened annually in Finland[1]. In 1994, about 190 mammographic units were used for mammographic examinations, of which 81 units were used for screening.

High requirements are placed on low-contrast and spatial resolution in order to detect small density differences and small microcalcifications in the breast. The whole imaging chain should be included in the performance measurements, because many parameters in the imaging chain affect the breast dose and image quality. Many studies show deficiencies in the performance of mammographic units. In a Swedish study[2], the tube current deviated from the recommended values in 25%, the focal spot size and spatial resolution in about 60% and the base plus fog in film processing in about 40% of the units measured. A study of the Dutch mammographic quality control[3] revealed deviations of over 30% in darkroom light tightness,in AEC exposure settings and phantom density, in AEC object thickness compensation, in viewing box illumination and in breast compression force. The Nordic countries and the European Union have issued recommendations for the performance and quality assurance of mammographic units[4,5]. We have measured the performance of about 50 mammographic units used for mammographic screening in Finland according to the Nordic QA recommendations.

MATERIAL AND METHODS

The measurements focused on radiation output and quality, image quality, the performance of the automatic exposure control (AEC) system, the sensitivity of image receptor, and film processing. The measurements were made according to the Nordic QA recommendations[4]. The x-ray tube voltage was measured with a kV_pmeter calibrated against an x-ray spectrometer. Focal spot size was measured with a star-test plate. Performance of AEC was tested with respect to object thickness compensation. Film processing was tested sensitometrically by measuring base & fog, gradient and relative speed. Film-screen sensitivity was tested by measuring the dose to image receptor for net optical density 1.0. Spatial resolution was measured with a spatial resolution test plate in a CGR test phantom perpendicular and parallel to the tube axis. Breast compression was measured with a bathroom scale. Radiation output, and linearity was measured by using an MDH 1515 radiation monitor provided with a mammographic ionization chamber (Model 10x5-6M) calibrated in the

secondary standard laboratory. The measuring parameters and methods are described in more detail in the Nordic recommendations for quality assurance in mammography[4].

RESULTS

Table 1 shows the mean values and ranges of the various measured parameters and the recommended values. The greatest deviations from the Nordic recommendations were found in the focal spot size and spatial resolution and in the performance of automatic exposure control. AEC performance fails in the sense that the film optical density depends on the object thickness. Figure 1 shows the distrubution of the focal spot length. Figure 2 shows the gradient distribution (OD 0.25..2.0) of sensitometric curves in film processing.

DISCUSSION

The results show major deviations from the Nordic recommendations. The greatest deviations were detected for older mammographic units. The performance of the new units meets to a great extend the Nordic recommendations. With regard to various parameters about 80% of the units meet the linearity recommendation of the radiation output, film processing meets about 80–90%, automatic exposure control about 45%, spatial resolution about 25–75% (two directions) and sensitivity of image receptors about 70% of the corresponding recommendations. Some of these deviations are similar to the findings reported earlier[2,3]. The performance of all units cannot be expected to meet the recommended values completely within a short time. When old units are replaced by new types, the recommended performance will be achieved gradually.

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Table 1. The mean values and ranges of the measured parameters.

Parameter	Mean value (range)	Recom. value[4]
Radiation output:		
- mGy/mAs	11.4 (5.0– 24.8)	
- linearity (%)	0.7 (–11.9 – +11.0)	± 5
Accuracy of tube voltage (kV)	–0.4 (–2.1 – +2.7)	± 1.0
Half value layer (HVL mm Al)	0.34 (0.26 – 0.40)	≥ 0.25
Focal spot size (large focus, mm)	1.1 (0.5 – 1.7) length 0.8 (0.3 – 1.4) width	≤ 0.4 (IEC)
Spatial resolution		
(parallel tube axis, lp/mm)	14.4 (10 – 16.6)	≥ 15
(perpendic. tube axis, lp/mm)	12.7 (9 – 16.6)	
Spatial resol. of image receptor (lp/mm)	20	≥ 20
Automatic exposure control		
- mean (OD) (4.5 cm phantom)	1.35 (0.8 – 2.4)	1.0 – 1.5
- object thickness comp. (OD) (3 – 6 cm)	0.5 (–1.1 – +0.7)	± 0.2
Sensitivity of image receptor (μGy)	73 (28 – 201)	< 70
Film processing		
- Developer time (s)	100 – 260	180 – 240
- Developer temperature (°C)	33 – 37	34 – 37
- Base & Fog (OD)	0.19 (0.16 – 0.25)	≤ 0.20
- Gradient (0.25 – 0.50 OD)	1.8 (1.5 – 2.2)	≥ 1.6
- Gradient (0.5 – 1.5 OD)	3.3 (2.4 – 3.9)	≥ 3.0
- Gradient (0.25 – 2.0 OD)	3.0 (2.2 – 3.4)	≥ 2.8
- Relative speed	2.0 (1.1 – 2.8)	

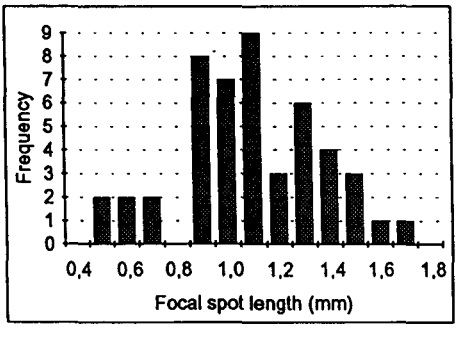

Figure 1. Distribution of focal spot length.

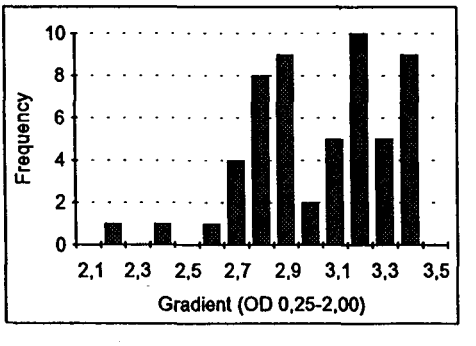

Figure 2. Gradient distribution of sensitometric curves (0.25..2.0 OD) in film processing.

LOW-CONTRAST PERFORMANCE OF CT UNITS

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ABSTRACT

Measurements of patient doses in head CT examinations and measurements of image quality at various surface doses were made for 20 CT units of various types and ages. The mean Multiple Scan Average Dose (MSAD) in head CT examinations was 73 mGy (range 44...121 mGy). Only about 35% of the examinations met the guideline of 50 mGy recommended by IAEA. Wide variations in noise and in low-contrast performance were found in new CT units. The doses needed to detect Ø 2 mm details of 0.6% contrast varied in the new CT units by a factor of about six. The threshold diameter of the low-contrast details varied between 2 and 4 mm and the noise by a factor of about two at a dose of 50 mGy. Spatial resolution varied between 0.8 and 1.0 mm.

INTRODUCTION

The technical performance of CT units has improved in recent years and new innovations have been taken into use, for example the spiral CT. In spite of the technical improvement, patient doses have not decreased as expected. In contrast, patient doses in CT examinations have increased[1]. Patient doses are high in CT examinations[2,3], often considerably higher than in conventional x-ray examinations. Although CT examinations account for only about 2% of all x-ray examinations in Finland[4], they account for about 15% of the collective dose. High doses are often used in order to obtain good image quality, even though lower quality would sometimes be sufficient for reliable diagnosis.

The X-ray spectrum affects both image quality and patient dose. The spectrum depends primarily on the anode material, tube voltage and total filtration. Wide variations appear in the total filtration between various CT units, but only small differences in tube voltages. The basic measures for image quality are spatial and contrast resolutions and image noise. High spatial resolutions are needed to detect small details and high contrast resolutions to detect low-density differences in soft tissue.

MATERIAL AND METHODS

The low-contrast performance of 20 CT units of various types and ages used in Finnish hospitals were measured at various patient doses. Multiple scan average doses (MSAD)[5] were measured on the surface (depth 1 cm) of a Ø 16 cm perspex phantom and on the rotation axes free in air by using an MDH 1015 radiation monitor provided with a CT ionization chamber (Model 10X5 - 10.3 CT) calibrated against the secondary standard. Spatial resolution and low-contrast performance were measured by using an RMI image quality phantom provided with test objects for spatial resolution and low-contrast (0.6%) resolution. Image noise was measured as the standard deviation (SD) of pixel values in a Ø 16 cm water phantom.

RESULTS

Table 1 shows the MSAD doses on the surface of the Ø 16 cm perspex phantom and the radiation output measured on the rotation axes free in air. Only about 35% of the MSAD doses corresponding to the mean mAs value of 440 used in head CT examinations in Finland meet the IAEA recommendations for CT examinations[6] of the head (MSAD of 50 mGy in the middle of Ø 16 cm perspex phantom).

Figure 1 shows the image noise of new CT units with various surface doses on a Ø 16 cm water phantom. Measurements were made by using slice widths of 8 – 10 mm, FOV 180 – 250 mm, matrix size 512 x 512 and standard algorithm. The noise level varies in various new CT units between 2.2 and 4.2 HU with a surface dose of 50 mGy.

Figure 2 shows the low-contrast resolution of new CT units at various surface doses. The doses needed to detect Ø 2 mm details of 0.6% contrast varied in the new CT units by a factor of about six. With a dose of 50 mGy, the low-contrast resolution of new CT units varied by a factor of about two. The differences in low-contrast performance are mainly caused by differences in image noise (See Figures 1 and 2).

Spatial resolution varied between 0.8 and 1.0 mm and no essential differences were detected between various CT units. If and edge algorithm was used, the spatial resolution improved slightly, to 0.5 – 0.6 mm.

DISCUSSION

The results described above show that even new CT units have high doses and wide variations in patient doses and in low-contrast performance. The following conclusions can be drawn from the results:

1 A high percentage of the head CT doses exceeds the IAEA dose recommendation. Many studies reported in literature show that patient doses in CT examinations can be decreased, often sharply.
2 Image noise does not fully describe the low-contrast performance of the CT units although some correlations are discovered between noise and low-contrast performance, as shown in Figures 1 and 2.
3 Optimization of image quality is needed to reduce patient doses in CT examinations. Reasonable image quality with the minimum patient dose can only be obtained with an exact medical indication and with an examination technique suitable for this indication. The optimization procedure is a challenge for the radiologist.

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6 International Atomic Energy Agency. International basic safety standards for protection against ionizing radiation and for safety of radiation sources; IAEA safety series No. 115–I. (Interim Edition) Table III–II: Dose guidance levels for computed tomography for a typical adult patient. IAEA Vienna 1994.

Table 1. MSAD doses on the surface of a Ø 16 cm phantom and on the rotation axes free-in-air.

CT unit	Tube voltage kVp	Filtration mmAl	Isocentre-distance cm	Dose[1] mGy/100 mAs	Air kerma dose on rotation axes mGy/100 mAs
Siemens					
CR (4 units)	125	2.2 mm Al + 0.2 mmCu	70	11.3 ± 1.4	12,1 ± 1,3
DRH (2 units)	125	"	70	11.6 ± 1.2	11,3 ± 3,3
AR.T	130	2.7 mm Al	51	23.5 ± 2.5	34,8
Plus S	120	2.7 mm Al+ 0.2 mmCu	70	10.7 ± 0.2	13,0
GE					
CT 9800	120	2,7 + s[2]	63	16.0 ± 0.3	25,0
CT Pace	120	2.7 + s	52.5	30.1 ± 0.4	46,5
Sytec 3000	120	2.7	52.5	24.4 ± 1.2	41,0
Sytec 3000 Plus	120	2.7	52.5	27.3 ± 1.6	
HiSpeed Advantage	120	2.7	63	14.6 ± 0.2	20,4
Toshiba					
TCT-80	120	3.2	50	23.9 ± 1.5	38,3
300 S	120	5.5	60	20.8 ± 2.6	22,0
600 HQ	120	6.4	50	25.2 ± 2.2	24,4
Xpeed	120	6.4	50	21.3 ± 1.1	21,7
Xpeed II	120	6.4	50	20.4 ± 0.4	
Xvision	120	3.4	50	24.2 ± 3.5	23,3
Philips					
Tomoscan SR4000	120	6.5	61	16.3 ± 1.4	18,0

[1] slice thickness 8 or 10 mm [2] s = shaped filter

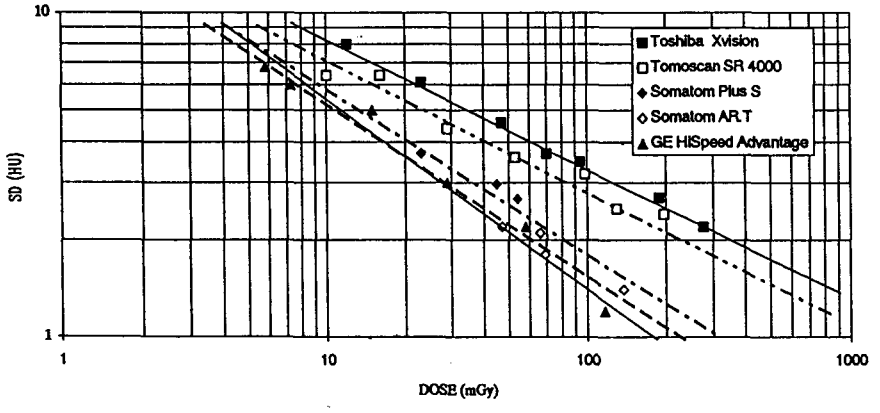

Figure 1. The measured image noise of new CT units with various surface doses using a Ø 16 cm water phantom.

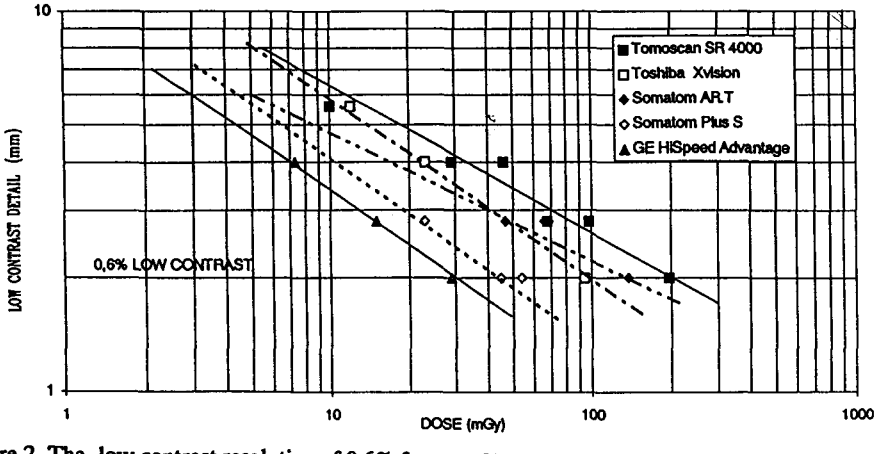

Figure 2. The low contrast resolution of 0.6% for new CT units at various surface doses using a Ø 16 cm perspex phantom.

CT DOSE MEASUREMENT

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Abstract - In this paper the doses for different CT scans are determined. It's found that x-field profiles are wider than declared scan widths, meaning that x field overlap occurs with neighbouring tomograms and that those regions get x-rayed twice. To lessen that unwanted side-efect, it's necessary to enlarge scan width, or to decrease the number of neighbouring scans. We suggest that table shift should equalise actual field width. In addition, the programme for using CT with minimal radiation risk for patients is given.

INTRODUCTION

Computorized tomograph (CT) has become the principal diagnostic machine of x-ray departments and their number represents the measure of the health standard of a country. Whereas in classic x-ray technique relevant details are imersed in scatered radiation background and masked by inadequate developing , on CT doctor gets a tomogramme with details and clarity of an anatomical preparation. The diagnosis is obtained quickly and very reliably, therefore scanners operate full time. However, due to high patient's doses CT is not an ideal diagnostic mean. That fact is not enough known to both patients and doctors. The needed resolution of forms and densities with CT is acchieved , among other programme requirements, by hard x-irradiation. This obviously is not a non-invasive technique and with such a hard spectrum patient's doses are high opposing any lax and inadvertent sending of patients to CT before other "softer" diagnostic techniques are used. In this paper the possibility of reducing the patient's dose is examined.

METHODS

Scanner's x-field is wholly atipical since it has the shape of triangular thin plate rotating in the circle in which there's the pacient, who successively translates orthogonaly to the beam. Dosimetric measurements in such a field demand more atteention in choosing instruments as well as in interpreting results. Because of field rotation one uses cylindrical chambers only, however in the field up to 10mm wide, it's difficult to obtain the usual situation with entire chamber in the field since the majority of chambers are longer than 10mm. Due to partial irradiation of chambers the calibration factor of the chamber changes , which is compensated for by measuring the dose distribution and beam width. In this way the dose in a scan's field is determined. If the entire chamber is caried through scanner's beam by progression of succesive exposition, the mean dose per scan is obtained with the overlapping efect of neighbouring scans included.

RESULTS

From x-ray film it's clearly visible that the field profile in the phantome,(patient), is considerably wider then the declared scan width. The divergence of nominal values of scan width is given in Table1. Because of tube rotation the traces of incoming beam were superimposed in field profile from all directions, as well as the traces of scattered irradiation. Although the mildest conditions on the scanner were used, all field components caused total film blackening, so that scattered-radiation traces could not have been differentiated from central- beam ones. Therefore the dose distribution in the beam profile could not have been densitometrically established. However it's clear that scan field is wider than successive table

translations for next tomogram and for that reason there is a partial field overlap with two neighbouring scans. That means that those patient's regions get x-rayed twice.

Table 1. x-field divergence

Nominal scan width	10 mm	5 mm	2 mm
Scan width at 1cm depth of cylindrical phantom (C_2H_3Cl)	12 mm	7 mm	4 mm
Scan width on the surface of cylindrical water phantom	12.5 mm	7.5 mm	4.5 mm

The dose efect of neighbouring fields overlap established by progressoin of successive tomograms, is given in Table 2.

Table 2. Doses per scan at 1cm depth of the phantom

Nominal scan width and conditions	10 mm 133 kV,350 mAs, 2s	5 mm 133 kV, 350 mAs, 2 s	2 mm 133 kV, 475 mAs, 2.7 s
Dose per scan without field overlap	2.8 cGy	4.4 cGy	5.5 cGy
Dose per scan with field overlap	3.2 cGy	5.9 cGy	10.6 cGy

CONCLUSION

Results show that field overlap considerably increases dose on CT. The only way of reducing x-field overlap efect would be the use of wider scans or decreasing the number of neighbouring tomograms. The question is how much of CT's capacities can be sacrificed for patient's protection?

In praxis several principles and recommendations should be followed in order to solve the diagnosis vs. protection dilemma:

- always use classical methods before CT examination.
- use layers as thick as possible.
- examine aimed layers with of contrast.
- use 10mm layers, which proved optimal for examination.
- work without native examination whenever possible.
- for cranium examination use 10mm layers, with initiating scan line passing through tragus and orbit's roof.

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TRACK STRUCTURE CALCULATION OF THE DOSE ENHANCEMENT IN TISSUE ADJACENT TO IMPLANTS OF HIGH ATOMIC NUMBER.

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Abstract

Present investigations deal with diagnostic x-ray qualities as applied in medical diagnostics. The track structure computer program PARTRAC [1] simulating the coupled photon - electron transport and interactions in complex geometries has been used. Differential cross sections for electrons in gold with primary energies in the range of 100 eV to 100 keV have been derived on the basis of Seltzer model in case of ionizations and excitations with new Hartree-Fock input data. The elastic cross section data were taken from Fink et al. and Riley et al.. These cross sections have been inserted into the existing track structure computer program to provide detailed simulations of electron interactions and to describe above mentioned interface effects. The tracks were simulated in a target volume of gold surrounded by water vapour to represent cell tissue and irradiated by 30 keV and 100 keV photons. The simulated geometry was chosen as in experiments using exoelectron emission. The tracks were calculated for comparison in pure water vapour under the same irradiation conditions.

Introduction

Implants and contrast media in the human body may constitute interfaces between tissue and high-Z materials. Such interfaces lead to dose discontinuities during radiation exposures as known from megavoltage therapy. Significant dose enhancement has meanwhile been found in tissue close to a gold layer by using exoelectron technology with a spatial resolution in the micrometer range as it is shown on the fig. 1 [2].

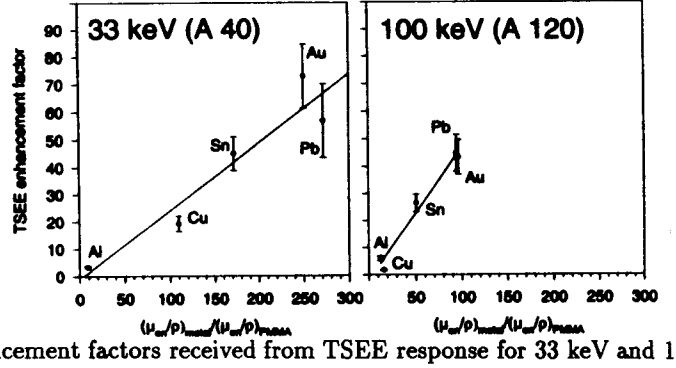

Fig.1 Enhancement factors received from TSEE response for 33 keV and 100 keV x-rays in tissue equivalent material at different metal surfaces under backscatter conditions [2]

Method

The photon-electron transport code **PARTRAC** [1] is used to calculate deposited energy distribution in the vicinity of the tissue - gold interface. The electron track simulation

covers energies from 10 eV to 100 keV. The electron cross sections for water vapour with appropriate density scaling are taken to simulate tissue [1].
New ionization and excitation cross section data for electrons in gold were calculated. Cross section differential in energy were derived for subshells using Seltzer method [3]. New values for the mean kinetic energies of orbital electrons U_j, the expectation values of the electron orbital radius $\langle r_j \rangle$ and the binding energies B_j were obtained from Hartree - Fock calculatins made by P. Indelicato [4]. Results are shown on Fig. 2, 3. Elastic cross section data were taken from Fink et al. [5] and Riley et al. [6].

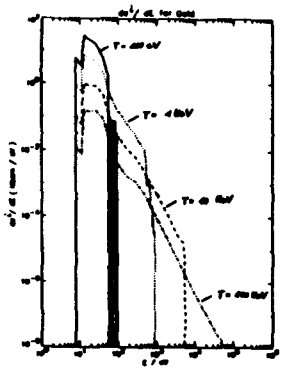

Fig. 2 Differential ionisation cross sections $(d\sigma/dE)^i$ in dependence on the energy transfer E are shown for electron primary energies T=100 eV, 1 keV, 10 keV, 100 keV. Energy edges of the outer shells could be seen (from the left side:6s, 5s,5p, 5d, 4s, 4p, 4d, 4f).

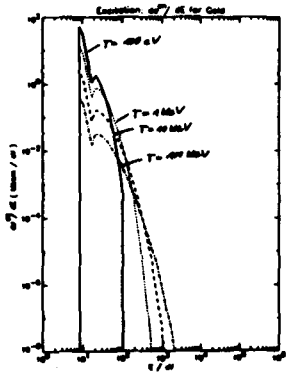

Fig. 3 Excitation differential cross sections $(d\sigma/dE)^{ex}$ in dependence on the energy transfer E are shown for electron primary energies T = 100 eV, 1 keV, 10 keV, 100 keV.

These cross section data have been implemented into the existing track structure computer program.
The experiment to be simulated consists of a PMMA slab ($100x100x5mm^3$), a 100 μm layer of tissue (the region for tracing the secondary electrons) in front of a 100 μm gold layer . The experiment was simulated for 30 keV resp. 100 keV monoenergetic photons.

Results

The results of our calculations are shown in Figures 4 - 5. The energy deposition is found to be considerably higher for the same irradiation conditions in tissue in front of the gold foil than in pure tissue. This can be explained by the large differences in photon interaction cross sections for gold and for tissue and by the different character of interaction. The photoeffect dominates in gold and the Compton effect dominates in tissue causing different structures of secondary electron spectra.
The maximum of the energy deposition occurs at a distance of several micrometers from the gold - tissue interface caused by a larger number high energy electrons emitted from the gold into the tissue. Different to experiments build up maximum of energy deposition is observed for 100 keV photons at a distance of 20 μm from the gold surface, for 30 keV at about 2 μm (Fig. 4 and 5).

Conclusion

The track structure modelling was performed at tissue - gold interfaces for X-ray energies of 30 keV and 100 keV using the **PARTRAC** code under backscatter conditions. This code allows a persecution of electron histories down to 10 eV also for non-equilibrium exchange of secondary electrons across the transition zone of the media interface. Sig-

nificant enhacement of energy deposition was found in tissue layers adjacent to the gold surface as compared with a homogeneous tissue phantom. The enhacement depends on the primary photon energy. Also the range of enhanced energy deposition was found to depend on the primary photon energy that reaches about 17 μm for 30 keV and about 130 μm for 100 keV. A build up maximum of energy deposition was found for 100 keV primary photons at a distance of about 20 μm from the interface, for 30 keV photons at about 2 μm. There is a wide distribution of electron energies up to the primary photon energies originating from gold which is not present for pure tissue conditions. These preliminary results indicate that the PARTRAC code offers a large potential to analyze enhanced energy deposition in tissue close to materials of higher atomic number what is useful for X - ray diagnostics.

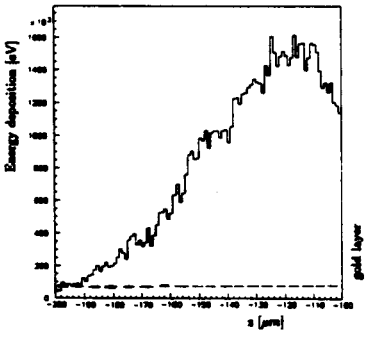

Figure 4 Depth distribution of electron energy deposition in front of the gold foil. The gold - tissue interface is at z = -100 μm. The y-axis are energy depositions in eV normalized to number of photon histories. The broken line shows the level of energy deposition if there is tissue instead of gold. Primary energy of photons: 100 keV.

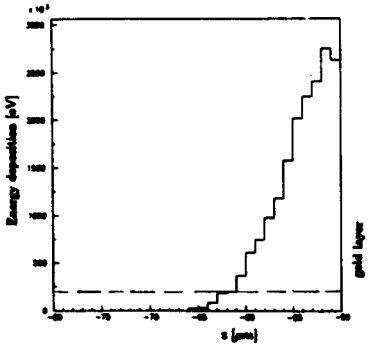

Figure 5 Depth distribution of electron energy deposition in front of the gold foil. The gold - tissue interface is at z = -50 μm. The y-axis are energy depositions in eV normalized to number of photon histories. The broken line shows the level of energy depositions if there is tissue instead of gold. Primary energy of photons: 30 keV.

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EFFECTIVE DOSE AND EFFECTIVE DOSE EQUIVALENT IN NUCLEAR MEDICINE

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INTRODUCTION

For years many physicians and physicists have employed the concept of "total body dose" or "whole body dose," i.e., the total energy deposited in the body divided by the total mass of the body, in evaluating the risks of different nuclear medicine procedures. Although this concept may be considered useful for comparing doses received from different procedures, it does not take into account the typical non-uniformity in dose distribution among the several body organs. The effective dose equivalent (H_E), first described in ICRP Publication 26 (1), has been accepted by some as a better quantity to use in evaluating the total risk of a procedure, but its use has been criticized by others primarily because the tissue weighting factors were intended for use with the population of radiation workers, rather than of nuclear medicine patients (2). Nevertheless, the ICRP has suggested in Publication 52 (3) that the H_E may be used in nuclear medicine, and provided in Publication 53 (4) a compendium of dose estimates, including H_E values, for various nuclear medicine procedures undergone at various ages. The effective dose (E) of ICRP Publication 60 (5) is perhaps more suitable for use in nuclear medicine, with tissue weighting factors based on the entire population.

METHODS

Comparisons of H_E and E have previously been published (6). We have used the program MIRDOSE 3.1 to compute total body dose, H_E and E in the adult male and adult (nonpregnant) female for 62 radiopharmaceutical procedures, involving 19 different radionuclides, based on the best current biokinetic data available. The improvements in this latest version of MIRDOSE include mathematical phantoms for the pregnant female at 3-, 6-, and 9-months gestation, and a new model for bone and marrow dosimetry based on the work of Eckerman (7).

RESULTS

The computed values of total body dose, H_E and E are listed in Table 1, for both the adult male and adult female phantoms in MIRDOSE 3.1. As found by Johansson et al. (6), the average ratio of E to H_E is about 0.8, with a relatively narrow spread from 0.48 to 1.77. The ratio of E to total body dose, however, ranges from 1.1 to almost 100. All ratios greater than 10 occur for the iodines; the values for Tc-99m agents fall between 1.4 and 6.9. In view of the non-uniform distributions of most radiopharmaceuticals, we believe that the total body dose is not a useful quantity for evaluating risks, and should be replaced by the quantity effective dose, or the effective dose equivalent, as an interim method in those countries that have not as yet adopted the ICRP 60 (5) methodology.

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Table 1. Total body dose, effective dose equivalent (H_E) and effective dose (E) (mSv/MBq), and ratios for several radiopharmaceuticals.

	DOSES						RATIOS					
	FEMALES			MALES			FEMALES			MALES		
Pharmaceutical	Total Body	H_E	E	Total Body	H_E	E	H_E/TB	E/TB	E/H_E	H_E/TB	E/TB	E/H_E
Au-198 colloid	4.58E-01	1.72E+00	1.16E+00	3.59E-01	1.38E+00	9.14E-01	3.76	2.53	0.67	3.84	2.55	0.66
C-11 Tryptophane	3.60E-04	5.92E-04	5.03E-04	2.87E-03	5.16E-03	4.32E-03	1.64	1.40	0.85	1.80	1.51	0.84
C-11 Iomazenil	2.79E-03	1.54E-02	1.39E-02	2.20E-03	1.19E-02	1.06E-02	5.52	4.98	0.90	5.41	4.82	0.89
Co-57 B-12, Normal	1.91E+00	3.73E+00	2.90E+00	1.53E+00	2.94E+00	2.25E+00	1.95	1.52	0.78	1.92	1.47	0.77
Co-57 B-12, PA	2.68E-01	6.18E-01	5.99E-01	2.15E-01	5.00E-01	4.90E-01	2.31	2.24	0.97	2.33	2.28	0.98
Co-58 B-12, Normal	3.66E+00	7.08E+00	5.45E+00	2.96E+00	5.70E+00	4.35E+00	1.93	1.49	0.77	1.93	1.47	0.76
Co-58 B-12, PA	6.08E-01	1.61E+00	1.59E+00	4.93E-01	1.30E+00	1.30E+00	2.65	2.62	0.99	2.64	2.64	1.00
Co-60 B-12, Normal	5.61E+01	1.08E+02	8.01E+01	4.56E+01	8.67E+01	6.39E+01	1.93	1.43	0.74	1.90	1.40	0.74
Co-60 B-12, PA	7.49E+00	1.48E+01	1.24E+01	6.08E+00	1.19E+01	1.00E+01	1.98	1.66	0.84	1.96	1.64	0.84
F-18 FDG	1.51E-02	3.81E-02	3.10E-02	1.20E-02	2.98E-02	2.41E-02	2.52	2.05	0.81	2.48	2.01	0.81
F-18 Sodium Fluoride	1.10E-02	3.56E-02	3.10E-02	8.75E-03	2.70E-02	2.31E-02	3.24	2.82	0.87	3.09	2.64	0.86
Ga-67 Citrate	8.23E-02	1.23E-01	1.20E-01	6.62E-02	1.03E-01	1.00E-01	1.49	1.46	0.98	1.56	1.51	0.97
Hg-197 Chlormerodrin	3.84E-02	2.05E-01	1.13E-01	3.00E-02	1.81E-01	9.66E-02	5.34	2.94	0.55	6.03	3.22	0.53
I-123 Hippuran	3.52E-03	3.39E-02	2.90E-02	2.70E-03	2.38E-02	2.01E-02	9.63	8.24	0.86	8.81	7.44	0.84
I-123 IMP	1.43E-02	2.44E-02	2.34E-02	1.15E-02	1.91E-02	1.82E-02	1.71	1.64	0.96	1.66	1.58	0.95
I-123 mIBG	1.14E-02	2.56E-02	2.21E-02	9.12E-03	1.93E-02	1.66E-02	2.25	1.94	0.86	2.12	1.82	0.86
I-123 Sodium Iodide	9.72E-03	1.47E-01	2.43E-01	8.03E-03	1.20E-01	2.00E-01	15.1	25.0	1.65	14.9	24.9	1.67
I-125 HSA	2.61E-01	3.89E-01	2.91E-01	2.07E-01	3.07E-01	2.29E-01	1.49	1.11	0.75	1.48	1.11	0.75
I-125 mIBG	2.93E-02	6.71E-02	4.86E-02	2.28E-02	5.08E-02	3.63E-02	2.29	1.66	0.72	2.23	1.59	0.71
I-125 Sodium Iodide	1.61E-01	7.62E+00	1.35E+01	1.32E-01	6.37E+00	1.13E+01	47.3	83.8	1.77	48.3	85.6	1.77
I-131 Hippuran	9.71E-03	1.35E-01	1.17E-01	7.27E-03	1.00E-01	8.58E-02	13.9	12.0	0.87	13.8	11.8	0.86
I-131 HSA	5.89E-01	1.30E+00	9.35E-01	4.68E-01	1.07E+00	7.43E-01	2.21	1.59	0.72	2.29	1.59	0.69
I-131 MAA	1.80E-01	6.35E-01	6.06E-01	1.41E-01	4.96E-01	4.72E-01	3.53	3.37	0.95	3.52	3.35	0.95
I-131 mIBG	1.03E-01	2.51E-01	1.95E-01	8.12E-02	1.95E-01	1.49E-01	2.44	1.89	0.78	2.40	1.83	0.76
I-131 Sodium Iodide	2.32E-01	1.27E+01	2.24E+01	1.92E-01	1.04E+01	1.84E+01	54.7	96.6	1.76	54.2	95.8	1.77
I-131 Rose Bengal	1.02E-01	1.02E+00	1.33E+00	8.13E-02	9.02E-01	1.21E+00	10.0	13.0	1.30	11.1	14.9	1.34
In-111 DTPA	1.11E-02	5.75E-02	5.02E-02	8.75E-03	4.10E-02	3.56E-02	5.18	4.52	0.87	4.69	4.07	0.87
In-111 Platelets	1.92E-01	6.18E-01	3.95E-01	1.55E-01	5.13E-01	3.26E-01	3.22	2.06	0.64	3.31	2.10	0.64
In-111 RBCs	1.80E-01	2.47E-01	2.24E-01	1.48E-01	2.04E-01	1.85E-01	1.37	1.24	0.91	1.38	1.25	0.91
In-111 WBCs	2.04E-01	7.62E-01	4.88E-01	1.63E-01	6.38E-01	4.09E-01	3.74	2.39	0.64	3.91	2.51	0.64
In-111 Pentetreotide	3.77E-02	1.46E-01	1.03E-01	3.02E-02	1.18E-01	8.14E-02	3.87	2.73	0.71	3.91	2.70	0.69

Table 1 (continued). Total body dose, effective dose equivalent (H_E) and effective dose (E) (mSv/MBq), and ratios for several radiopharmaceuticals.

	DOSES						RATIOS					
	FEMALES			MALES			FEMALES			MALES		
Pharmaceutical	Total Body	H_E	E	Total Body	H_E	E	H_E/TB	E/TB	E/H_E	H_E/TB	E/TB	E/H_E
Kr-81m	5.72E-06	3.35E-05	3.39E-05	4.42E-06	2.65E-05	2.65E-05	5.86	5.93	1.01	6.00	6.00	1.00
N-13 Ammonia	1.99E-03	2.81E-03	2.56E-03	1.58E-03	2.22E-03	2.01E-03	1.41	1.29	0.91	1.41	1.27	0.91
P-32 Sodium Phosphate	1.96E+00	2.40E+00	2.29E+00	1.51E+00	1.93E+00	1.80E+00	1.22	1.17	0.95	1.28	1.19	0.93
Tc-99m Alb. µspheres	5.43E-03	1.78E-02	1.77E-02	4.30E-03	1.45E-02	1.45E-02	3.28	3.26	0.99	3.37	3.37	1.00
Tc-99m DISIDA	4.93E-03	3.00E-02	2.15E-02	3.99E-03	2.51E-02	1.78E-02	6.09	4.36	0.72	6.29	4.46	0.71
Tc-99m DMSA	4.76E-03	1.85E-02	1.07E-02	3.81E-03	1.62E-02	9.12E-03	3.89	2.25	0.58	4.25	2.39	0.56
Tc-99m DTPA - iv	2.85E-03	1.11E-02	9.66E-03	2.29E-03	8.19E-03	7.09E-03	3.89	3.39	0.87	3.58	3.10	0.87
Tc-99m DTPA Aerosol	2.20E-03	7.90E-03	7.50E-03	1.75E-03	6.06E-03	5.76E-03	3.59	3.41	0.95	3.46	3.29	0.95
Tc-99m Glucoheptonate	3.36E-03	1.35E-02	1.00E-02	2.69E-03	1.04E-02	7.42E-03	4.02	2.98	0.74	3.87	2.76	0.71
Tc-99m HDP	4.20E-03	7.45E-03	6.07E-03	3.40E-03	6.12E-03	4.80E-03	1.77	1.45	0.81	1.80	1.41	0.78
Tc-99m HEDP	2.95E-03	7.86E-03	6.55E-03	2.37E-03	6.10E-03	4.96E-03	2.66	2.22	0.83	2.57	2.09	0.81
Tc-99m HMPAO	4.69E-03	1.68E-02	1.29E-02	3.78E-03	1.38E-02	1.09E-02	3.58	2.75	0.77	3.65	2.88	0.79
Tc-99m HSA	5.30E-03	9.59E-03	7.54E-03	4.28E-03	7.85E-03	6.21E-03	1.81	1.42	0.79	1.83	1.45	0.79
Tc-99m MAA	5.22E-03	1.62E-02	1.54E-02	4.12E-03	1.27E-02	1.20E-02	3.10	2.95	0.95	3.08	2.91	0.94
Tc-99m MAG3	2.04E-03	1.64E-02	1.40E-02	1.60E-03	1.18E-02	9.99E-03	8.04	6.86	0.85	7.38	6.24	0.85
Tc-99m MDP	3.27E-03	7.64E-03	6.19E-03	2.64E-03	6.08E-03	4.75E-03	2.34	1.89	0.81	2.30	1.80	0.78
Tc-99m MIBI/stress	4.65E-03	1.55E-02	1.31E-02	3.77E-03	1.27E-02	1.07E-02	3.33	2.82	0.85	3.37	2.84	0.84
Tc-99m MIBI/rest	5.26E-03	1.83E-02	1.63E-02	4.26E-03	1.49E-02	1.33E-02	3.48	3.10	0.89	3.50	3.12	0.89
Tc-99m Pertechnetate	3.94E-03	1.32E-02	1.40E-02	3.18E-03	1.06E-02	1.14E-02	3.35	3.55	1.06	3.33	3.58	1.08
Tc-99m PYP	4.12E-03	7.46E-03	6.31E-03	3.34E-03	6.03E-03	4.95E-03	1.81	1.53	0.85	1.81	1.48	0.82
Tc-99m RBCs/in vitro	4.65E-03	9.19E-03	7.83E-03	3.75E-03	7.28E-03	6.11E-03	1.98	1.68	0.85	1.94	1.63	0.84
Tc-99m RBCs/in vivo	4.95E-03	8.95E-03	7.59E-03	3.99E-03	7.17E-03	5.99E-03	1.81	1.53	0.85	1.80	1.50	0.84
Tc-99m RBCs/heat	6.25E-03	5.55E-02	2.66E-02	4.94E-03	4.64E-02	2.24E-02	8.88	4.26	0.48	9.39	4.53	0.48
Tc-99m Slfr Cld/Normal	6.24E-03	1.69E-02	1.03E-02	4.99E-03	1.36E-02	8.04E-03	2.71	1.65	0.61	2.73	1.61	0.59
Tc-99m Slfr Cld/Disease	6.11E-03	2.60E-02	1.59E-02	4.88E-03	2.16E-02	1.32E-02	4.26	2.60	0.61	4.43	2.70	0.61
Tc-99m Slfr Cld/Oral	5.28E-03	3.00E-02	2.88E-02	4.72E-03	2.77E-02	2.68E-02	5.68	5.45	0.96	5.87	5.68	0.97
Tc-99m Teboroxime	4.75E-03	1.49E-02	1.23E-02	3.86E-03	1.24E-02	1.00E-02	3.14	2.59	0.83	3.21	2.59	0.81
Tc-99m WBC's	6.08E-03	2.39E-02	1.54E-02	4.87E-03	2.00E-02	1.29E-02	3.93	2.53	0.64	4.11	2.65	0.65
Tl-201 Chloride	6.79E-02	1.87E-01	1.65E-01	5.46E-02	3.16E-01	2.74E-01	2.75	2.43	0.88	5.79	5.02	0.87
Xe-127, 10 min breathold	2.22E-04	2.92E-04	2.92E-04	1.80E-04	2.36E-04	2.36E-04	1.32	1.32	1.00	1.31	1.31	1.00
Xe-133, 10 min brteahold	2.59E-04	3.79E-04	3.86E-04	2.02E-04	2.98E-04	3.04E-04	1.46	1.49	1.02	1.48	1.50	1.02

^{131}I-RETENTION MEASUREMENTS AND LOCALIZATION BY A HIGH-SENSITIVITY WHOLE-BODY COUNTER

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INTRODUCTION

A widely used method to investigate thyroid gland carcinomas but also disorders of the thyroid gland is the measurement of the ^{131}I-retention behaviour using a high-sensitivity whole-body counter (HWBC). Normally just the retention of oral administered Radioiodine is controlled. However, it is possible to visualize the distribution of ^{131}I (using a modified profilescantechnique) beside the retention in the human body and this provides much better information on human metabolism as will be demonstrated in the following.

MATERIALS AND METHODS

Experimental equipment

The profile scanning system used consists of four 8"×4" NaI(Tl)-crystals with multi-slit focused lead collimators inside a massive shielding chamber (total weight of the order of 74 t). Two of these linear scanning detectors are placed above and two below the bed and can be arranged independently in three dimensions. The scanning motion and the data aquisitation was controlled separatly for each detector by a central computer.

Clinical studies

To demonstrate the power of the high sensitive profilescantechnique we investigated 28 patients with total thyroid gland ectomy before and after ^{131}I-therapy. To ensure the absence of new metastasis and to localize remains of thyroid gland oral dosis with 20 µCi of radioactive Iodine has been provided. One hour and 72 hours after administration linear profile scan measurements relative to a comparative standard have been done by the HWBC. Thereby each patient was placed at a fixed position on the bed. The profile of count rates was measured by the profile scanning system (scanning time 600 s, scanning length 190 cm) using a special detector configuration shown in Fig.1.

To study the rate of accumulation of ^{131}I in the human body after its oral administration the retention value has been calculated from the measured count rates and a correction of background radiation and patient activity has been done.

$$R = \frac{P_{72h} - LW_R}{P_{1h} - LW_P} \cdot 100 \cdot \frac{S_{1h}}{S_{72h}} \quad [\%]$$

P_{1h}	count rates 1 h after administration (patient)
S_{1h}	count rates 1 h after administration (standard)
P_{72h}	count rates 72 h after administration (patient)
S_{72h}	count rates 72 h after administration (standard)
LW_P	patient activity
LW_R	background radiation

The profiles thus obtained for each scanning detector represents the longitudinal localization of the ^{131}I-distribution. To achieve the three-dimensional portrayal of the ^{131}I-distribution by the linear profile scan measurements a computational model[1] has been used. The model based on the special detector configuration allows to determine the distribution of radioactivity in the human body from the measured counts (beside the ^{131}I-retention behaviour). So count rates can be directly attributed to organs of the human body.

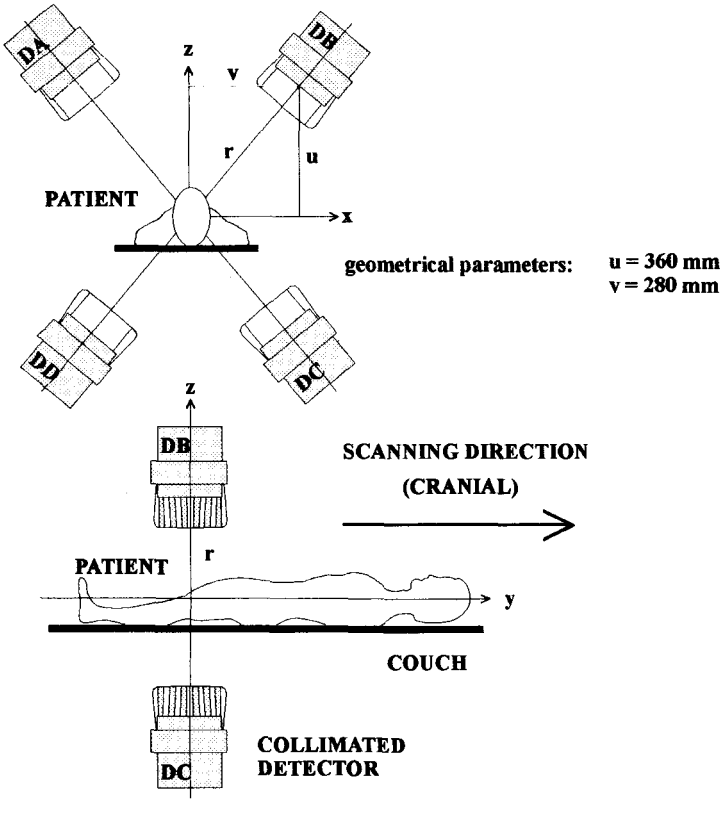

Fig.1. Detector configuration.

RESULTS

After oral administration of ^{131}I, the inorganic iodide is absorbed from the gastrointestinal tract within one hour and is completely transformed to plasma and extraiodide pool. Major portion of the total iodine content is taken up by metastases, most of which is in the form of iodinated aminoacids and the rest of the portion is excreted by the kidney. Normally a retention value of more than 3 % 72 hours after administration is interpreted as a possible of metastasis. Regarding the spatial I^{131}-distribution in addition a decision between localization and dislocalization is possible. Thereby it turned out that one of 10 patients with total thyroid gland ectomy before ^{131}I-therapy showed a retention value of 2.32 % although a localization could be detected shown in Fig.2. On the other side none of the patients after ^{131}I-therapy showed up a localization indicating a metastases even 25 % had a retention value of more than 3 %. Looking just at the retention values the possibiltiy of disorders cannot be excluded of certainty but regarding the three-dimensional distribution in addition a decision between localization and dislocalization is possible (Fig.3). So the physiological or pathalogical meaning of the retention value can be easily interpreted without further investigations by a gammacamera where much higher dosis (20 mCi) is applicated.

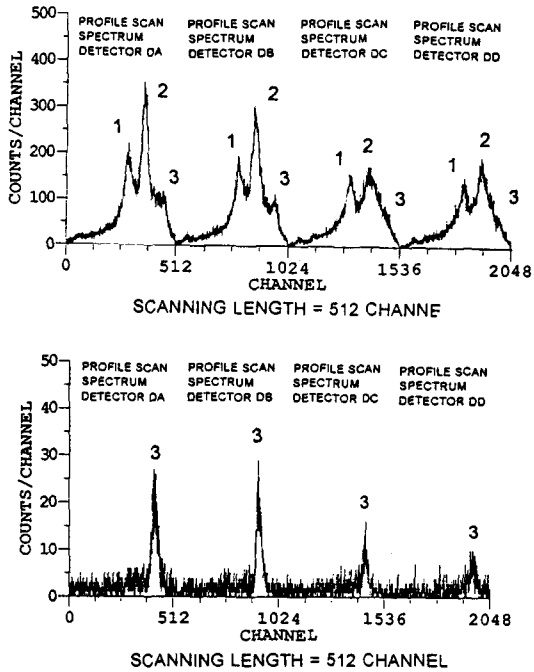

Fig.2. Patient with total thyreoid ectomy before ^{131}I-therapy, retention value: 2.32 %.

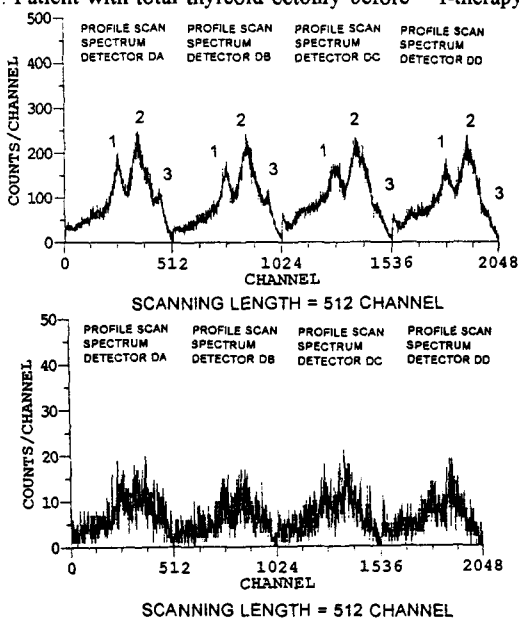

Fig.3. Patient with total thyreoid ectomy after ^{131}I-therapy, retention value: 7.58 %.

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DEVELOPMENT OF MATHEMATICAL PHANTOMS FOR CALCULATING INTERNAL DOSES FROM RADIOPHARMACEUTICALS USING PATIENTS' DIGITAL PICTURE OF BONE SCINTILLATION

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ABSTRACT

We made a new mathematical phantom using the patients' digital pictures of bone scintillation in nuclear medicine. The data of ^{99m}Tc bone scintillation pictures include the information on the body sizes and shapes. In the bone scintillation pictures, no three-dimensional data are available, so that the shapes and sizes of whole body and bones were modelled based on standard anatomical geometry. The organs except bone were also modelled after construction of the bone mathematical model. The mathematical phantoms were developed for each patient. The specific effective energy for each phantom can be calculated by the Monte Carlo code to compare it among the patients. Our mathematical phantoms would provide new calculation of internal doses from radiopharmaceuticals in place of the MIRD phantom.

INTRODUCTION

For the estimation of internal doses, many phantoms have been made(1). The calculation model based on the MIRD method have been used for the internal doses of the patients from radiopharmaceuticals on nuclear medicine(2-5). However, the mathematical phantom is based on the reference man(6), and have the uncertainty on its application to each patient because of the variety of body sizes among the patients. It is difficult to correct the uncertainty for each patient. On the other hand, the data of ^{99m}Tc bone scintillation pictures include the information of body sizes and shapes. We made a new mathematical phantom using the ^{99m}Tc bone scintillation pictures which include the information on the body sizes and shapes.

METHOD

The shapes and sizes of whole body and bones were modelled based on standard anatomical and physiological geometry(7). The organs except bone were also modelled after construction of the bone mathematical model. Every shape is geometrically defined as a function of x,y and z-coordinates. In our model, the determination of the x and y-coordinates is based on the picture data and the z-coordinate depends on the body size and weight of standard anatomical geometry. The calculation of the values of specific absorbed energy of a patient needs the exact organ shape and disposition of radiopharmaceuticals. However, it is very difficult and not so useful to define the body in detail for the propose of internal dose estimation. In our study, sensitivity analysis for adjustment was done for examining how the size and shape have an influence on the internal dose.

The ^{99m}Tc bone scintillation pictures are scanned and digitized in order to be read easily. For the bones and whole body, the x and y-coordinates of some points characterizing the shapes of the body and organs are directly read on the digital picture. On the pictures of the bone scintillation, the kidneys are often shown clearly, and the shapes can also be determined.

Those data are the basis for the definition of mathematical phantom as not only the geometry but also the source organs in the patient's body.

After reading pairs of coordinates, the solid geometry of phantom is defined by fixing the scale and the z-coordinate using the values of body height and weight as parameters. The variation of the patients' organ sizes is large compared with the mean value from reference data. This shows that the exact determination of one patient organs is impossible. The relations among organs, body height and weight are dealt with as parameters in the new model taking variations into account.

The new phantoms based on patients' data are able to be classified into some groups by body shapes and ages. In this study, the number of patients is limited, so we classified the patients into four groups, i.e. adult males, adult females, children and babies as the classification in which we take only body geometry into account. Using the classified phantoms, the internal doses of the patients who have no whole body data can be calculated.

MATHEMATICAL PHANTOM

As an example of mathematical phantoms, the shapes of main skeletons of a patient are shown in Figure 1 with the scintillation picture that is basis of the construction. The shapes of the bones are simplified into geometrical solids, and the disposition is based on the shapes on the picture. Considering the density and anatomy of body composition, the volume of the body and bones are determined respectively.

DISCUSSIONS AND CONCLUSIONS

On making mathematical phantoms, the information of shapes, sizes and components of both whole bodies and each organ are needed for each patient. In the bone scintillation pictures, no three-dimensional data are available. However, it is useful to make the phantom using two dimensional scintillation pictures compared with three dimensional data such as MRI or CT, because any subjects examined in nuclear medicine can be estimated without additional exposure to CT using our method.

The organ sizes are closely related with body volume and weight, and the relationship among them, e.g. brain etc.(8), have been reported. For the calculation, however, it is very difficult to apply the

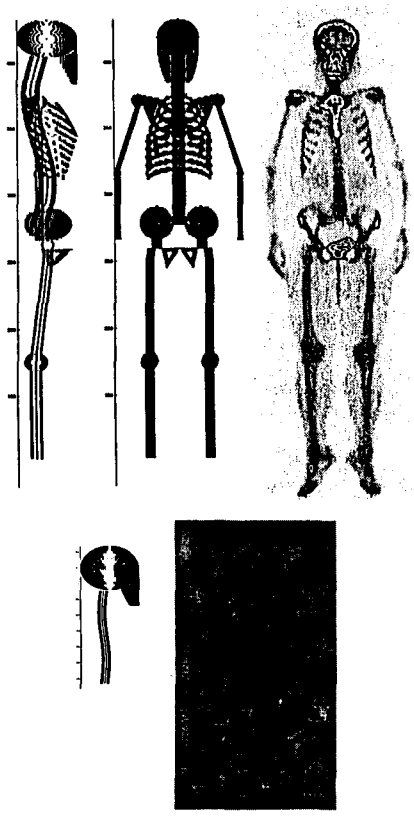

Figure 1. Adult and newborn patients' pictures and skeleton of mathematical phantoms

relation, because the organs reported are not all but several, and the individual variation of values is large.

The human organs are not stable in their lives including patients in hospitals. Arms and legs are typical parts, and the disposition of internal organs like stomach also does not rest in response to their move. The disposition among source and target organs those radiation sensitivities are high is more important than the exact definition of the patient shapes.

Although there are not exactly equal to the patients' bodies anatomically by the uncertainty, the merit of this phantom-construction method using picture data in nuclear medicine is to consider the body size and age of an individual patient for dose calculation. The phantom can also be applied for the patients who have not enough whole body picture data using the classified phantoms according to sizes and shapes. Especially, the age dependence of phantoms is available. The specific effective energy for each phantom can be calculated by the Monte Carlo code to compare it among the patients.

The study of differences of internal dose for the phantom of each body size based on anatomical and physiological data is more important than applying the mean phantom for each patient. Our method for constructing mathematical phantoms would provide new calculation of internal doses from radiopharmaceuticals in place of the MIRD phantom.

ACKNOWLEDGEMENTS

The authors wish to thank Dr.Sato of the Nippon Univ. and Mr.Hashimoto of the Jikei-idai Univ. for their suggestion.

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Internal Dosimetry on Intakes of Various Radiopharmaceuticals from External Measurement with TLD

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INTRODUCTION

Internal dosimetry resulting from nuclear medicine is important in comparing the benefit of a procedure with its potential risk. The estimation of internal dose due to intake of radioisotopes has been established by the Medical Internal Radiation Dose(MIRD) committee of the Society of Nuclear Medicine (1). In the MIRD method, the doses absorbed in target organs are estimated from the activities accumulated in source organs. The information on cumulated activities has been estimated from a few direct measurements with the positron emission tomography(PET) (2) and with a whole-body counter (3), but most data from the direct measurements with PET and SPECT(single photon emission computed tomography) relate to specific organs necessary for nuclear medicine treatment and are very scarce for other source organs in humans. Organ biodistribution of cumulated activities has usually been measured in animals and then extended to humans, despite the metabolic difference.

We have developed a new method to obtain the biodistribution of radioactivities, which are injected into the patient in nuclear medicine procedures, from external exposure measurement with thermoluminescent dosimeters(TLDs). In our new method, a number of TLDs are attached to the patient's body surface close to source organs to obtain information on body-surface doses. As the surface dose is connected to the cumulated activities in source organs through radiation transmission in the human body which can be estimated with the aid of a mathematical phantom, the organ biodistribution of radioactivities can be obtained by the inverse transform method. Here in this study, we estimated by this external measrement the organ biodistributions and absorbed doses of subjects to whom the radiopharmaceuticals of ^{18}F-labelled FDG(Fluoro-Deoxy-Glucose), ^{11}C-labelled Doxepin, ^{11}C-labelled YM 09151-2, ^{11}C-labelled Benzotropin, ^{67}Ga Citrate, ^{99m}Tc MDP(Methylene Di-Phosphonate) were administered in clinical nuclear medicine procedure.

This new method has great advantages, in that cumulated activities in several organs can be obtained easily with a single procedure, and that the measurements of body surface doses are done simultaneously with nuclear medicine procedure, as TLDs are too small to interrupt other medical measurements. We further compared our estimated results for human with the organ biodistribution and absorbed dose for animals directly obtained by the activity measurement and investigated the metabolic difference between human and animal.

MATERIALS AND METHODS

In the MIRD method, the absorbed dose in a target organ is expressed by the sum of contributions from several source organs, as follows:

$$D_i=S_{i,1}.A_1+S_{i,2}.A_2+..... \quad (1)$$

where D_i is the absorbed dose in the i-th target organ, A_j is the cumulated activity in the j-th source organ, and S_{ij} is the absorbed dose in the i-th target organ per unit cumulated activity in the j-th source organ, which means the transmission fraction of radiation.

In our new TLD method, the term of "target organ" is replaced by the term of "TLD position", as follows:

$$C_i=R_{i,1}.x_1+R_{i,2}.x_2+.... \quad (2)$$

where C_i is the absorbed dose in the i-th TLD position, x_j is the integrated activity in the j-th source organ during the TLD attachment on the body surface, and R_{ij} is the absorbed dose at the i-th TLD position per unit cumulated activity in the j-th source organ. The C-vector can be obtained by the TLD measurements, and the R-matrix can be calculated with the VADMAP code(4) by using the MIRD mathematical phantom, so that one can obtain the x-vector by performing the inverse transform of the above matrix equation with the SAND-II

unfolding technique.(5)

CLINICAL EXPERIMENTS

The measurements of body-surface doses with TLDs of BeO were done in clinical PET studies with ^{18}F-FDG, ^{11}C-Doxepin, ^{11}C-Benzotropin and ^{11}C-YM-09151-2 performed at the Cyclotron and Radioisotope Center (CYRIC) of Tohoku University, and in clinical scintigraphy with ^{67}Ga Citrate and ^{99m}Tc MDP performed at the Research Institute for Aging and Cancer of Tohoku University.

The subjects were five normal volunteers for ^{67}Ga Citrate and four normal volunteers for other radiopharmaceuticals. The positron emitters of ^{18}F (half life of 110min) and ^{11}C (half life of 20.4min) also emit 511 keV annihilation gamma rays, ^{67}Ga (half life of 3.26days) emits only gamma rays of 93.3keV (branching ratio of 37%), 185keV (20%) and 300keV (17%), ^{99m}Tc also (half life of 6.01hr) emits 140.5keV(89%) gamma rays.

Three sets of TLDs were attached at each position that was regularly selected without any interruption to clinical studies. The calibration of TLDs was done with 662keV gamma rays of ^{137}Cs whose activity was well characterized. The dose equivalent values in mSv units were directly obtained from the TLD reader after converted with the calibration factor. The average values measured with three TLDs were determined to be the C values in Eq.(2).

It is revealed in our previous study(6) that the TLDs must be attached on a body surface as close to source organs as possible in order to get the results with good accuracy. The following source organs were selected considering the metabolism of these radiopharmaceuticals;

1) For FDG, Doxepin, YM-09151-2, and Benzotropin: Brain, Heart, Lungs, Lever, Kidney, Spleen, Pancreas, Bladder, and Rest of the body; 2) For ^{67}Ga Citrate: Brain, Heart, Lungs, Lever, Kidney, Spleen, Pancreas, Bladder, Stomach, Small Intestine, Upper Large Intestine, Lower Large Intestine, and Rest of the body; 3) For ^{99m}Tc MDP: Brain, Heart, Lungs, Lever, Kidney, Spleen, Pancreas, Bladder, and Rest of the body. The TLD attachment positions were dertermined at 9 to 11 positions on a body surface close to source organs..

RESULTS AND DISCUSSIONS

As in this study only physical decay is considered neglecting the excretion from a subject, total cumulated activities in all source organs are kept constant to the injected activity and the cumulated activity in each organ was normalized to it.

Figures 1 and 2 show the cumulated activities in source organs measured with TLDs for ^{18}FDG and ^{11}C-Doxepin, respectively, as examples. The values are averaged for all of four subjects with errors.

In order to investigate the accuracy of the TLD method, we compared them with the cumulated activities in several source organs, brain, heart, lung, liver, kidney, pancreas, spleen, bladder and the rest of the body, directly measured with the PET image, for ^{18}FDG by Mejia et al.(2), and only in brain for ^{11}C-Doxepin.

This comparison reveals that the cumulated activities estimated with our TLD method give rather good agreement with those estimated with PET within a factor of 2. A big discrepancy only for the lever may be explained as follows; the lever is a large organ which situates near a body surface and its shape is complicated, which means the large deviation of individual from the reference man, and that the response matrix calculated from the MIRD model may include large errors.

From these figures and other results, the following facts are found out,

1) For ^{18}FDG, the brain, bladder and the remainder of the body are the source organs which mostly accumulate the radioactivities,

2) For ^{11}C-Doxepin, YM09151-2 and Benzotropin, the lungs and the remainder of the body highly accumulate the radioactivities,

3) For ^{67}Ga-Citrate, the small intestine and the remainder of the body highly accumulate the radioactivities,

4) For ^{99m}Tc-MDP, the bladder, bone and the remainder of the body highly accumulate the radioactivities.

The cumulated activities in human organs are also compared with those in animal organs, in order to investigate the influence of the metabolic difference between human and animals. The cumulated activities of organs in mouse for Doxepin and Benzotropin and in rat for YM-09151-2 were obtained from the time course of the activity data of each organ after the radioactivity injection through the direct activity measurements of the dissected organs. In Fig. 2, the animal's data of cumulated activities in source organs are compared with the human's data for Doxepin, as an example. These comparison revealed the following facts,

1) For Doxepin and YM09151-2, the cumulated activities of all source organs in human are larger than those in mouse and rat, except that lever activities are close together for human and mouse. The cumulated activity of lung in human is especially much higher than that in mouse and rat.
2) For Benzotropin, the cumulated activities of heart, liver, kidney and spleen in mouse are, on the contrary, higher than those in human, although brain activity in human is higher than that in mouse.

This result clearly indicates that the organ biodistributions in human and in small animals like mouse and rat are quite different each other for different radiopharmaceuticals because of their metabolic and blood flow differences and that it gives quite poor accuracy or wrong result in some cases to estimate the organ biodistribution and absorbed dose to human by measuring the organ biodistribution of animals.

CONCLUSION

It can be concluded from these clinical results that the new method to estimate organ biodistribution in humans from the surface dose measured with TLDs gives sufficiently good results considering experimental errors. This TLD method has great advantages, in that cumulated activities in several human organs can very easily be estimated, in contrast with the PET study that requires many procedures to estimate the biodistribution, and that the TLD measurements can be done simultaneously with medical study without interrupting it. Thus, this method would be useful in estimating the personal dose from medical procedures because of the easy handling of TLDs.

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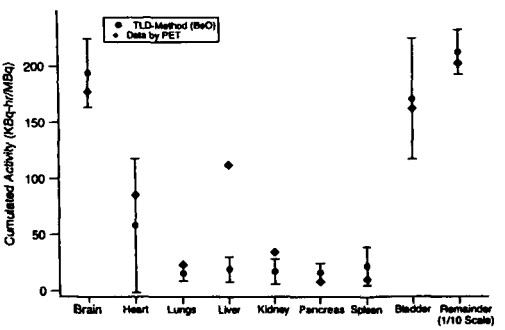

Fig. 1 Cumulated activities of source organs on intake of ^{18}F-FDG

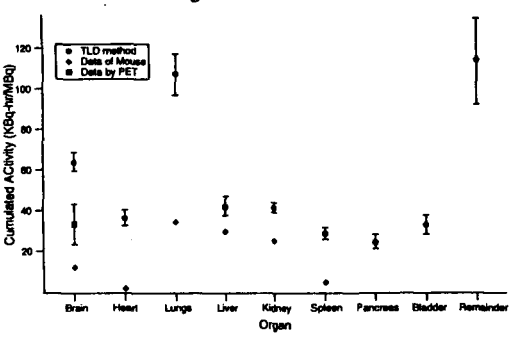

Fig. 2 Cumulated activities of source organs on intake of ^{11}C-Doxepin

EFFECT OF URINARY BLADDER MODELS ON RADIATION RISK ESTIMATES IN INDIAN PATIENTS DUE TO NUCLEAR RENAL DYNAMIC INVESTIGATIONS

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ABSTRACT

Effective dose estimation due to the administration of I-131-IOH and Tc-99m-DTPA, is carried out by modified procedure using constant volume standard MIRD model and dynamic urinary bladder model. These dose estimates are transformed to patients of India physique using data of Jain et al on specific organ masses and does transformation technique of Yamaguchi. For the same activity administered, the effective dose in Indian population is found to be 10-15% higher in comparison to their ICRP counterpart. The new bladder model gives higher effective does estimates due to I-131-IOH than the other model. In case of Tc-99m-DTPA, the dose estimates are found to be invarient of model employed. Further, the additional total stochastic risk caused due to these investigations are found to be marginally different in Indian patient population from ICRP population.

INTRODUCTION

Radiosotopic renal dynamic studies provides unequivocal information on renal failure, urinary tract infaction and obstruction, and renal transplant evaluation. The radiation dosimetry in such investigations is involved with calculation of radiation dose to inner mucosal surface and to the urinary bladder wall from the administered radiopharmaceuticals. Although Tc-99m MAG_3 has many desirable characteristics, however due to its prohibitive cost, it is seldomly used and still I-131-IOH and Tc-99m-DTPA are the radiopharmaceuticals of choice in the Indian Context. ICRP 62(1) has given effective dose coefficients for these radiopharmaceuticals in adults using constant volume standard MIRD phantom with ellipsoidal configuration (2). Besides the limitation of fixed volume (202.6 ml) in the model, no attempt was made to represent the physical dynamics of bladder filling and emptying. A new urinary bladder model of expanding sphere and with many other innovations to take into consideration some of the complexicities of the dynamic situation, has been proposed (3). A limited study carried out recently (4) showed that employment of the new model did alter the bladder wall dosimetry. A study is undertaken at the Institute to estimate effective dose due to the administration of I-131-IOH and Tc-99m-DTPA employing the modified procedure(s) and to investigate the effect of the two urinary bladder models on

the radiation risk estimates in Indian patients due to these radiopharmaceuticals.

MATERIALS AND METHOD

The effect of urinary bladder models on effective dose and radiation risk estimates is confined to adult patient population. These radiopharmaceuticals are excreted rapidly and deliver significant doses to urinary bladder wall which contribute more than 50% to the effective dose. As a first step, absorbed dose coefficients to the urinary bladder wall for similar initial volume and voiding schedule due to these radiopharmaceuticals for both the urinary bladder models, are taken from Thomas et al (3). As an average Indian differs from Reference man as regards anatomical and physiological characteristics, Dose TransformatioN Factor (DTF) are computed using data of Jain et al (6) for specific organ masses and the technique of yamaguchi (7) for Indian physique. Thus the DTF weighted organ absorbed dose coefficients gives absorbed dose coefficients for Indian subjects. Effective doses are calculated by employing the revised values of tissue weighting factor and modified procedure (5). The effective doses are multiplied by nominal probability coefficients for stochastic effects and the total activity used in the investigation for getting an idea of the risk associated with the procedure.

RESULTS

Comparison of the absorbed dose and effective dose coefficients for the two radiopharmaceuticals using the two different urinary bladder models, in Indian adults is shown in the table. Both the models give comparable values; however, the new model estimates higher effective dose due to I-131-IOH while there is practically no change in the case of Tc-99m-DTPA.

In general, the effective doses are 10-13% higher in Indian patients in comparison to the estimates made in Reference man for the same activity. In radiation risk estimates there is no effect of urinary models if Tc-99m-DTPA is employed while the use of I-131-IOH give marginally higher values.

Table Absorved and effective dose coefficients in Indian Patients due to administration of I-131-IOH and Tc-99m-DTPA

Radio Pharmaceutical	Absorbed dose coefficients in Indian patients in mGy/MBq		Effective dose coefficients in Indian patients mSv/MBq	
	MIRD	NEW MODEL	MIRD	NEW MODEL
I-131-IOH	0.31	0.37	5.30	6.10
Tc-99m-DTPA	0.036	0.036	0.52	0.52

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Radiation Dose to the Pediatric Population of Slovak Republic from Diagnostic Nuclear Medicine Procedures.

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Abstract

The increased number of *in vivo* diagnostic nuclear medicine examinations has created the need for more precise determination of radiation dose to the population, specially to the children.

A questionnaire survey has been performed on all nuclear medicine facilities in Slovak Republic through 1982 to 1994 with a special attention to pediatric patients in 1994. The information obtained was about the age distribution, number of different types of examinations, radiopharmaceuticals used and the value of mean administered radioactivity per exam. These data were used to evaluate the mean effective dose per exam and per capita, the collective effective dose for special type of examinations, for different radiopharmaceuticals and for radionuclides used in diagnostic procedures. In calculations we used the best available biokinetic models of the distribution of radiopharmaceuticals in organs as a function of age (2).

The results show that the Slovak Republic appeared favorable in comparison to other countries in the judicious use of diagnostic nuclear medicine procedures performed on pediatric population.

Materials and Methods

The effective dose provides a possibility of expressing the radiation risk to patients undergoing different radiodiagnostic procedures by means of single figure. To evaluate the radiation dose to population from radiodiagnostic procedures we used a mathematical formalism which is described in our previous paper (1).

In Slovak Republic 12 nuclear medicine departments are located. For years 1992 through 1994 we send our questionnaires to all nuclear medicine facilities in Slovakia and received 100 % response. The requested information was about the type of radiopharmaceuticals used and the mean administered radioactivity per examination, about the frequency of examinations and in 1994 also the information about the age distribution of the pediatric patients was received (there were 4 groups defined - up to 1 year, 5 years, 10 years and 15 years old children).

The mean effective dose for the procedure was determined by multiplying the administered radioactivity by the value of effective dose per unit of applied radiopharmaceutical's activity. These latter quantities were taken from the work of Stabin at al.(2), where the values of effective dose per unit of administered activity were published for 6 different age groups (newborn, 1 year, 5 years, 10 years, 15 years old children and for adults) and for different radiopharmaceuticals used in nuclear medicine procedures.

Results and Discussion

In 1994 in all 12 nuclear medicine departments in Slovakia 1742 diagnostic procedures on children have been performed what represents about 6.2 % of all exams performed that year. For comparison in 1992 it was about 7 % and in 1993 about 4.3 % (1). The frequency of pediatric examinations is approximately 1.3 nuclear medicine procedures per thousand children.

In Table 1 summary of all diagnostic nuclear medicine procedures performed in Slovak Republic on children in 1994 is listed, their number, the used radiopharmaceuticals, the values of mean effective dose per exam, collective effective dose and the quantity of mean administered radioactivity per different exams. The data are divided according to child's age. The collective effective dose from these procedures is about 3581 man mSv and the value of mean effective dose per exam was evaluated to be about 2.06 mSv. The most common procedures performed were: kidney dynamic scintigraphy (30.3 %), radiorenography (22 %) and bone static scintigraphy (13.1 %). The kidney dynamic scintigraphy was also

the main source of overall collective effective dose (22.8 %), the second biggest contributor was the bone static scintigraphy (20.5 %).

The data in Table 2 indicate relative frequency of radionuclides and their contribution to the annual collective dose equivalent in 1994 in Slovak Republic for pediatric patients. Approximately 75.3 % of radionuclide pediatric studies in Slovak Republic in 1994 were performed with ^{99m}Tc labeled radiopharmaceuticals, contributing about 88.3 % to the collective effective dose (Table 2).^{131}I was second most common used isotope (23 %) with the contribution to the S_E about 5.2 % ^{67}Ga and ^{51}Cr-based radiopharmaceuticals contributed to the S_E about 6.1 % and 0.4 % respectively.

Conclusion.

The radiation dose in nuclear medicine examinations might be used as the criteria for the quality of health care of patients. This is done by evaluating the mean effective dose per examination. In Slovak Republic this value for pediatric patients is approximately half of value for adult population - in 1992 the mean value of effective dose per exam was about 3.9 mSv (1). This is in agreement with the requirement that radiation dose to the child could not be higher than for adult. In other site the number of nuclear medicine examinations per 1000 inhabitants (1.3 exams/1000 children and 5.5 exams/1000 adults) is several times lower than in Czech Republic and other countries (1). The Slovak Republic appeared favorable in comparison to other countries in the judicious use of diagnostic nuclear medicine procedures.

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Radionuclide	number of exams	SE [man mSv]	number of exams in [%]	SE [%]
99mTc	1311	3164	0.753	0.8826
131I	401	187.8	0.230	0.0524
67Ga	12	219.6	0.007	0.0613
51Cr	18	13.5	0.010	0.0038
SUM	1742	3584.9		

Table 2: Relative frequency of used radionuclides and their contribution to the anual collective effective dose (SE) in 1994 for pediatric patients in Slovak Republic.

Examination	CTP codes of AMA	Radiopharma-ceuticals	1 year old child			5 years old child			10 years old child			15 years old child			SUM			Average Activity
			# of exams	ED /exam [mSv]	SE [man mSv]	# of exams	ED /exam [mSv]	SE [man mSv]	# of exams	ED /exam [mSv]	SE [man mSv]	# of exams	ED /exam [mSv]	SE [man mSv]	# of exams	SE [man mSv]	ED /exam [mSv]	[MBq]
Bone stat.scintigraphy	78300	99mTc-MDP	2	5.1	10.2	30	3.9	117	40	3.5	140	156	3	468	228	735.2	3.224	281
Angiography	78445	99mTcO4	10	11.8	118	21	11.2	235.2	17	9.2	156.4	13	9.9	128.7	61	638.3	10.46	185
Kidney dyn. scintigraphy	78707	99mTc-DTPA	29	2.5	72.5	33	2.7	89.1	76	2.2	167.2	95	2.1	199.5	233	528.3	2.267	40.6
Brain stat.scintigraphy	78600	99mTcO4	0	0	0	2	4.3	8.6	12	6.5	78	36	5.9	212.4	50	299	5.98	36
Kidney dyn.scintigraphy	78707	99mTc-MAG3	29	1.14	33.06	60	1.07	64.2	82	0.97	79.54	123	0.91	111.93	294	288.73	0.982	43.1
Kidney stat.scintigraphy	78700	99mTc-DMSA	23	1.35	31.05	45	1.6	72	53	1.4	74.2	58	1.35	78.3	179	255.55	1.427	87
Tumor stat.scintigraphy	78800	67Ga-citrate	0	0	0	0	0	0	6	22.2	133.2	6	14.4	86.4	12	219.6	18.3	107.2
Lung perfusion scintigraphy	78596	99mTc-MAA	3	1.85	5.55	19	1.8	34.2	24	1.6	38.4	30	1.6	48	76	126.15	1.659	47
Adrenal stat.scintigraphy	78075	131I-MIBG	0	0	0	11	8.7	95.7	1	5.6	5.6	1	3.5	3.5	13	104.8	8.061	18.5
Thyroid stat.scintigraphy	78010	99mTcO4	17	1.3	22.1	12	1.19	14.28	12	1.12	13.44	52	1	52	93	101.82	1.094	26.1
Imunoscintigraphy	78805	131I-MIBG	1	18	18	1	16	16	1	13.8	13.8	1	12.4	12.4	4	60.2	15.05	37
Cholescintigraphy	78220	99mTc-HIDA	3	3.51	10.53	3	3.8	11.4	2	2.7	5.4	7	2.7	18.9	15	46.23	3.082	67.8
Liver&spleen st. scintigraphy	78215	99mTc-hepatate	2	2.09	4.18	3	1.9	5.7	2	1.8	3.6	18	1.6	28.8	25	42.28	1.6912	95
Joint stat.scintigraphy	78300	99mTcO4	0	0	0	1	1.8	1.8	8	1.6	12.8	11	1.7	18.7	20	33.3	1.665	81.1
Meckel's diverticul	78290	99mTcO4	8	1.9	15.2	4	2	8	2	3.5	7	1	2.4	2.4	15	32.6	2.173	82.8
Radiorenography		131I-hippuran	3	0.007	0.021	55	0.07	3.85	172	0.06	10.32	154	0.056	8.624	384	22.815	0.059	0.27
Schilling test	78270	99mTcO4	1	4.32	4.32	2	4.02	8.04	0	0	0	0	0	0	3	12.36	4.12	80
Kidney stat.scintigraphy	78700	99mTc-glukonate	1	1.1	1.1	3	1.1	3.3	4	0.96	3.84	3	0.92	2.76	11	11	1	50
Bone marrow scintigraphy	78102	99mTc-nanocolloid	0	0	0	0	0	0	0	0	0	1	7.8	7.8	1	7.8	7.8	444
Red cell survival study	78130	51Cr-ERY	0	0	0	2	1.2	2.4	3	1.1	3.3	0	0	0	5	5.7	1.14	1.3
Platelet survival study	78191	51Cr-TROMB	0	0	0	0	0	0	8	0.3	2.4	5	0.27	1.35	13	3.75	0.288	0.87
Testes scintigraphy	78760	99mTcO4	0	0	0	0	0	0	0	0	0	2	1.55	3.1	2	3.1	1.55	111
Venography	78445	99mTc-MAA	0	0	0	0	0	0	0	0	0	1	0.9	0.9	1	0.9	0.9	60
Salivary gland scintigraphy	78230	99mTcO4	0	0	0	0	0	0	0	0	0	2	0.42	0.84	2	0.84	0.42	30
Parathyroid scintygraphy	78070	99mTcO4	0	0	0	0	0	0	0	0	0	1	0.52	0.52	1	0.52	0.52	33
Lymphatics scintigraphy	78195	99mTc-lymfoscint	0	0	0	0	0	0	0	0	0	1	0.176	0.176	1	0.176	0.176	10
SUM			132		345.8	307		790.8	525		948.4	778		1496	1742	3581		

Table 1: Summary of all diagnostic nuclear medicine procedures performed in SR on children in 1994; their number, radiopharmaceuticals used, mean effective doses per exam (ED/exam), collective effective dose (SE) and the mean administered radioactivity per exam.

Control of the biodistribution and biokinetics of 165-dysprosium-ferric-hydroxide after radiation synovectomy using the clinical whole-body counter in the General Hospital Vienna

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INTRODUCTION

Treatment of chronic rheumatoid synovitis is directed to control the inflammatory process causing pain and disability. Radiation synovectomy is suggested to be an alternative to surgical treatment /1/,/2/. Safety is one of the most important aspects when this method is applied. The physical properties suggest minimal exposure to non- target organs by the reduction of leakage of the radionuclide Dy-165 /3/,/4/,/5/. The aim of this study was to evaluate the applikation of the whole-body counter in the General Hospital Vienna for the study of the biodistribution and biokinetics of 165-dysprosium-ferric-hydroxide.

MATERIALS AND METHODS

From March 1995 to October 1995 it was possible to measure 6 out-patients and 3 in-patients, who were treated with 165-dysprosium-ferric-hydroxide (Dy-165). The use of Dy-165 has some considerable advantages: The half-life of Dy-165 (only 2.3 houres) is important to reduce the whole-body dose due to the relative high activity of approximately 11000 MBq (≈ 300 mCi) applied. The maximum soft tissue penetration of its β-particles is 5.7 mm, which is the range necessary to penetrate the inflammed synovia. The emission of γ-radiation acounts to two thirds of the 6 % of the whole disintegration is with an energy of 95 keV. This radiation is used to monitor the kinetic distribution of Dy-165 after injection with the whole-body counter.

The clinical whole-body counter

The clinical whole-body counter is placed in a room of the Department of Nuclear Medicine, which is specially built for the use of this instrument. The walls to the adjacent rooms are shielded with 6 mm lead to reduce the environmental background radiation. All coulers, lacquers and floor coverings used have very low radioactive concentration to reduce the background in the room (Figure 1).The tunnel construction consists of the transport construction and the shielding. The transport construction carries the guide rails of the movable patient-bed, the shielding of the measurement area and the holding device of the detectors and collimators. The lead shielding is divided in a measurement tunnel with 10 cm lead shielding and a shadow shield with 5 cm lead shielding. The patient-bed is made of acrylglass which guarantees a negligible absorption of the gamma radiation in the bed. Two slit collimators of 10 cm thickness can be brought in front of the detectors. The slit width can be varied continuously from 0 mm to 300 mm with an accuracy of ± 0.3 mm.

The localisation of Dy-165 is measured with four NaI-detectors of 6" diameter and 4" thickness. The detectors are arranged in pairs above and below the shadow shield. The scanning motion of the bed, the data acquisition separately for each detector, the storage of the data and the output of the results are controled by a computer.

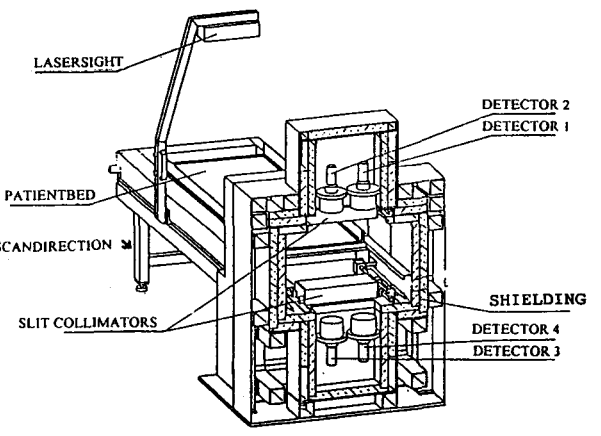

Figure 1.Inside arrangement of the whole-body counter

The scanning parameters used in this study are shown in table 1.

home position	150	scanning time	600 sec
start position	200	measurement steps	345
end position	3650	slit width	2,5 mm
scan length	3450 mm		

Tab.1. Scanning parameters used for the measurements of Dy-165 in patients

Patient	Sex	Joint	Bladder	Lymphatic node	Activity [MBq]
1	m	right knee	✓	-	519
2	f	left knee	✓	-	8445
3	m	left knee	-	-	9407
4	m	right knee	-	-	6444
5	f	right knee	✓	-	6436
6	f	right knee	-	right inguinal region	8888
7	f	left knee	-	-	3352
8	m	right knee	-	-	8144
9	m	right knee	-	-	6404

Tab.2.

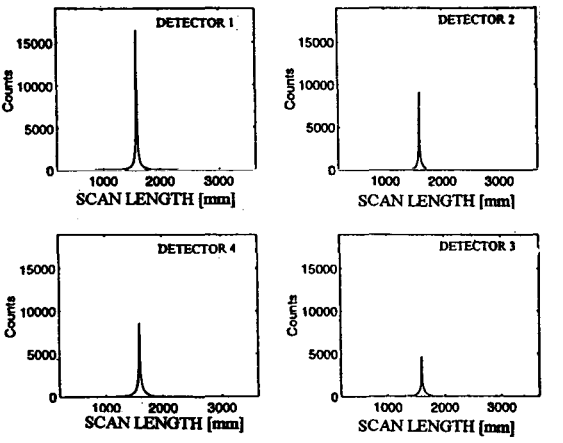

Figure 2.

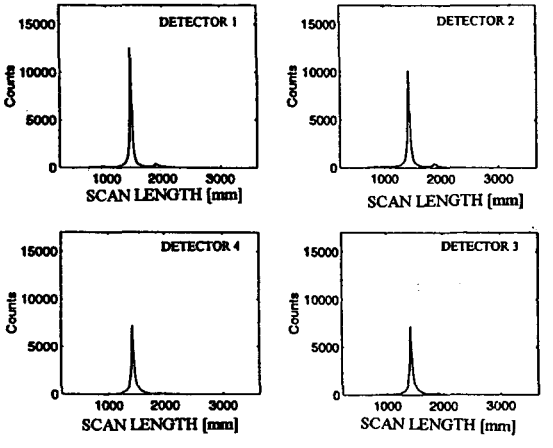

Figure 3.

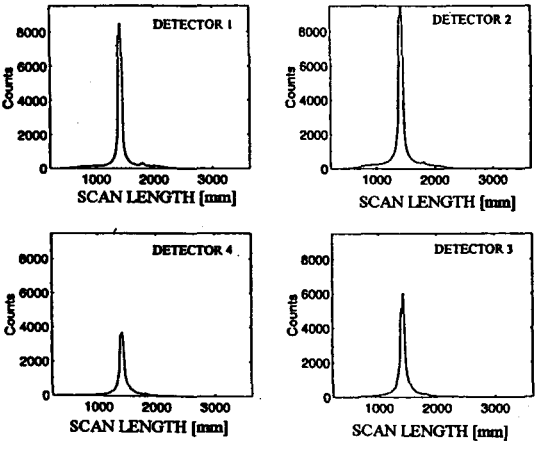

Figure 4.

RESULTS

Measurements with the whole-body counter result in activity profiles displayed as a plot of counts (impulses/channel≈1cm) against position along the scan-length (= longitudinal axis of the body), showing a peak as a activity deposit is traversed. Figures 2-4 show such activity profiles.

The results of the measurements of 9 patients can be summerized as followes:

With the aid of the whole-body counter it was possible to monitor the distribution of activity. The applied activity ranged from 500 MBq to 9000 MBq (15 mCi to 250 mCi) (Tab.2.). The activity profiles of 3 patients showed a uniform distribution of the nuclide in the whole knee joint. The measurement of 1 patient showed a concentration in the area of the hollow of the knee and in the case of 5 patients with small or no effusion the Dy-165 became concentrated in the knee. The activity profiles of 7 patients showed a considerable increase of the counts on both sides of the peak, as can be seen in figure 4. This increase is not the result of a leakage of the injected activity outside the knee joint, but can be explained as an effect of scattered rays of the highly energetic peaks in the γ-energyspectrum of Dysprosium-165. This effect arised only at activities above 6000 MBq. In 5 patients no leakage could be monitored. In 4 patients it was possible to determine some leakage outside the knee joint. In 3 cases occured a concentration of this activity in the urinary bladder. Using the software

MIRDOSE II (Oak Ridge Iac.) we determined the dose to the bladder, which amounted to about 15 mGy, 64 mGy and 50 mGy respectively. A more detailed interpretation of the activity profiles of the fourth patient revealed a small leakage of the injected activity out of the joint whereby a concentration was seen in the inguinal lymphatic nodes of the same leg.
Monitoring of activity profiles of more complex anatomical or pathological structures was also possible using the clinical whole-body counter. One patient suffered for an approximately 12 cm long effusion in the recessus suprapatellaris. The Dy-165 was homogenously distributed into both the joint and the recessus. The analyses of the activity profiles clearly showed an increase in the counts in the direction of the thigh, corresponding with the fact that Dy-165 was distributed in the joint and in the recessus.

DISCUSSION

The measurements with the whole-body counter represent a reliable method for the lokalisation of Dy-165 in the course of some hours after application. By the aid of the activity profiles of the four detectors it is possible to localize an incorporated or injected radionuclide with an accuracy of ± 1 cm in the three coordinates. The main peak marks the position of the knee joint, the small peak on its right side (Figures 3 and 4) represents the position where a small amount of activity had escaped. The count-rates of the four detectors make it possible to determine the position of this small peak and relate the peak to an anatomical structure. The rather high background activity results from scattered rays (Figure 4). With data received from phantom-measurements it was possible to determine the amount of activity applied and the exposure to non-target organs i.e. the urinary bladder and the lymphatic nodes. Using the MIRDOSE II software we determined the energy dose in these organs. The radiation exposure of the patient in the case of a concentration of escaped activity in the urinary bladder can be reduced by advising the patient to use the toilet immediately after detection of the leakage.
This method guarantees a very detailed information about the activity distribution. Our results show, that the leakage of 165-dysprosium-ferric-hydroxide out of the joint is neglectable. The radiation exposure due to this treatment is relative low.
The clinical whole-body counter of the Department of Nuclear Medicine in the General Hospital Vienna with its scanning device can be properly used for the localization of small areas of activities within the whole body of a patient.

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EFFECTIVE DOSE FROM DIAGNOSTIC NUCLEAR MEDICINE PROCEDURES IN NORTHERN GREECE

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INTRODUCTION

To estimate total radiation risk of diagnostic medical procedures, epidemiological studies are necessary. Concerning nuclear medicine investigations, only a few studies have been published in the literature (1,2). Furthermore in Greece, the number of nuclear medicine examinations has been increased during the last five years while new radiopharmaceuticals (^{131}I MIBG, ^{201}Tl chloride, ^{67}Ga citrate, etc) and new techniques are available.

A term that permits the comparison of exposure of patients undergone different medical examinations using ionising radiation is the effective dose, E. According to the recommendations of I.C.R.P. (3), the probability of stochastic effects depends not only on the absorbed dose but on the type and energy of radiation and the tissue irradiated, as well. This is taken into account by weighting the absorbed dose for radiation quality and for the relative contribution of that organ to the total detriment. The doubly weighted absorbed dose is called effective dose, E, and has replaced the previously used effective dose equivalent, E.D.E.

Minor changes have been applied to tissue weighting factors, w_T which resulted in reduction of effective dose from radiopharmaceuticals labelled with ^{99m}Tc and in increase of effective dose from iodide radioisotopes.

MATERIALS AND METHODS

Effective dose per MBq was calculated for a number of radiopharmaceuticals commonly used in nuclear medicine procedures. The calculations referred to adults of equal numbers of both sexes. For that reason for gonads we used the mean value of absorbed dose of testes and ovaries.

Effective dose is expressed by the formula

$$E=\sum_T w_T.H_T = \sum_T w_T \sum_R w_R.D_{T,R}$$

where w_T is the tissue weighting factor, H_T is the equivalent dose in a tissue or organ, w_R is the radiation weighting factor and $D_{T,R}$ is the mean absorbed dose over tissue T caused by radiation R.

The radiation weighting factor, w_R was taken equal to unity for all the γ, X and beta emitting radionuclides (3). To estimate the effective dose for a radiopharmaceutical, the absorbed dose $D_{T,R}$ was multiplied by the relative tissue weighting factor, w_T and the products were summed. Data regarding the mean absorbed doses per organ, $D_{T,R}$ for each one of the radiopharmaceuticals was taken from ICRP 53 (4), except for ^{99m}Tc sestamibi. Data for this radiopharmaceutical was taken from a description of a commercial kit.

We considered that the remainder consisted of the following tissues and organs: adrenals, muscle, pancreas, spleen, kidney, brain, upper large intestine, small intestine and uterus. To each of them, except uterus, we attributed a weighted factor of 0.0059. In the case of uterus (half population) the weighing factor was 0.0029. In exceptional cases, in which one of the remainder receives an equivalent dose in excess of the highest dose in any of the twelve organs, a weighting factor of 0.025 was applied to that tissue and a weighting factor of 0.025 to the average dose in the rest of the remainder.

The data concerning the number and the type of nuclear medicine examination used in this study, has been collected from the nuclear medicine departments of the hospitals AHEPA, Ippokratio and Theagenio, which cover all the nuclear medicine procedures done in public hospitals in Thessaloniki, second largest city in Greece. The number of examinations was 50102 during the period 1990-1994.

RESULTS

Effective dose per MBq of administrered radiopharmaceutical was calculated based on data from ICRP 53 and ICRP 60. The values of effective dose, E, are given in table 1. Effective dose equivalent (E.D.E.) values published in ICRP 53 are also given for comparison. The effective dose (E) values are compatible with those calculated by other investigators (5).

Table 1 also contains the type of examination/organ, the radiopharmaceutical used, the range of activities administered, the number of examinations, the value of E.D.E. and effective dose, E, and the collective E.D.E. and collective effective dose, S, in man-Sv.

Table 1. Effective dose per MBq of administrered radiopharmaceutical, collective E.D.E. and collective effective dose, S, from all examinations.

Exanination Organ	Radiopharmaceutical	Activity (MBq)	Number of examinations	% of total examinations	E.D.E (mSv/MBq) ICRP 26	E (mSv/MBq) ICRP 60	Collective E.D.E. (man-mSv)	Collective effective dose, S (man-mSv)	% of S
Thyroid	^{99m}Tc pertechnate	74-111	13660	27,26	0,013	0,013	18069	18069	13,50
Parotid gland	^{99m}Tc pertechnate	148-185	75	0,15	0,013	0,013	151	151	0,11
RBC	^{99m}Tc pertechnate	629-740	566	1,13	0,013	0,013	5146	5146	3,85
Meket	^{99m}Tc pertechnate	185	56	0,11	0,013	0,013	135	135	0,10
Bone	^{99m}Tc phosphonates	740	11750	23,45	0,008	0,006	69560	52170	38,98
Liver/Spleen	^{99m} Tc colloid	130-148	13584	27,11	0,014	0,01	26868	19192	14,34
Kidney	^{99m} Tc DTPA	259-296	4295	8,57	0,0063	0,0055	7716	6736	5,03
Kidney	^{99m} Tc DMSA	148	924	1,84	0,016	0,009	2188	1231	0,92
Brain	^{99m} Tc DTPA	740	187	0,37	0,0063	0,0055	872	761	0,57
Brain	^{99m}Tc pertechnate	740	450	0,90	0,013	0,013	4329	4329	3,23
Lung ventilation	^{99m} Tc DTPA	74	154	0,31	0,007	0,0063	80	72	0,05
Biliary	^{99m} Tc IDA	185-259	655	1,31	0,024	0,015	3182	1989	1,49
Lung perfusion	^{99m} Tc MAA	148-185	893	1,78	0,0012	0,0012	1632	1632	1,22
Red marrow	^{99m} Tc nanocolloid	370	45	0,09	0,014	0,01	233	167	0,12
Lymph nodes	^{99m} Tc microcolloid	74	156	0,31	0,014	0,01	162	115	0,09
Heart	^{99m} Tc Sestamibi	740	72	0,14	0,0043	0,0012	229	64	0,05
Vitamin B12 abs.	^{57}Co vitamin B12	0,019	195	0,39	2,7	2,2	10	8	0,01
Vitamin B12 abs.	^{58}Co vitamin B12	0,019	195	0,39	5,1	4,2	19	16	0,01
RBCV*	^{51}Cr chloride	2,22	315	0,63	0,11	0,074	77	52	0,04
RCST**	^{51}Cr chloride	4 - 5,55	184	0,37	0,11	0,074	97	65	0,05
Kidney	^{51}Cr EDTA	1,85	150	0,30	0,0023	0,0021	1	1	0,00
Ga-67	^{67}Ga citrate	155-185	268	0,53	0,12	0,11	5237	4800	3,59
Plasma volume	^{125}I HSA	0,11	315	0,63	0,34	0,28	12	10	0,01
Kidney	^{131}I hippuran	2,0-6,0	215	0,43	0,066	0,055	40	33	0,02
Adrenals	^{131}I MIBG	19	32	0,06	0,2	0,15	122	91	0,07
Thyroid-uptake	^{131}I iodide	1,11-1,85	420	0,84	15	25,1	6416	10736	8,02
Heart	^{201}Tl chloride	74-111	291	0,58	0,23	0,21	6553	5983	4,47
Total			50102	100			159135	133752	100

* Red Blood Cell Volume ** Red Cell Survival Time

The mean value of E.D.E. is 3.18 mSv per examination and the mean value of effective dose, is 2.67 mSv per examination. The frequency of examinations in effective dose range is shown in figure 1.

DISCUSSION

The collective effective dose compared to collective E.D.E. appears to be decreased, due to the different weighting factors. We must notice that effective dose from ^{131}I, which is used for thyroid uptake tests, is remarkably increased. Thyroid uptake with ^{131}I represents only 0.8% of the total number of examinations, but due to the relatively large radiation dose per investigation, contributes to the 8% of the collective effective dose. We should also mention that thyroid scans with ^{99m}Tc are 27.3% of the number of examinations, but its relative contribution is limited to 13.5% of collective effective dose.

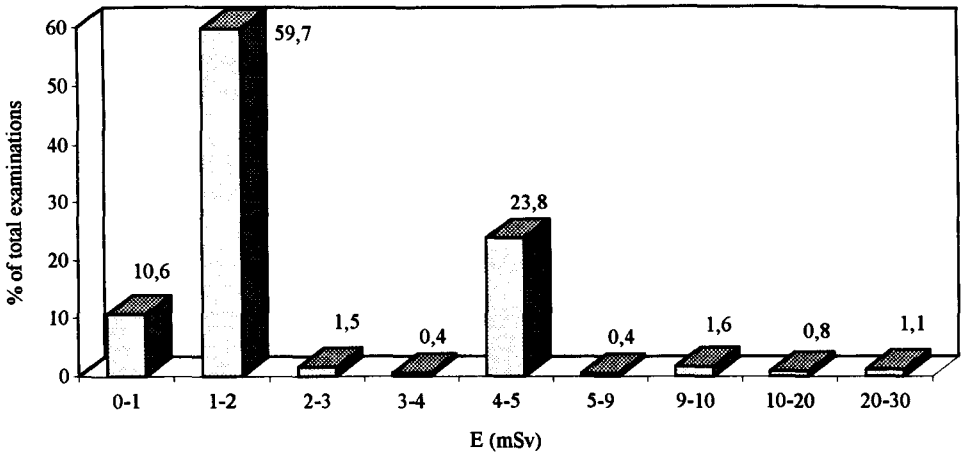

Figure 1. Percentage of examinations in effective dose range.

Although the nuclear medicine diagnostic procedures performed at the three hospitals are twenty eight in total, 80% of the collective effective dose, S, is due only to five types of investigations (thyroid, bone, liver/spleen, kidney-DTPA and thyroid uptake), which correspond to the 87% of the total nunber of examinations. Furthermore, bone scans contribute to 39% of the collective effective dose.

Various nuclear medicine examinations result to different effective dose values, but as it can be seen from figure 1, the 70% of the examinations corresponds to 0-2 mSv and the 24 % of them, due mainly to bone scans, corresponds to 4-5 mSv.

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NATIONAL COMPARISON OF ^{131}I SOLUTION MEASUREMENTS IN ROMANIAN NUCLEAR MEDICINE UNITS

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INTRODUCTION

The national comparison regarding ^{131}I solution activity measurement, organized by the Radionuclide Metrology Laboratory (RML) aimed at the quality assurance of the measurements of activity along the process of production, quality control, distribution and use of radiopharmaceuticals, Na^{131}I being one of the most used.

Such trial comparisons are often organized in various countries as a step for the assurance of the measurement traceability up to national and international standards. Their necessity was imposed by the fact that many errors are made in hospitals during the administration of radiopharmaceuticals, generally resulting in a supplementary irradiation of the patients [1-4]. RML , recognized as an authorized metrology laboratory, organized previously national comparisons regarding environmental radioactivity measurements[5-6].

The comparison involved the following steps: preparation, standardization, check and distribution of the solution, measurement and processing of the obtained results.

1. PREPARATION AND DISTRIBUTION OF THE SOLUTION

1.1. Preparation and standardization of the solution

The standard ^{131} I solutions had a radionuclidic purity superior to 99%. Three successive gravimetric dilutions were made, with a maximum uncertainty of the dilution factor of ± 0.1%. The third dilution solution was standardized absolutely, by using the $4\pi\beta$-γ coincidence method, in the variant of efficiency extrapolation. The activity concentration, a_{III} , was determined as the mean of nine sources prepared gravimetrically from the solution. The expanded uncertainty of the activity concentration was 0.52% (for a 99% confidence level). The concentration of the first and second dilution solutions, a_I and a_{II} , were calculated from a_{III} value, by using the dilution factor value. The activity concentrations and their expanded uncertainties , U, are presented in Table 1.

Table 1

Dilution	Activity concentration a, MBq/g	Expanded uncertainty, U,%
I	19.14	± 0.72
II	1.931	± 0.62
III	0.4837	± 0.52

1.2. Preparation of vials and check of activity

10 ml vials, normally used in the distribution of radiopharmaceuticals were selected to have as uniformly thick walls as possible; the selection was made by weighing empty vials; a difference of ± 0.5% from the mean weight was accepted. The solution was dispensed gravimetrically in vials, with a maximum uncertainty of ± 0.05%. The mass of solution in a vial m_i , was situated in the interval 4.8...5.2 g. The activities of the vials, A (MBq), were calculated according to the relationship

$$A_i = am_i \qquad (1)$$

where a is one of the values presented in the table 1 A total number of 36 vials were prepared. All were measured in the calibrated high-pressure ionization chamber CENTRONIC type IG12/A2. The measurements aimed at the assay of the weighing precision and uniformity of vial walls and of the dilution factors values used for the calculation of the values in Table 1.

The difference between the measured and calculated from weighing values was <±0.1%.

1.3. Distribution

The participants were selected as to cover the most important institutions involved in ^{131}I use and measurement, as our Radioisotope Department, as sole producer of radiopharmaceuticals in Romania, the National Institute of Metrology; the Control Laboratories of the Ministry of Health, 18 hospitals using mainly ^{131}I and ^{99m}Tc, analysis laboratories and the producer of Romanian calibrators. A form requiring detailed information regarding the type of the measurement equipment, calibration data, method of measurement, applied corrections, was issued. The participants were asked to asses the value of ^{131}I vial activity, in MBq, on the reference data, with the uncertainty value for a 68.3% confidence level and the possible presence of other contaminant radionuclides.

2. MEASUREMENT AND RESULTS

2.1. Assay of Radioisotope Department Calibrators

The first calibration, in the production department, is very important as in many cases this value is taken as reference when ^{131}I is administrated to the patients. Three calibrators (ionization chamber type), used currently in measurements, were verified. The check of calibration was made on a wide interval of activities A, situated between 2.4MBq and 95MBq, by using vials prepared from the three dilutions. The calibrations of the calibrators were made by their manufacturers and were verified regularly by using a reference source. Table 2 represents the obtained results, expressed as mean ratios between the measured activities, A_m , and A values on the whole interval

$$R = (A_m / A) \quad (2)$$

The standard deviation values, S_R , are presented, too.

Table 2

Calibrator	ROBOTRON,1989	ROBOTRON,1990	PITMAN,1978
R	0.969	0.986	0.948
S_R	±0.42%	±0.67%	± 0.58%

The conclusion is that all three calibrators assure calibration uncertainties less than 5.0%, generally declared by the manufacturers; however, a small underestimate of activity values can be noticed.

2.2. Measurements of the other participants

In contradiction with the literature reported intercomparisons, involving only hospitals and ionization chamber calibrators, in our case the area of participants was different, at the same time, the measurements were performed by using ionization chamber radionuclide calibrators, 12 results reported, and spectrometric Ge(Li) or NaI(Tl) systems, 6 results reported. The distributed vials had generally activities of about 10MBq with an expanded uncertainty of ±0.67%. In the case of calibrators, the measurements were made directly; no correction regarding background counting rate was applied. In the case of spectrometric measurements some elaborated procedures were applied. They included the following steps: preparation of measurement samples from the distributed solution; calibration of the spectrometric equipment using appropriate standards delivered by our laboratory [7] or other well-known laboratories as: IAEA, LMRI, Amersham plc, UVVVR. The measurements of samples were made generally by using the 364.5 keV quantum photopeak; all required corrections were applied. Table 3 represents the obtained R values, defined according to the relationship (2), by different participants, their reported uncertainties, S_R, measurement method and type of participant laboratory.

Table 3

No.of particip.	R value	S_R,%	Method of measurement	Type of lab.
1	1.276	10	Picker calibrator,10MBq vial	hospital
	0.768	10	Picker calibrator, 100MBq vial	

2	1.062	2.2	Picker calibrator	hospital
3	1.039	-	Mediac calibrator	hospital
4	1.007	1.6	Capintec CRC 12 calibrator	hospital
5	0.976	3.6	gamma spectrometry	control lab
6	1.007	1.4	standardized ionization chamber*	Metrology Institute
	0.997	0.94	gamma spectrometry	
7	0.998	-	Capintec CRC 15 calibrator	Control lab
	1.025	-	Pitman Calibrator	
8	0.906	-	calibrator	hospital
9	0.991	1.6	Robotron calibrator	hospital
	0.862	1.7	gamma spectrometry	
10	1.383	1.6	gamma spectrometry	analyses lab
11	1.669	-	Mediac calibrator	hospital
12	0.999	1.3	Romanian RI14G calibrator	manufacturer
13	0.978	2.5	gamma spectrom, point source	analyses lab.
14	1.006	3.2	gamma spectrom.point source	analyses lab.

*The same manufacturer as 12.

3. ANALYSIS OF RESULTS

The mean value, R , calculated from the tables 2 and 3 is R=1.040; S_R =±4.0%, and if we exclude the highly erroneous value R=1.669, the new result is R=1.008, S_R =±2.9%. The results are normally distributed near R=1.008 what denotes the fact that no systematic error is made during the measurements at the national level. The best results were obtained with Capintec and Romanian Calibrators and some spectrometric measurements, carefully made. By comparing the obtained results with the requirements of European and Romanian Pharmacopoeia [8,9] which impose an upper limit of measurement uncertainties of ±10% for radiopharmaceuticals, one may conclude that the ratio of laboratories measuring ^{131}I activity in a satisfactory manner is 76%. This figure is inferior to Germany 84% and the United Kingdom 96%. The situation of Romanian hospitals is worse, as we delivered samples to 18 hospitals and only 7 of them submitted results; the percentage of satisfactory results is only 66%. These situations were notified to these units and the correction was made.

CONCLUSIONS

1. The large scale ^{131} I solution national comparison demonstrated the traceability of methods and equipment of measurements to national standards; special attention must be paid in the future to the hospitals measurements.

2. The very good results obtained by some participants, using radionuclide calibrators and standard solutions delivered by well-known foreign producers proved once more, indirectly the traceability of our standards and measurement methods to the international standards.

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MEASUREMENT OF INCORPORATED HIGH ACTIVITIES WITH NUCLEAR MEDICINE EQUIPMENT

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INTRODUCTION

The purpose of this work is the comparison between two instruments frequently used in nuclear medicine: the Whole Body Counter and the SPECT-camera.
Since decades the WBC is a well-established method for quantitative measurement of activity. Apart from some methods of localisation the WBC serves exclusively for the proof of incorporated low activities.
The SPECT-camera is a relatively new imaging system with increasing popularity in nuclear medicine diagnostics. Because of its excellent tomographic abilities it is used almost always in a qualitative way. In the last years one tried to involve the SPECT-camera more and more also in quantitative tasks.
This work tests the capabilities of two of such instruments concerning the quantitative determination of such high activities like they are used for SPECT-acquisitions.

MATERIALS AND METHODS

In order to execute the quantitative measurement of activity under conditions near to such when measuring patients, a torso-kidney-phantom was developed according to MIRD (1) specifications. The phantom made of Plexiglas represents a 20cm high section of the human torso. Two kidney-cavities located at half height of the phantom can be filled with activity (see Figure 1).

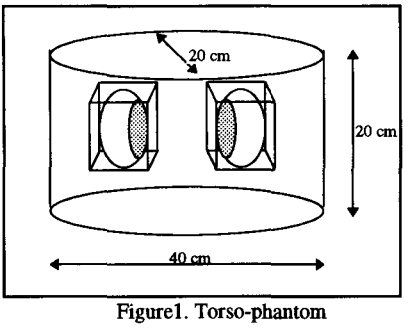

Figure1. Torso-phantom

The WBC (Canberra ACCUSCAN II) is build in *shadow shield*-geometry with 10cm strong steel slabs. Two detector-systems are available for the detection of radiation: A NaJ-scintillation-detector with a crystal size of 5x3x20 inches. It is mostly used for the quantitative evaluation because of its higher efficiency. And a co-axial Germanium-detector with 30% relative efficiency mainly used for the element-identification because of its higher energy-resolution.
The evaluation of the WBC-spectra was done using ABACOS-plus software installed on a DEC MicroVAX 3400.
At activities of 400 MBq of Tc^{99m} the detector-system of the WBC delivers deadtimes of 97%. With the help of a simple collimator built of 1cm strong lead-slabs, which leave a 5mm gap in front of the detector, the deadtime is reduced to 25%.
The source can be measured in *scan-* or in *fixed*-geometry. While deadtime-losses can not be corrected when the source is scanning under the detector, they can be partly corrected in *fixed*-geometry by adding the lost time to the predetermined acquisition time.

For the SPECT-measurements an ELSCINT Apex Helix HR doublehead-camera with an Apex SP-1 processing-station and CLIP-software was used. The acquisitions of the Tc^{99m}-filled kidney-torso-phantom were performed with those acquisition-parameters, which are standard in clinical kidney-studies: Isotope Tc^{99m}, 3° step-and-shoot acquisition, acquisition time 20min, matrix 128x128.

The following correction- and segmentation-algorithms were performed on the acquisition-data before and after the reconstruction: decay-correction, attenuation-correction according to the *first-order Chang* method (2), Compton-scatter-correction according to the *multi energy window* method (3) and finally for the determination of the active volume a segmentation according to a 30% *fixed threshold* method (4,5) or the method of the second deviation (6).

RESULTS

When measuring Tc^{99m}-sources from 1 to 500 MBq with the WBC in scan-geometry, the NaJ-detector shows an activity-dependent efficiency caused by short-term deadtime-effects (see Figure 2). This leads to negligible quantitative errors at activities under 50 MBq, but to errors up to 30% at higher activities. By using an efficiency-curve calculated by means of *least-squares-fit* instead of one single efficiency-value the accuracy for the entire area of activity can be increased to only 2.5% deviation.

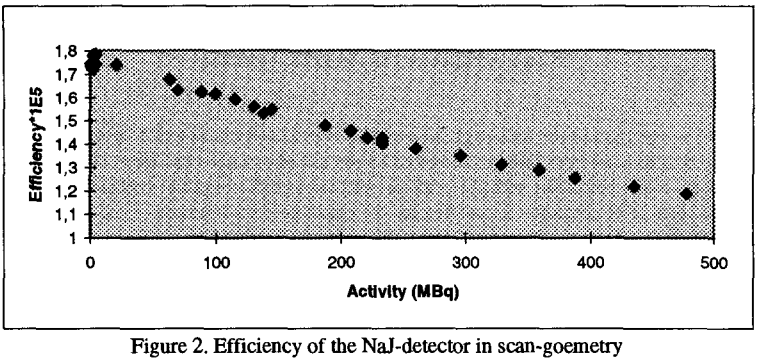

Figure 2. Efficiency of the NaJ-detector in scan-goemetry

Measurements with the Germanium-detector in scan-geometry deliver qualitatively analogous results, but with a more flat curve because of the detector's lower efficiency and the better time resolution (see Fig. 3).

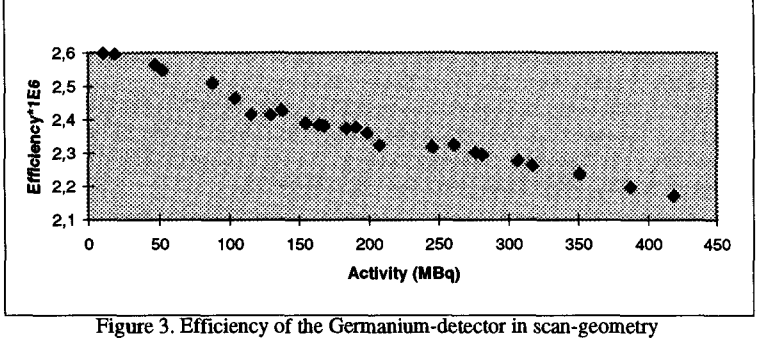

Figure 3. Efficiency of the Germanium-detector in scan-geometry

In a geometry with the source placed motionless under the detector, pile-up effects lead to a similar efficiency course despite electronic deadtime-correction. The above mentioned correction reduces in this geometry too the quantitatve deviations from 15% to 2.5% for NaJ- and Ge-detector.

The evaluation of the SPECT-acquisitions showed different accuracy for quantitative measurement of activity depending on the methods of correction used. The Compton-correction method implemented on this system was useless in this case, since the algorithm itself causes 20% quantitative error.

At the determination of the active volume no essential difference between the two segmentation methods was recognisable. With attenuation-corrected acquisition-data only a negligibly higher accuracy was to reach, than with uncorrected data. Under the ideal conditions of measurement present when working with a motionless phantom placed always identically a quantitative accuracy of 9% is possible with a SPECT-system. No activity dependence could be seen (see Figure 4).

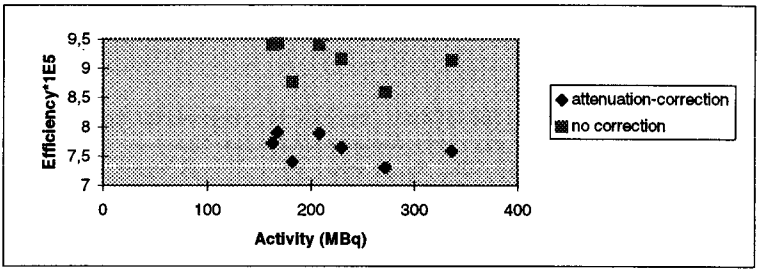

Figure 4. Efficiency of the SPECT-camera

DISCUSSION

Although the WBC is not thought for such high activities, it could easily be adapted by a simple slit-collimator. The still remaining deadtime effects could be corrected by the use of activity-dependent efficiency-values.
At activities between 1 and 500 MBq with the NaJ- as well as with Germanium-detector using this procedure an accuracy of less than 2.5% deviation is possible for the quantitative determination of activity. Even at slight changes in the position or size of the measured object this accuracy can be kept.

The most essential role in quantitative measurement of activity with a SPECT-system has the selection of the mechanisms of correction. Only certain combinations of correction- and segmentation-algorithms have a positive impact on the accuracy.
Only under the ideal conditions of acquisition when working with a phantom errors of 9% for the quantitative determination of activity and 25% for the determination of volume are possible with a SPECT-system. Deviations from this ideal conditions like the above mentioned cause uncertainties of more than 50%.

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PAPER TITLE QUANTIFICATION OF I-131 UPTAKE IN METASTASES FROM THYROID CARCINOMA

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MAJOR SCIENTIFIC TOPIC NUMBER 6.2. (see page 7)

ABSTRACT (See instructions overleaf)

Quantification of I-131 uptake in metastases from thyroid carcinomas is essential for appropriate patient selection and therapy dose calculation. Measurement of activity in opposing projections and tissue attenuation from a transmission scan has been considered the method of choice (M.J.Myers et al.,Br.J.Radiol., 1981 ,54,p.1062). The technique is both cumbersome and high activity of I-131 filled flood phantom pose a radiation hazard. Transmission measurements with a collimated point source and a uptake probe address safety concerns but make positioning difficult. Measurement of the reference standard under tissue equivalent absorbers selected to simulate soft tissue thickness calculated from anatomical imaging studies is both convenient and safe. Phantom studies have shown accuracy equivalent to the transmission technique.

Results.

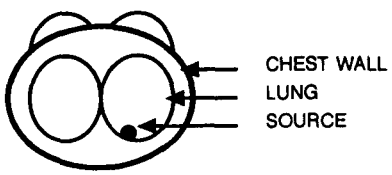

Fig.1 3M X-ray chest phantom used for simulations

activity	transmission method	reference std method
174 kBq	130 kBq	146 kBq
118 kBq	105 kBq	117 kBq
96 kBq	90 kBq	100 kBq
63 kBq	50 kBq	56 kBq
33 kBq	27 kBq	30 kBq

Tab.1 Evaluation of phantom activity using trans. and ref. std. methods

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PAPER TITLE Use of GafChromic Dosimetry Film for HDR Brachytherapy for Quality Assurance

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MAJOR SCIENTIFIC TOPIC NUMBER 6.3 (see page 7)

ABSTRACT (See instructions overleaf)

Both a blessing and a curse in the use of high dose rate (HDR) remote afterloading brachytherapy devices is the complete computer control of the clinical treatment. In our hospital's commercial system (Nucletron MicroSelectron), the treatment planning computer system downloads all information on source movement, both in time and in distance, onto a EPROM card that is used at the treatment console to control the Ir-192 source during treatments. Our QA procedure for the last 1.5 years involves testing this card prior to the patients first treatment to verify its correctness in controlling the source.

As a test we place clear GafChromic dosimetry medium (Model 37-041) in intimate contact with the applicator. The specific patients card then controls the irradiation (1-6 mins.). Examination of the now blue-colored film with a ruler allows quantitative determination of the distance between dwell positions seen as small dots and the offset of the first dwell position from the end of the applicator. Qualitative determination of relative dwell times is by inspection. It takes ~ 33 Gy to achieve an O.D. = 1 and to date no clinical case has given us a film too light or too dark to use. Advantages of this film include no need for external development so the image may be viewed immediately, insensitivity to normal room light, and archivability. The cost is ~ 2 $U.S./case. Examples will be given to demonstrate practical techniques of this radiochromic film.

ACCIDENT IN A LINEAR ELECTRON ACCELERATOR FOR MEDICAL USE

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ABSTRACT

Early in 1993, at a medical centre located in a north-eastern city in Argentina, the first 32 patients treated with a 6 MV photon beam electron linear accelerator, were accidentally irradiated for a total five weeks period at about twice the expected daily dose rates.

The accident occurred as a result of an erroneous calibration of the ionization chamber dosimeter of the medical centre causing an incorrect calibration of the accelerator's photon beam. Due to abnormal clinical effects shown by some patients, the radiotherapist took some actions that avoided an extension of the accident.

Fifty percent of the patients showed local injuries, while the severity observed in a patient's evolution could be associated with the received doses.

INTRODUCTION

During the commissioning in late 1992, the dosimetric acceptance tests of the linear accelerator were performed with a dosimeter owned by the supplier, because the ionization chamber dosimeter of the medical centre was only available in early 1993.

The radiotherapist had 8 year's experience in telecobaltherapy and had just finished his training in electron beam therapy, while the physicist who had been part of the operating team as required by the Argentine regulations, had completed his training in 1990. Since then the physicist had had further experience in telecobaltherapy without any additional performance in photon beam calibration.

INITIAL ACCIDENT CONDITIONS

1 - Situation before the photon beam calibration

The accelerator's physicist did with the following data before calibrating the photon beam: 1) The calibration certificate of the ionization chamber dosimeter, on whose accuracy the physicist did not doubt at all; 2) The results of the dosimetric acceptance tests performed by the supplier, indicating a value of approximately 1.4 m.u./cGy for the ratio between the readings -in monitor units (m.u.)- provided by the dose monitor installed in the radiation head of the accelerator and the doses to water delivered by the accelerator.

2 - Situation after the photon beam calibration

A beam calibration factor of Dw=2.3 m.u./cGy was obtained (absorbed dose in water phantom under standard irradiation conditions). This value could not be adjusted to the usual value of about 1 m.u./cGy and, consequently, an unsuccessful attempt was made to obtain advice from another physicists in Buenos Aires. The Dw value was not compared with the results of the accelerator acceptance tests, no double independent beam calibration was requested (1) and no inter-comparison of the medical centre's dosimeter and other dosimeter was made.

DEVELOPMENT OF THE ACCIDENT

1. The calibration factor for exposure in air of the medical centre's ionization chamber dosimeter, used to calibrate the photon beam when therapy started, was:

$$N_X = 47.41 \text{ Roentgen /digit}$$

2. Between the mid-second and late-third weeks in operation, some of the patients showed radiation induced early effects (diarrheic defecations, radiodermitis). Considering this situation, therapy was discontinued for the most compromised patients.
3. At such time, another physicist was called in and he verified that the doses planning was correct. An inter-comparison of dosimeters was planned but was not performed.
4. During the fourth week, treatments were restarted for some patients and treatments was started for some new ones. By the end of the fifth week, this second group of patients showed early clinical effects. Consequently, an inter-comparison of dosimeters was performed and the conclusion was reached that the doses received by the patients in the accelerator were 2.3 times higher than planned. At that time, all the treatments involving the use of the linear accelerator was discontinued.
5. A third physicist was called in and he assessed that the correct value of the calibration factor for exposure in air of the ionization chamber dosimeter was:

$$N_X = 109 \text{ Roentgen/digit.}$$

6. A total number of 32 patients were treated with dose rates higher than planned, before the dosimeter's calibration error was corrected.
7. Two weeks later, the accelerator's operation was restarted after reprogramming the treatments and since then no other patients have shown any unusual effects.
8. With decreasing frequency, an external physicist supervised planning work and doses administration with the accelerator.

RADIOPATHOLOGIC FEATURES AMONG THE AFFECTED PATIENTS

On the basis of the received doses, most of the patients showed early effects, particularly in their skin and mucosa, that is compartimental and high mitotic indexes tissues. Both the skin and the intestine are tissues with high α/β ratio values (10 Gy), in which the variation of the isoeffect dose with fractionating is weak (2); this fact explains the acceleration of the appearance of early effects and the increase of their severity with the administration of higher doses per fraction.

Table I shows the time T before appearance of early effects as of the initiation of the treatment and the doses received by the organs under treatment; Fig. 1 shows the excess of doses received by the patients with respect to that planned for the organs under treatment. The doses are indicated in terms of the instantaneous equivalent dose **De** given by Walinder's expression (3):

$$\mathbf{De(T)} = \int_0^T \mathbf{0.568}\ \dot{\mathbf{D}}\mathbf{(t)}\ \mathbf{(T-t)}^{-0.29}\,\mathbf{dt}$$

	Radiodermitis	Enterocolitis	Esophagitis
De (Gy)	13 to 26	14 to 24	17 to 21
T (days)	8 to 28	10 to 16	7 to 21

Table I: Instantaneous equivalent dose **De** received in organs under treatment and time T before appearance of early effects

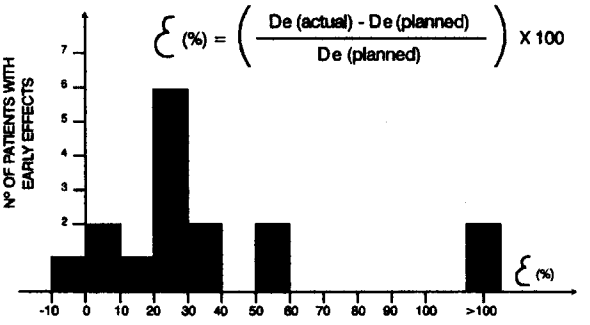

Fig. 1: Excess (%) of instantaneous equivalent dose **De** received, with respect to that planned for the organs under treatment

Concerning late effects, one of the patients showed radio-induced fibrosis and another one died after an intestinal resection a few months after receiving an instantaneous equivalent dose **De**=25 Gy. In the second case, there were early proctorrhage and symptoms compatible with those of a gastrointestinal syndrome. This involves the possibility of causality between this late effect and the cause of death.

The geographical isolation of the site, as well as the previous vital commitment of these patients and the scarce completeness of the clinical records of the patients, hinder in this accident the assessment of an unequivocal causality between late effects and the morbimortality associated with the treatments.

CONCLUSIONS AND LESSONS LEARNED

The accident was caused by an error in the calibration of the medical centre's ionization chamber dosimeter. The professionals involved did not perceive the hazard, in spite of the strong indications found (paradigm (4) of the accuracy of the calibration of the dosimeter).

The initial doubts were focused to the physicist's work; however, later on, considering the chronology of early signs and symptoms, suspicion arose that the doses being used might not be that foreseen and the decision was taken to inter-compare dosimeters. This confirms the relevance that the clinical follow-up of the patients has in order to early detect accidental situations, as it was pointed out in previous cases (5).

In order to prevent the occurrence of these type of accidents, a double and independent initial dosimetric calibration must be implemented (1) for electron linear accelerators, while technical training for the personnel must be intensified and made compatible with the complexity of the system they must use.

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Abstract for IRPA 9 (1996)

LESSONS LEARNED FROM ACCIDENTS IN RADIOTHERAPY

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There are situations that are unique to radiotherapy: persons (patients) are intentionally delivered very high radiation doses by exposing them to direct radiation beams and radiation sources are incorporated to their bodies as part their treatment. A departure of the prescribed doses may have severe or even fatal consequences. Not only overexposure but also doses below the intended ones are accidental exposures in radiotherapy. According to the Basic Safety Standards for Radiation Protection and for the Safety of Radiation sources, accidental exposures include any treatment delivered to the wrong patient, the wrong tissue, using the wrong source or wrong radiation beam, with a dose or dose fractionation differing significantly from the values prescribed or which may lead to undue secondary effects.

Lessons learned from previous accidents can avoid reoccurrence, at not only at the same facility but at any other facility in the world. For this reason, information on accidents has been collected and a review of more than 50 events has been made. The result is being published in an IAEA document.

The lessons learned, summarized in the document, can be used as checklist for testing the vulnerability of any given facility to the initiating events which triggered the accidents reviewed and factors which made it possible that an initiating event culminated in an accident are present. Initiating events and contributing factors are then classified and measures are derived from them to reduce the vulnerability of any facility. The lessons can also be used as a checklist for self control as well as for external audits to radiotherapy facilities.

NEUTRON SPECTRA OBTAINED FROM PROTONS ON A LITHIUM TARGET FOR BORON NEUTRON CAPTURE THERAPY

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INTRODUCTION

The Boron Neutron Capture Therapy (BNCT) will give a powerfull treatment of surface and deeper seated tumors. The basic principle of it is relatively straightforward. The stabile nuclide boron-10, with a large cross section for absorption by thermal neutrons, is concentrated in a tumor by means of a transfer substance like p-boronphenylalanine. The tumor is exposed to neutrons and the radiation emitted after neutron capture, $^{10}B(n,\alpha)^{7}Li$, has a short range in the tissue, in such a way that only tumor cells are destroyed. Thermal neutron beams can be applied for near surface tumors, while the epithermal neutron beams have their application in treatment of deeper-seated tumors. The epithermal neutron beams can be produced by reactors and accelerators. With accelerators different targets and reactions are used to obtain an appropriate neutron beam. This paper presents the neutron distribution and neutron doses obtained from lithum target where neutrons are produced in $^{7}Li(p,n)^{7}Be$ reaction. The proton energies considered are in the 2.0-2.4MeV range.

BNCT

BNCT is a binary radiotherapy modality (1). In BNCT, boron-10, which has a particulary large cross section for capture of thermal neutrons (neutrons having energies less than 0.5eV) is preferentially introduced into the malignant tissue by administration of a suitable boronated pharmaceutical. A thermal neutron field is then generated within the irradiation volume by application of an external beam produced by a nuclear reactor, accelerator or, possibly some other neutron source. The objective of this procedure is to cause selective destruction of the malignant tissue by energetic secondary charged particles, specifficaly helium-4 and lithium-7 ions that results from neutron capture interactions in boron-10. The total energy of this charged particle pair is 2.35MeV. This energy is deposited along charged particle tracks that are comparable in length to cellular dimensions, thereby offering the possibility of cancer cell inactivation with only limited damage to nearby healthy tissue. In recent years, the direction of BNCT research worldwide has been strongly influenced by the recognition that an epithermal neutron beam (neutrons predominantly in the energy range of 1-10keV) will ultimately prove to be optimal for human clinical application. Such a beam is expected to have an advantage compared to the thermal neutron beam currently used in Japan because epithermal neutrons will penetrate a few centimeter into tissue before forming a thermal peak. The properly designed neutron beam is, itself, not expected to cause unacceptable normal tissue damage. Therefore, with an epithermal neutron beam it may be possible to treat some deeper seated tumors without surgery during irradiation and also spare healthy surface tissue.
It is the purpose of our work to develop therapeutical neutron sources based either on a reactor or an accelerator properly designed for operation and irradiation under the condition,

i.e. infrastructure of an hospital. With this paper neutronic calculations and considerations of the layout are discussed.

REACTORS AND ACCELERATORS

Epithermal neutron beams for BNCT can be generated by nuclear reactors and by accelerator based neutron sources. So far, however only reactors have actualy been used to produce therapeutically useful epithermal neutron beams for BNCT. Some low intensity prototypes of accelerator based sources, generally featuring the use of proton beams and beryllium or lithium targets have already been constructed.

Our group suggested a reactor based on MTR type fuel plates as a therapeutical neutron source (2). Water is used as coolant and in-core moderator. Jackets of various materials surrounding the core for ex-core slowing down have been optimized for epithermal neutron extraction. The reactor core design is based on the core envisaged for the FRM-II research reactor at Munich. The neutronic calculations show, that the necessary flux intensity at the irradiation position can be achieved with a power of about 200kW, which is much lower than the power of the reactors (some 10MW) used hitherto.,

One of the promising concepts for use of accelerator based neutron sources for BNCT is the use of 2MeV protons on lithium target, using the $^{7}Li(p,n)^{7}Be$ reaction. There are four important properties of this reaction which makes it a promising candidate for a neutron source for BNCT. First, the energies of emitted neutrons are below 800keV which is not too far from the required neutron energy needed for BNCT. The neutron energy required for BNCT is between 1keV and 10keV with reference of the treatment of surface or deeper seated tumors. Second, the total cross section of the $^{7}Li(p,n)^{7}Be$ reaction is relatively high and it is in the order of a few hundreds of mb for proton energy of 2-2.5MeV. Third, the target is a solid one and not radioactive. Fourth, the power released on the target is 25kW for 2.5MeV protons and 10mA of current. This current reaches the upper limit achievable with accelerators at present. It is, therefore, necessary to optimize the yield of neutrons with energies suitable for BNCT, i.e. 1-10keV.

NEUTRON SPECTRA FOR LITHIUM TARGET

The neutron spectra ($Y_n(E_n,\Theta)$) for maximum proton energy (E_{pmax}) of 2400keV is calculted by equation [1].

$$Y_n(E_n,\theta)\left[\frac{neutrons}{ssrkeVmA}\right] = 6.242\times10^{15}\,\frac{1}{(E_{pmax}-E_{th})}\,\frac{N(Li)}{\left|\partial E_n/\partial E_p\right|}\,\frac{d\sigma(E_p,\theta)}{d\Omega}\,\frac{1}{\left[dE_p/dx\right]} \qquad [1]$$

A solid lithium thick target (some 100μm) is considered. The neutron spectra is calculated for different neutron angles (θ). The calculations are done for three proton regions. For $E_p \geq 1950$keV the energy dependent cross section is used (3). For proton energies between threshold E_{th}=1881keV and 1950keV the cross section of equation [2] is used for neutrons that have positive cosφ in the c.m. system. In addition, a third region of proton energy considered was between 1881keV and 1920.6keV, within which the neutrons may have also negative cosφ in the c.m. system.

$$\frac{d\sigma(E_p,\theta)}{d\Omega} = \frac{\sigma_t(E_p)}{4\pi}\left|\frac{\sin\varphi d\varphi}{\sin\theta d\theta}\right| \qquad [2]$$

θ is the direction of the neutron with respect to proton in the laboratory system and φ is the neutron emmision angle in the centre of mass system.

The total neutron yield obtained from 2.4MeV protons is higher than from 2MeV protons, but the neutrons obtained from these protons have lower energies. For BNCT the neutrons between energies from 1keV and 10keV are needed. The neutrons obtained from 2MeV protons are not far in their energies from the BNCT required energies. With the proton beam of 2MeV the neutrons emitted at $\theta=45°$ have about a constant neutron distribution between 40keV and 160keV. The neutron yield is about 1.2×10^{6} neutrons per (keV s sr mA). For neutrons emitted at $\theta=60°$ the neutron spectrum between 30keV and 120keV is constant with a yield of 10^{6} neutrons/(s sr mA). This energy is so close to the therapeuticaly required that a minimum of slowing down is necessary. Protons with higher energies will yield more neutrons, but those neutrons have also higher energies demanding for more slowing down afflicted with losses. Therefore we have concentrated our calculations on the 2MeV proton beam for creating neutrons.

MODERATOR CONSIDERATIONS FOR LITHIUM TARGET

The lithium target is situated in the centre of Ni sphere with radius of 100cm. The Ni acts as a reflector. In the Ni sphere the channel for proton beam (200mm diameter), a cooling gap (1cm) and a neutron channel (300mm diameter) are simulated. The axis of the latter is rotated for 60° toward axis of proton channel. Zu addition a D_2O moderator (20cm radius) is located at the begining of the neutron channel. The forward, direct neutron beam with too high energies is slowed down by moderator and appropriate axis angle of neutron channel.
The results of first calculations made with MCNP code (4) are given in Table 1. Different collant materials are tested. A modified reflector/moderator combination ($Ni+Ni/D_2O$) is also considered.

Moderator	$Ni+D_2O$	$Ni+Ni/D_2O$	$Ni+D_2O$	$Ni+D_2O$	$Ni+D_2O$
Coolant	H_2O	H_2O	D_2O	N_2	Ar
Φ_{et}(n/cm^{2}s)	1.80×10^{-3}	1.79×10^{-3}	1.82×10^{-3}	1.81×10^{-3}	1.81×10^{-3}
D_{et}(Sv/s)	3.65×10^{-14}	3.64×10^{-14}	3.67×10^{-14}	3.68×10^{-14}	3.68×10^{-14}
Φ_{es}(n/cm^{2}s)	5.30×10^{-5}	5.27×10^{-5}	5.05×10^{-5}	5.19×10^{-5}	5.19×10^{-5}
D_{es}(Sv/s)	8.35×10^{-16}	8.3×10^{-16}	8.12×10^{-16}	8.35×10^{-16}	8.35×10^{-16}

Table 1. Relative values of epithermal fluxes and doses on target and Ni sphere surface

CONCLUSION

According to the calculations, the assumed neutron source intensity based on a proton beam with 25kW will provide a sufficient neutron flux at the irradiation position. (The flux ratio between the target and the Ni sphere surface (1m distance) is 100).
The comparison of different material combinations shows low difference in flux and dose values, so that there is enough flexibility for shielding materials and cooling.

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INVESTIGATION OF DEUTERATED MODERATORS FOR D(d,n)^{3}He NEUTRON CAPTURE THERAPY

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INTRODUCTION

Neutron capture therapy (NCT) is becoming an important technique for brain and skin cancer therapy. An atom of high cross-section for neutron interaction (mostly ^{10}B) is incorporated into tumor cells, then, following a neutron interaction, charged particles are emitted causing cell lethality. So far, in Japan and the US, nuclear reactor were used as neutron sources. We propose the use of neutron generators as the source of neutrons for this purpose, because it has some advantages as compared with the nuclear reactor (1). The neutron generators use charged particles (electrons, protons and light nuclei) or photons interaction with a target atoms (D, Be, W) for the neutron production. One of the basic parts in the proposed neutron generator is a moderator-reflector assembly, which will transform the primary neutron energy spectrum to that required for NCT. The requirements for NCT (a low γ-rays dose and neutron energy in the range 1eV-50keV) impose the use of moderator-reflector assembly of complex geometrical shape and the use of most effective materials. The best material for that purpose is a heavy water (D_2O) (1-3), however it has several disadvantages such as: high price, the necessity of making of complex shape container, because D_2O can be used only in liquid phase, and difficulty in free trade. The purpose of our work is to find the materials that can serve the purpose as heavy water, free of these disadvantages.

MATERIALS AND METHODS

We have investigated neutron moderation properties of the deuterated polyethylene $(C_2D_4)_n$, deuterated polystyrene $(C_8D_8)_n$, heavy water, polyethylene and paraffin. Calculations were made with the code MCNP, version 3.A . We have used the cross-section library ENDFB.4, corrected with the code NJOY for 300^{o}K. We have investigated a simple geometrical model where the neutron source was located on the symmetry axis between the moderator and the reflector, both having a cylindrical shape. Beryllium (Be) was chosen as the reflector material. The thickness of the reflector was 10.5 cm (1). Two types of the neutron sources were assumed in the calculations: monodirectional and isotropic. The neutron energy was 2.8 MeV, corresponding the D(d,n)^{3}He reaction (1). In these calculations the following parameters were considered: the number of neutrons with energy below and above 100 keV, crossing an external surface of the moderator, in cylindrical sectors corresponding to angle intervals of 10^o in the 0^o-70^o range as a function of the moderator thickness; the number of photons, produced by neutron interactions with the moderator and reflector; the photons energy spectrum; neutrons energy spectrum, with 50 keV energy steps in the mentioned sectors and the average neutrons energy in the mentioned sectors.

RESULTS

An optimal thickness was found for each of investigated material so that a maximum number of neutrons with energy below 100 keV was obtained at the external surface of the moderator, in the angle interval 0^o-70^o. In Table 1 are shown the optimal thicknesses of the moderators, in cm and the normalized numbers of neutrons of energy below 100 keV, for each material, for the two sources. It is seen in Table 1 that the neutrons number for

deuterated polyethylene is smaller then that for heavy water, and that deuterated polystyrene provides

Table 1: Relative Epithermal Neutron Intensities

Source type \ Material		Heavy water	Heavy polyethylene	Heavy polystyrene	Polyethylene	Paraffin
Mon	Relative No. of neutrons	1	0.992	0.889	0.43	0.43
	Opt.tickness	18	14	18	7	8
Isot.	Relative No. of neutrons	1	0.969	0.904	0.482	0.497
	Opt.tickness	14	12	14	6	6

12% less neutrons. The neutron energy spectra for D_2O and $(C_2D_4)_n$ in the angle intervals 0^o-10^o and 40^o-50^o (the monodirectional source case) are given in Fig.1. It is seen that deuterated polyethylene decreases the number of neutrons in the higher energy interval better than heavy water, in the 0^o-10^o interval and there is practically no difference in the 40^o-50^o interval.

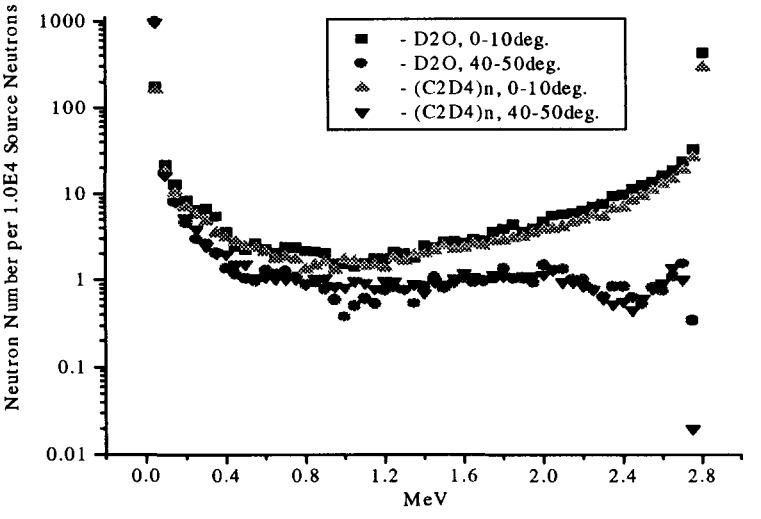

Figure 1. Neutron Energy Spectra

In case of an isotropic source the same variation in high energy neutrons intensity is obtained, for D_2O and $(C_2D_4)_n$, for both directions. A comparison between deuterated polystyrene and heavy water gives inferior results, but significantly better than that for polyethylene and the paraffin. The values of the average neutrons energies at the external surface of the moderator in the angle intervals 0^o-10^o and 40^o-50^o are shown in Table 2. It is seen that the average neutron energy after moderation with $(C_2D_4)_n$ is less than that with D_2O for both neutron sources, it is about the same for $(C_8D_8)_n$. In Figs. 2 and 3 are shown the photons intensities produced by neutron interactions in the moderator and reflector, and their energy spectrum

respectively, for the isotropic source. It is seen in Fig.2 that $(C_2D_4)_n$ and $(C_8D_8)_n$ moderators produce more photons than D_2O, but together with the reflector, the difference in photons number is smaller.

Table 2: Average Neutron Energy

Source type \ Material		Heavy water	Heavy polyethylene	Heavy polystyrene	Polyethylene	Paraffin
Mon	0 - 10 deg.	1.943	1.802	1.872	2.343	2.315
	40 - 50 deg.	0.118	0.117	0.158	0.362	0.346
Isot.	0 - 10 deg.	0.518	0.408	0.55	1.063	1.13
	40 - 50 deg.	0.278	0.208	0.304	0.763	0.817

Fig.3 shows that there are no photons of energy 4 MeV from D_2O, but estimating the contribution of these photons for other moderators to the total ionization of all photons, gives about 13-15%.

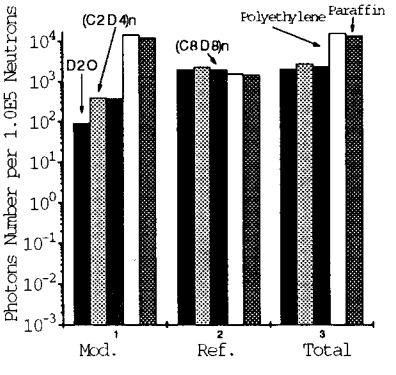

Figure 2. Number of photons produced by neutrons

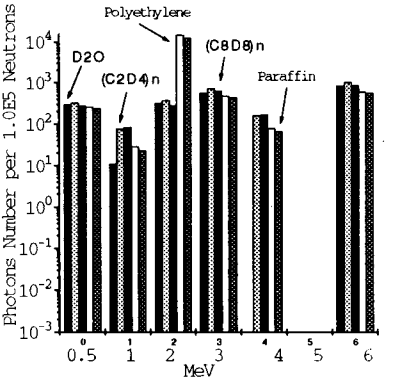

Figure 3. Photon energy spectra

CONCLUSIONS

The results show clearly that deuterated polyethylene can replace heavy water as a neutron moderator for NCT. Deuterated polyethylene was produced in the past (4) and was used as a thin target in accelerator experiments. Its production might be cheaper and easier than that of heavy water.

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SUBSIDIARY NEUTRON DOSE TO PATIENTS WITH PROTON THERAPY

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INTRODUCTION

In the treatment planning of proton beam irradiation, owing to its excellency in dose distribution, radiologists can administrate doses larger than those in the conventional therapy with high energy X-rays. The remarkable performance of the proton therapy with deep-seated tumors is now widely recognized[1]. In the irradiation port of proton beam, collimators and other devices inserted upstream of the beam to the patient interrupt a considerable part of the protons bearing kinetic energy up to 200 MeV. As is well known, high energy protons can generate high energy neutrons when they interact with matter[2]. These strongly penetrating neutrons cause subsidiary exposure to the patient treated with proton beam. One of the authors (Takada) had estimated that transporting efficiency of the irradiation facility employing so-called (single) scattering method is ten percent or so at best[2]. Hence, for the purpose of providing information in making a clinical decision to select the treatment modality and for radiation protection, it is of interest to evaluate these neutron doses and know their importance. Investigations were made at the Proton Medical Research Center (PMRC), the University of Tsukuba.

A CRUDE ESTIMATION OF SUBSIDIARY NEUTRON DOSE

Firstly, we show a crude estimation of the patient neutron dose based on a simplified model. The configuration of collimators in the vertical irradiation port of PMRC is depicted in Fig.1. Assuming that the protons propagate as a corn shaped beam from the scatterer with the two-dimensional Gaussian distribution in the plane perpendicular to the beam axis, it is estimated that ten per cent of protons incident on the beam port are interrupted by the Upper Collimator (UC), fifty per cent by the Lower Collimator (LC), and at least thirty per cent by the Block Collimator (BC). According to the typical settings of irradiation control devices for liver tumor (a common cancer in Japan) treatment, kinetic energy of protons are approximately 180 MeV at UC and 135 MeV (in average) at LC and BC. The data of IAEA[3] (Fig.2) shows that protons of 180 MeV and of 135 MeV stopped in a copper target generate about 0.6 and 0.3 neutrons per proton, respectively.

The fluence of proton beam at this treatment is evaluated as follows. The spectrum of proton beam at the frontal edge of the spread-out Bragg peak (SOBP), $\Phi(E_i)/\Phi$, can be derived form the design shape of the ridge filter. With thus-calculated spectrum, the absorbed dose to water in the region of SOBP by unit fluence of protons, D/Φ, is estimated as a weighted average of mass collision stopping power of water (Fig.3). The value of D/Φ being evaluated as 5.9 MeV/(g/cm^2), the proton fluence corresponding to the absorbed dose of 1 Gy at the target volume in this typical situation is estimated as 1.1×10^9 cm^{-2}.

The calculated value of conversion coefficient from neutron fluence to dose equivalent at 10 mm depth in the tissue equivalent slab[4] varies only 10 % in the energy region between 2 MeV and 100 MeV. So, the value at 10 MeV, 7.2×10^{-10} Sv$\cdot$cm^2 (factor 2 is multiplied following the 1985 Statement of ICRP), is used in this estimation.

The subsidiary neutron dose of 0.01 Sv per 1 Gy of proton dose at the tumor region is obtained in this estimation. The results are summarized in Table 1.

MONITORING OF SUBSIDIARY NEUTRON DOSE BY SODIUM ACTIVATION

The time structure of the proton beam at PMRC being strongly bunched (FWHM = 50 ns), conventional methods of neutron dosimetry are not applicable. Therefore, measurement of activated urinary sodium of the patient is utilized to estimate thermal neutron fluence through the body. There are many materials, *e.g.*, ^{109}Ag and ^{195}Au, being used for activation analysis, but of these materials only sodium is suitable for our purpose. Because, the quantity that is important in

clinical decision making is not a neutron fluence at some specified point on the patient, but the averaged fluence throughout the body. Sodium, the ninth most abundant element in the human body (0.08 %), and having large capture cross section (474 mb) is the most adequate material for this purpose. Urinary sodium is always in equilibrium with blood serum sodium. The ratio of sodium concentration in urine and in blood serum is almost constant with a value of 0.5[5]. Thus, we can obtain average thermal neutron fluence by measuring the activation of urinary sodium.

To confirm the feasibility of this idea, sodium activation measurement was conducted with a patient who was administrated 3 Gy of proton dose to an abdominal tumor. The urine was sampled about 20 minutes after the irradiation. The radioactivity of 9.5×10^{-1} (±13 %) Bq of ^{24}Na at the time of irradiation was measured with the urine sample of 50 cm^3. Using the normal concentration of urinary sodium (7×10^{-5} mol/cm^3), average thermal neutron fluence is estimated as 3.7×10^7 cm^{-2}. Thus, the thermal neutron dose to this patient is evaluated as 0.6 mSv using the conversion coefficient for thermal neutrons given by ICRP[4].

Neutrons are thermalized in the matter surrounding the patient as well as in the body of the patient itself. A series of phantom experiments were conducted to verify which is dominant. The amount of activation of NaCl solution set inside the phantom (polyethylene, cubic) decreased when bottom part of the phantom was removed. The result shows that the thermalization in the body of the patient is dominant.

In order to estimate from the thermal neutron fluence, knowledge of the neutron spectrum is needed. The spectrum of neutrons to which the patient is exposed, however, is influenced by the configuration of irradiation control devices such as collimators, filters and degraders. As a result, the spectra are each different. However, detailed structure of each spectrum is not so important for our purpose, because it is sufficient to know the ratio of subsidiary neutron dose to the administrated proton dose, and moreover the value of dose equivalent is not so sensitively affected by the detailed structure of neutron spectrum. So, a computer simulation was conducted to obtain a neutron spectrum in the phantom with a "typical" configuration of irradiation control devices. The HERMES[6] code was used to simulate generation and propagation of high energy neutrons. To simulate transportation of fast and slow neutrons, the DOT[7] and ANISN[8] codes were employed.

A sample of geometries and spectra in the numerical analysis are shown in Figs. 4 and 5. In this case, the neutron spectrum is considered to be the "hardest", since neutrons are generated just upstream the phantom. The dose due to whole neutrons is estimated about 25 times of thermal neutron dose from this investigation.

Consequently, the subsidiary neutron dose per unit dose of administrated protons is estimated about 5 mSv/Gy.

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FIG.1 COLLIMATORS OF VIRTICAL LINE

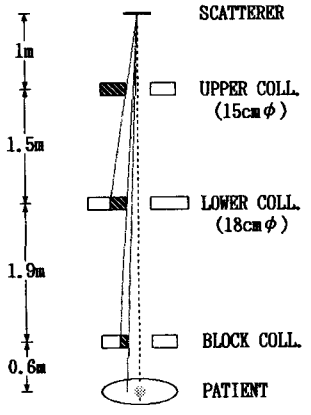


FIG. 2 NEUTRON YIELD PER PROTON (n/p)

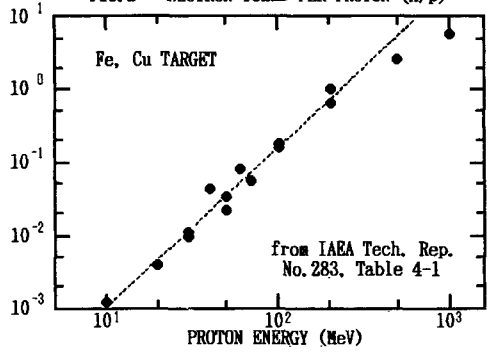
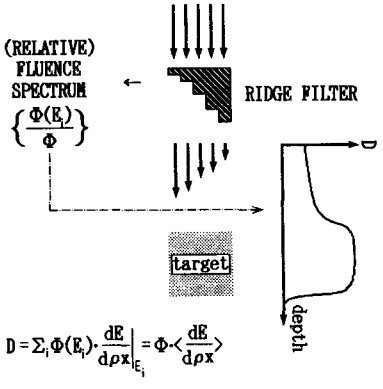


Fig. 3 EVALUATION OF PROTON FLUENCE



$$D = \Sigma_i \Phi(E_i) \cdot \left.\frac{dE}{d\rho x}\right|_{E_i} = \Phi \cdot \left\langle \frac{dE}{d\rho x} \right\rangle$$

TABLE 1 ESTIMATION OF SUBSIDIARY NEUTRON DOSE*

	UPPER COLLIMATOR	LOWER COLLIMATOR	BLOCK COLLIMATOR
PROTONS INTERRUPTED	2.2×10^{11} [p/Gy]	1.1×10^{12} [p/Gy]	6.6×10^{11} [p/Gy]
PROTON ENERGY	180 MeV	135 MeV	135 MeV
NEUTRON GENERATION	0.6 [n/p]	0.3 [n/p]	0.3 [n/p]
NEUTRON YIELD	1.3×10^{11} [n/Gy]	1.2×10^{11} [n/Gy]	1.9×10^{11} [n/Gy]
NEUTRON DOSE EQUIVALENT	1.9×10^{-4} [Sv/Gy]	1.2×10^{-3} [Sv/Gy]	1.2×10^{-2} [Sv/Gy]

* NEUTRON DOSE EQUIVALENT PER UNIT TARGET PROTON DOSE

FIG. 4 GEOMETRYOFSIMULATION

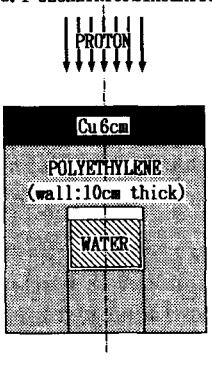
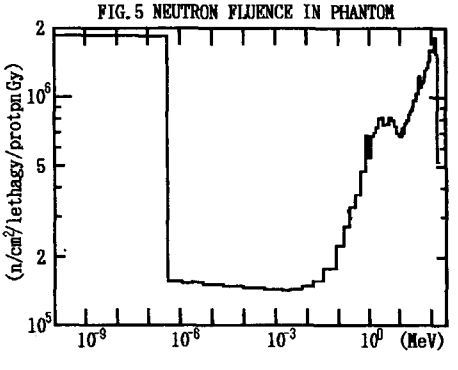


FIG. 5 NEUTRON FLUENCE IN PHANTOM



DOSIMETRY OF PROTON BEAMS AT THE MEDICAL FACILITY OF THE JINR PHASOTRON IN DUBNA.

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Radiation therapy with a proton beam has a number of important advantages over conventional radiation therapy. The proton beam allows the maximum dose to be confined to the treatment volume while the dose to surrounding normal tissues is minimized. Realization of these advantages requires higher precision of the proton beam dosimetry.

For absorbed dose rate measurements of therapeutic proton beams we use the KD-27012 clinical dosimeters with thimble air–filled ionization chambers VAK-251 (volume 50 mm^3) and VAK-253 (volume 1.5 cm^3) with air–equivalent walls. The dosimeter calibration was made with the ^{60}Co source of the therapeutic γ-unit, placed in one of the cabins of the clinico–physical facility in accordance with the recommendations of the "Code of practice for clinical proton dosimetry"[1]. The ^{60}Co source was calibrated against the primary standard of the Prague Institute of Radiation Dosimetry with the accuracy of 1.3% [10]. Using the γ–unit as a calibrated stand for ionization chambers of our clinical dosimeters was described in [3,4].

The absorbed dose calibration factor for the proton beams A_{cal} may be obtained from the calibration of the ionization chamber in the ^{60}Co source using the relations:

$$A_{cal} = N_k * C_p = N_x * (W_{air}/e)_\gamma/(1-g) * C_p \ ,$$

where N_k (Gy/reading) is the air kerma calibration factor and N_x (R/reading) is the exposure calibration factor in the ^{60}Co beam.

The proton conversion factor C_p may be obtained using the formulae from [1]:

$$C_p = A_{wall} * [(\bar{S}/\rho)^{tissue}_{air}]_p * k \ ,$$

where

$$k = (1-g) * \frac{(W_{air}/e)_p}{(W_{air}/e)_\gamma} * \frac{[(\bar{\mu}_{en}/\rho)^{wall}_{air}]_\gamma}{[(\bar{L}/\rho)^{wall}_{air}]_\gamma}.$$

The following coefficients were used for our dosimeter calibration:

- A_{wall} is the wall perturbation factor that takes into account absorption and scattering of γ-rays produced in the ionization chamber walls. The value $A_{wall} = 0.99 \pm 0.01$ is in accordance with these values for all chambers with similar geometrical dimensions.

- $[(\bar{S}/\rho)^{tissue}_{air}]_p$ is the ratio of the mass stopping powers of tissue to air for the proton beam - may be obtained from [5,6]. The "Supplement to the Code of Practice for Clinical Proton Dosimetry" [2] incorporates the new stopping power data from [6]. Differences in the ratios calculated from these two compilations are less than 2%. For proton energies 50 - 1000 MeV this ratio varies very slowly and

in [3] it was used as a constant value 1.136 ± 0.016 (1.4%). Table 1 also presents A_{cal} calculated for proton energies 10 MeV and 200 MeV. As we can see A_{cal} very slightly depends on the energy of protons;

- (1 - g) - correction to bremsstrahlung in the air in the ^{60}Co beam (g = 0.003) [1];
- $(W_{air}/e)_p = 35.2eV \quad (4.0\%) \qquad (W_{air}/e)_\gamma = 33.97eV \quad (0.2\%)$ - the average energy required to produce an ion pair in dry air for γ-rays and protons [1]. These values are recommended in [1,2] for all proton energies. But this value of $(W_{air}/e)_p$ is based on measurements at a proton energy of 1.8 MeV [7]. Recently many investigations of the energy required to produce an ion pair for heavy charged particles and in particular a correlation between $(W_{air}/e)_p$ and the LET of particles were performed. For example, in the paper [9] values of $(W_{air}/e)_p/(W_{air}/e)_\gamma$ for different heavy charged particles with different energies from the published data were plotted versus the unrestricted LET. This plot shows that for protons with energy higher than 10 MeV $(W_{air}/e)_p/(W_{air}/e)_\gamma$ is equal to 1.0 and increases with increasing LET (or decreasing proton energy). This value is in agreement with the $(W_{air}/e)_p$ obtained in [8]. This plot also confirms the value $(W_{air}/e)_p/(W_{air}/e)_\gamma$ measured at a 1.8 Mev in [7] and recommended in [1,2]; The $(W_{air}/e)_p$ value of 34.2 ± 0.5 for the dry air found in latest measurements [11] by using a water calorimeter to calibrate an ionization chamber in proton and ^{60}Co beams also confirmed this plot. This value undoubtedly has to be recommended for determination of the dose in therapeutic proton beams.
- $[(\bar{\mu}_{en}/\rho)^{wall}_{air}]_\gamma = 1.0227$ (1%) - the ratio between the mean mass energy absorption coefficients of the chamber wall material and the air for γ-rays for our ionization chambers [4];
- $[(\bar{L}/\rho)^{wall}_{air}]_\gamma = 1.0076$ (1%) - the ratio between the mean restricted collision mass stopping powers of the chamber wall material and the air for γ-rays [4].

The results of the ionization chamber calibration with different parameters are presented in Table 1.

The full estimated uncertainty of the A_{cal} value with $(W_{air}/e)_p = 1.007$ from [11] is about 3% (one standard deviation).

CONCLUSIONS

- The method for determination of the dose rate absorbed by tissues for the JINR medical proton beam on the basis of clinical dosimeter calibration with the ^{60}Co γ–source is described. The results of the ionization chamber calibration using different parameters are presented. The basic parameters used for calculations were taken from various papers.
- The energy dependence of the calibrated factor is very slight and influence of the proton beam energy distribution is negligible.

Table 1. Calculation of the absorbed dose rate for the therapeutic proton beam at the JINR phasotron in Dubna

A_{wall}	0.99±0.01 [3]				
(1 - g)	0.997 [1]				
$[(\bar{\mu}_{en}/\rho)^{wall}_{air}]_{\gamma}$	1.0227 (1%) [4]				
$[(\bar{L}/\rho)^{wall}_{air}]_{\gamma}$	1.0076 (1%) [4]				
$(W_{air}/e)_p, eV$	35.2 (4 %) [1]			34.1 (2 %) [8]	34.2 ±0.5[11]
$(W_{air}/e)_p/(W_{air}/e)_{\gamma}$	1.0362 [1]			1.01 [8]	1.007 [11]
Energy,MeV	50 – 1000 MeV	200 MeV	10 Mev	200 MeV	250 MeV
$[(\bar{S}/\rho)^{tissue}_{air}]_p$	1.136 [5]	1.130 [6]	1.139 [6]	1.130 [6]	1.130 [6]
K	1.049	1.049	1.052	1.022	1.019
C_p	1.18	1.1735	1.183	1.1433	1.1396
A_{cal}	1.031	1.025	1.034	0.999	0.996

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PAPER TITLE Dosimetrical certification apparatus for brachytherapy as the means of overirradiation protection.

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MAJOR SCIENTIFIC TOPIC NUMBER 6.3

ABSTRACT

The main reason of overirradiation in brachytherapy is an incorrect dose definition in support point on the distance of 20 mm from the source centre. Small geometry change source situation, inexactitude its activity definition can lead to considerable errors because of great dose change near by the source.

Removal these mistakes can be reached by the perfect means of certification. For this purpose the special device is suggested. It allows to define support point dose with the mistake no more than +5% and answers dosimetry and radiobiology requirements.

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PAPER TITLE Brachytherapy Dose Inhomogeneity Correction Using I-125 Seeds in the Management of Carcinoma of the Nasopharynx

AUTHOR(S) NAME(S) Kevin Amir Sadeghi, William Bice, and Bradley R. Prestidge

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) Radiotherapy

ABSTRACT (See instructions overleaf)

The influence of bone inhomogeneity in brachytherapy dose distribution is commonly ignored in dose calculation for treatment of carcinoma of the nasopharynx (NPC). This results in an uncertainty in dose determination for the tumor and critical organs near bony structures such as the nasopharynx. The influence of bone inhomogeneity in dose distribution from an ^{125}I source model 6702 was measured. For comparison, a similar study was conducted using ^{192}Ir. another commonly used isotope in NPC. The measurements were performed using LiF thermoluminescent dosimetry (TLD) chips and ion chamber dosimetry. Special phantoms were designed to hold the sources and TLD chips. These phantoms consisted of solid water for homogenous phantoms or a combination of solid water and solid water and solid bone in different configurations to create inhomogeneous phantoms. A human skull was also used to determine the dose to the critical tissues around the nasopharynx from both ^{125}I and ^{192}Ir sources. For ^{125}I a decrease in the dose delivered to the proximal side of the bone of as much as 25%. On the digital side of 1.5 cm of bone, the decrease was as much as 70% as compared to the homogeneous phantom. No significant dose reduction due to the presence of the bone was noted for ^{192}Ir.

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PAPER TITLE brain injury in leukemia patients

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MAJOR SCIENTIFIC TOPIC NUMBER 2.2. (see page 7)

ABSTRACT (See instructions overleaf)

Brain injury in leukemia patients.

Todua F. Beraia M.

Purpose:Leukemia, a malignancy of the bone marrow, is the most common form of childhood cancer. Leukemia is always disseminated at the diagnosis and can often involve nervous sistem. Cases of neuroleikemia in infants without specific treatment is 65%.

Methods and Materials: We studied CT findings of brain in 85 cases of infant neuroleukemia. 60- limphoid form and 25- mieloid. Patients were treated by scheme BFM-90 in 12 cases treatment was associated with brain radiotheraphy - 25 Gray.

Results: All patients with mieloid form lneuroleukemia demonstrated typical findings of hydrocephalia with normal intracranial pressure, ventricular and sulcar dilatation, brain atrophy. The changes were more significant in radiotheraphy group. In not treated patients brain atrophy was regionally-mostly in frontal region and were diffuse after chemotheraphy.

Conclusion: In leukemia patients with clinically normal nervous sistem was revealed brain injury in 10% of cases and mostly after radiotherafhy. CT imaging findings are not always specific for patologic conditions and must be interpreted in conjuction with chemical and laboratory date.

NEW RESULTS ON SHORT-TERM ANNEALING FOR CLINICAL TL DOSIMETRY

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ABSTRACT

Traditional annealing procedures for lithium fluoride are time-consuming and last between 6 to 24 hours; they allow one read-out cycle a day only. This represents problems especially in clinical routine if the number of detectors is limited. A short-term annealing procedure was developed in 1971 for the use in radiation protection with doses below 0.1 Gy [3]. This method enables the user to perform a large number of read-out cycles a day. The present study shows a modified short-term annealing cycle that can also be used in radiation therapy for dose levels between 0.5 and 6 Gy. The procedure requires about 1 h annealing time in total, including heating-up, 15 min at 400°C and an oven specific cooling-down profile. It can be applied without any drawback in precision. No further temperature treatment like e.g. thermal stabilization, was used in the present investigations. The low temperature peaks have been eliminated by using a self-developed computerized glow curve analysis software. A reproducibility test with 10 measuring cycles yielded precision of 0.2-0.4 % (1 s.d.) for individual LiF:Mg,Ti (TLD-100) chips. Further advantages of the described annealing procedure are an increase of response and a reduction of supralinearity.

INTRODUCTION

Precision in dose assessment with LiF:Mg,Ti (TLD-100) detectors suffers from the change of sensitivity caused by the required temperature treatments. Detector annealing subsequent to the readout cycle (e.g. pre-irradiation annealing) should minimize the change. Traditional annealing procedures with a duration of 6 to 24 hours [1, 2] allow only one read-out cycle a day. In 1971 the procedure of short-term annealing was developed for the use of thermoluminescence detectors in radiation protection for doses below 0.1 Gy [3]. The proposed procedure of pre-irradiation annealing lasts 5 min at 400°C, while pre-readout annealing consists of 5 min at 100°C to thermally stabilize the glow-curve. This method was modified here for doses up to 6 Gy as used in radiotherapy, with the pre-irradiation annealing time at 400°C increased to 15 min. The pre-readout annealing is substituted by a numerical stabilization procedure using a GSF developed computerized glow curve deconvolution (CGCD) software, called TLD_GLOW [4]. The procedure enables the user to perform a number of readings a day. The present study proves that short-term annealing after only slight modifications can also be used in radiation therapy up to at least 6 Gy.

MATERIAL AND METHOD

The investigations have been carried out with LiF (TLD-100) chips of the dimensions 1/8" x 1/8"x 0.035". Similar investigations were performed earlier with LiF (TLD-100) rods of 1 mm in diameter and 6 mm length [5].

Three series of tests with (a) long-term conventional annealing, (b) short-term annealing with low heating-up and cooling-down rates and (c) short-term annealing with fast heating-up and cooling-down rates were carried out at GSF. Each test sequence contains the evaluation of calibration factors, the determination of a supralinearity correction function and a blind test. All necessary calculations were performed using the GSF-developed software TLD_CALC for precise TL dosimetry [6].

The thermal treatments, i.e. pre-irradiation annealing for series (a) and (b) were performed in a

volume-type oven TLDO manufactured by PTW, Freiburg, Germany. The pre-irradiation annealing for serie (c) was carried out with a disc-type oven, developed at GSF [3, 7].

The time-temperature profile for TLD-100 detectors, as used in most hospitals, consists of 1 h at 400°C followed by 2 h at 100°C. The whole process lasts about 4 h and was used for serie (a) experiments. The short-term annealing procedures as used in the present investigations lasts about 1 h for serie (b) and about 30 min for serie (c) including 15 min at 400°C as well as heating-up and cooling-down periods. For the serie (c) experiment the detectors were placed manually on an aluminum plate and further on a disc pre-heated to 400°C. After 15 min the detectors were removed, also manually and placed on a cooling element of room temperature. The cooling-down period to about 50°C takes about 15 min (figure 1).

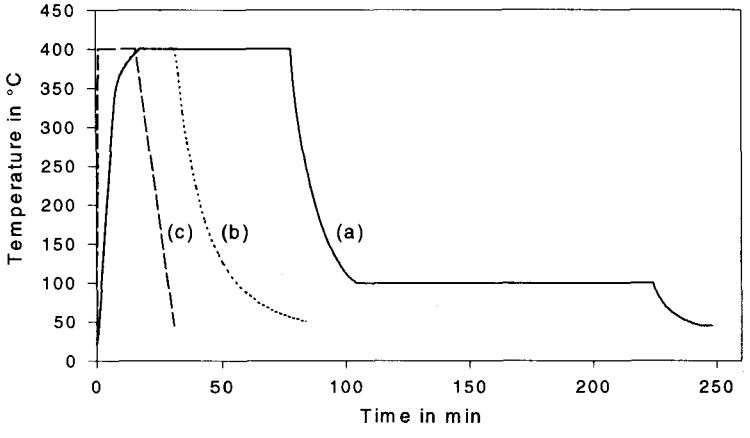

Figure 1. Comparison of duration of the three annealing cycles used for the present investigations: (a) Conventional annealing as used at most hospitals, (b) short-term annealing with low heating-up and cooling-down rates and (c) schematic view of the short-term annealing with fast heating-up and cooling-down rates.

For detectors with doses higher than 3 Gy the short-term annealing does not provide to a complete regeneration. In this case four annealing cycles have to be performed to get back the original calibration factors.

The TL read-out equipment was an automatic turn-table nitrogen hot-gas-reader, type H5500, manufactured by Harshaw/Bicron, Cleveland, USA, with a PC controlled linear heating profile. It starts at 120°C and reaches 320°C at a heating rate of 10°C/s. Before detector change the H5500 cools down to 50°C. Linearity of the heating profile is essential, because the obtained glow curves were supposed to be processed with the CGCD software TLD_GLOW, to eliminate the low temperature glow peaks without using a thermal stabilization [4]. For dosimetry the current integrals of the peaks 4 and 5 were used.

All irradiations were carried out at a ^{137}Cs radiation source in the Secondary Standard Dosimetry Laboratory (SSDL) of GSF using a Buchler Calibrator, type OB20, manufactured by Amersham/Buchler [8]. The TL detectors were exposed in 1 m source distance under a 3 mm thick build-up layer of acrylic glass to provide charged particle equilibrium. The measuring quantity was absorbed dose to water.

RESULTS AND DISCUSSION

With the above described methods and procedures the following results were achieved:

(1) Coefficients of variation of about 0.3% for individual calibration factors were obtained at all three test series for 10 calibration cycles each.

(2) Detector responses increased by 17% for serie (b) and 31% for serie (c) experiments if compared with conventional annealing in test serie (a).

(3) Supralinearity turned out to be exactly the same for all three test series - in opposite to the earlier reported investigations [5]. There might be two reasons for that: (a) In the present investigations 'chips' have been used instead of 'rods'. Due to differences in batch production both may show different supralinear behaviour. (b) The changes in the supralinear behaviour between conventional and short-term annealing are due to peaks above peak 5. These peaks have been disregarded in this study by using the CGCD software. By contrast, the investigations in [5] used thermal stabilization, a Harshaw ATLAS reader with continuous 350°C gas flow and a region of interest exceeding peak 5. We assume that argument (b) is more probable, but further investigations are necessary to clarify this discrepancy.

(4) Three blind tests for the test series (a), (b) and (c) showed a total statistical uncertainty in dose assessment better than 0.8%. The complete blind test results are given in table 1.

Table 1. Results of three blind tests with 10 detectors for each dose as obtained with (a) conventional, (b) short-term annealing with a low heating-up and cooling-down rate and (c) short-term annealing with a fast heating up and cooling-down rate. Unknown doses by ^{137}Cs gamma radiation (CPE) applied at the SSDL of GSF.

Series	Unknown dose	Evaluated dose ± 1 s.d.	D(evaluated)/D(unknown)
(a)	0.700 Gy	0.698 Gy ± 0.003	0.997
(a)	2.600 Gy	2.622 Gy ± 0.016	1.008
(b)	0.800 Gy	0.797 Gy ± 0.005	0.996
(b)	2.400 Gy	2.401 Gy ± 0.010	1.000
(b)	3.600 Gy	3.627 Gy ± 0.020	1.007
(c)	0.600 Gy	0.603 Gy ± 0.005	1.005
(c)	2.300 Gy	2.316 Gy ± 0.015	1.007
(c)	3.100 Gy	3.123 Gy ± 0.023	1.007

CONCLUSION

The modified short-term annealing procedure for therapy-level doses has been applied at GSF for research purposes since 1992 without any problems. Dosimetric advantages combined with considerable time saving in all calibration and measurement cycles as well as successful tests in clinical routine [5] revealed, that short-term annealing is superior to conventional annealing procedures.

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CANCER INCIDENCE AFTER RADIOTHERAPY FOR SKIN HAEMANGIOMADURING INFANCY

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INTRODUCTION

Infants who were irradiated because of cutaneous haemangiomas have been followed up through registers for up to 55 years after the treatment and incident cancers have been recorded. The aim of the study was to describe the cohort, the treatment techniques and dosimetry and present the incidence of new cancers in comparison with an age and sex matched Swedish population.

MATERIAL AND METHODS

Between 1930 and 1995, 12 055 infants (8 028 girls, 4027 boys) were treated with Radionuclides for haemangioma of the skin at the Radiotherapy Department, SahlgrenskaUniversity Hospital, Göteborg, Sweden according to the treatment registers.

Altogether 14 954 haemangiomas were treated in 12 055 infants. The most common location was in the head and neck region, together representing 37 % of all treated haemangiomas. The median age of the children at the first treatment was 5 months. Eighty–eight per cent were treated before 12 month of age and only 4.5% had their first treatment after 18 month of age.

In all 11 867 patients were traced and only 100 could not be identified. Sixty individuals were either dead (n = 35) or know to have emigrated (n = 25) before 1958. The remaining 11 807 persons contributed 370 517 persons years at risk (123 977) males, 246 540 females) during the years from 1958 to 1989, at which period the cohort was followed up in the Swedish Cancer Register for incident cancer cases.

STATISTICAL METHODS

The expected numbers of cancers for all sites of this population were calculated on the basis of age, sex and year specific cancer incidence rates for the West of Sweden Health Care Region.

The standardized incidence ratio (SIR) was calculated as the ratio between observed and expected numbers of cancer. Confidence intervals (CIs) were estimated assuming a Poisson distribution of the cases. The excess relative risk (ERR) was calculated as the ratio between the excess and the expected numbers of cancer. The excess number was defined as the difference between observed and expected numbers. The ERR per Gy was derived from the ERR divided by the mean absorbed dose in Gy, and the excess absolute risk (EAR) was estimated as the excess number of cancer patients per person–year (PY) and per Gy (3).

DOSIMETRY

The methods and procedures for the estimation of the cumulative absorbed dose in the different risk organs for the whole cohort have been described previously (1).
The location of the haemangiomas was coded in different regions over the surface of the body. A phantom of the size of a 5–6 monthold child was constructed and 11 risk organs were specified. The absorbed dose to these organs per 100 mg of ^{226}Ra in the source of application (mgh ^{226}Ra) was calculated for all the regions separately using a computed dose–planning
system.

We only included the gamma irradiation in the dose response evaluation since it contributes to the main part of the risk organ doses. The contribution to the doses from the β–irradiation was not included.

RESULTS

In the cohort o1 11 807 children, 248 cancers were found in 225 individuals against 204 expected. The standardized incidence ratio (SIR) was 1.21 (95% CI 1.06–1.37). The exclusion of the first 5 years after exposure did not change the SIR (1.22, CI 1.07–1.38). No overall difference was observed between the sexes. In males, 62 cancers were found (SIR 1.25, 95% CI 1.03–1.38).

Table 1 shows the number of cancers of the different organ systems and the relative rates in comparison with an age–, sex– and time–matched population. The data are further divided according to the type of radium therapy, γ–ray treatment(329 771 person years) or β–ray treatment (40 746 person years). There was a significantly increased number of cancers only among those who had been treated with γ–rays (SIR 1.26, 95% CI 1.09–1.42) but not among those treated with β–rays only (SIR 0.90, 95% CI 0.57–1.37).

Table 1

Numbers of cancers according to site and the standardized incidence ratio (SIR) is presented. Numbers are given for the total material and according to treatment technique. Assuming a Poisson distribution, 95% confidence intervals (CI) are given for the total material

Site (ICD-7)	^{226}Ra γ		^{226}Ra β		Total		
	Number of cancers	SIR	Number of cancers	SIR	Number of cancers	SIR	95% CI
Oral, digestive organs (140–158)	20	1.12	0	–	20	0.98	(0.60–1.51)
Respiratory organs (160–164)	5	0.96	0	–	5	0.83	(0.27–1.94)
Breast (170)	41	1.15	3	0.62	44	1.09	(0.79–1.47)
Genital org., female (171–176)	24	0.86	3	0.80	27	0.85	(0.56–1.24)
Genital org., male (177–179)	5	0.92	0	–	5	0.81	(0.26–1.88)
Kidney, urinary bladder (180–181)	6	1.02	0	–	6	0.88	(0.32–1.93)
Malignant melanoma (190)	25	1.31	1	0.67	26	1.21	(0.78–1.76)
Skin non-melanoma (191)	5	2.5	1	3.44	6	2.62	(0.96–5.72)
CNS (193)	29	1.78*	5	2.41	34	1.85*	(1.28–2.59)
Thyroid (194)	13	1.83*	2	2.20	15	1.88*	(1.05–3.09)
Other endocrine glands (195)	23	2.92*	0	–	23	2.58*	(1.64–3.87)
Bone (196)	2	0.72	0	–	2	0.65	(0.06–2.32)
Soft tissues (197)	0	–	2	5.71	2	0.65	(0.06–2.33)
Lymphoma (200–201)	14	1.19	2	1.33	16	1.21	(0.69–1.96)
Leukaemia (204–207)	11	1.47	2	2.04	13	1.54	(0.82–2.63)
Miscellaneous (192,199,202,203,208,209)	3	0.61	1	1.42	4	0.71	(0.20–1.81)
All sites	226	1.26*	22	0.90	248	1.21*	(1.06–1.37)

A significant difference from 1.0 ($p < 0.05$) is indicated (*).

Significantly increased numbers of tumours were found in the central nervous system (ICD–7 193), 34 cases, SIR 1.85 (95% CI 1.28–2.59), the thyroid (ICD–7 194), 15 cases, SIR 1.88 (95% CI 1.05–3.09) and other endocrine glands (ICD–7 195), 23 cases, SIR 2.58 (95% CI 1.64 – 3.87). Of the 15 thyroid cancers 14 were reported as papillary cancers and one as a medullary cancer. There were 23 tumours in other endocrine glands (ICD–7 195): one inthe adrenals, 13 in parathyroid glands (SIR 3.03, 95% CI 1.61–5.17), 5 in the pituitary (SIR 1.96. 95% CI 0.63–4.59), 3 in the thymus and one was an insulinoma.Two of the parathyroid tumours were detected at the time of treatment for thyroid cancer. The excess relative risk (ERR) was 7.5 per Gy (95% CI 0.4–18.1) for thyroid cancer, and the excess absolute risk (EAR) 1.6 per 10^4 PY and Gy (95% CI 0.092–3.9). For cerebral tumours EAR was 10.9 per Gy (95% CI 3.6–20.5) and the EAR was 5.4 per 10^4 PY and Gy (95% CI 1.8–10.1).

DISCUSSION

We have used a cohort of nearly 12 000 children treated with ^{226}Ra for haemangioma to investigate carcinogenic effects of ionizing radiation in infancy. In this cohort, there was a slight but significant excess of neoplasms (SIR = 1.21, 95% CI 1.06–1.37).

We only found a significant excess of tumours in those who were treated before 1950. This was probably due to their longer follow-up and higher age compared to those who were treated in 1950 or later.

For individual sites, only the rates of cancers of the CNS, the thyroid and other endocrine glands were significantly higher than expected. Neoplasms of other endocrine glands were dominated by parathyroid and pituitary tumours, adding to the excess of intracranial and neck tumours.

A similar overall excess of neoplasm was found in patients treated in much the same way in Stockholm (4). As in our study, thyroid neoplasm's were also in excess in their study. The ERR per Gy for thyroid cancer in the Stockholm study (4) was 4.9 compared with 7.5 (95% CI0.4-18.1) in our cohort. The estimates from the two Swedish cohorts should be compared with the Rochester study (5) where the ERR was 9 per Gy infants irradiated for thymic enlargement. For brain tumours our cohort showed a significant excess but the Stockholm cohort had no excess at all. The mean absorbed dose in the CSN was almost the same for the Stockholm (6) and Gothenburg cohorts, 0.07 Gy and 0.08 Gy repectively. The EAR for brain tumours in our study, 5.4 per PY and Gy (95% CI 1.8-10.1) should be compared with the Israeli tinea capitis study (7) with an EAR of 0.6 per 10^4 PY and Gy. Except for chance, there is no obvious explanation for the difference in relative rates of CNS neoplasm's between the Gotenburg and the Stockholm study and why our cohort has a much higher EAR for brain tumours than the Israeli study.

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1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE APPLICATION OF PROMPT GAMMA-RAY SPECTROMETRY FOR REGISTRATION, SPECTROMETRY AND DOSIMETRY OF NEUTRONS IN NCT INVESTIGATION

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

IR-8 reactor of the Russian Research Center (RRC) Kurchatov Institute was used for neutron capture therapy (NCT) investigations with experimental animals. Besides that, new methods and devices for operative neutron fluxes control, on line prompt γ -ray neutron dosimetry for operative control NCT and tailoring neutron beam for NCT were worked out. The following major methodic results are obtained:

1. Methods and devices for neutron beam control.

The devices are suggested for operative control of neutron fluxes, their spectral and dose characteristics with the energy range from 10 neV to 10 MeV, in which spectrometry of prompt γ - rays produced in the interaction of neutrons with the multilayered targets, is used.

2. Method on line neutron dosimetry.

Also a method for in situ neutron dosimetry on line by means of prompt γ -ray spectrometry is offered. This method is based on the determination of nuclear reactions quantity in the irradiated objects by the registered intensity of γ -rays. The energy of each reaction product, absorbed in the object is computed with the help of nuclear and atomic data tables.

3. Thermal and epithermal neutron beam tailoring method.

The hydrogenous scatterer of small thickness is suggested for tailoring of intensive therapeutic neutron beams with small contribution of fast neutrons (1%) and γ -rays. The scatterer is disposed in tangential experimental channel of reactor. Decreasing of fast neutron contribution and consequently fast neutrons KERMA reducing of the neutron beam are achieved due to peculiar energy dependence of neutron scattering cross section on hydrogen nuclei.

ASSESSMENT OF EFFECTIVE DOSE AND UTERUS DOSE FOR WORKERS IN MEDICAL RADIOLOGY

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INTRODUCTION

Using model calculations for x-ray shielding and dosimetry, we analysed the protective effect of lead aprons on the effective dose and organ doses to workers in medical radiology. We investigated the relationship between effective dose and personal dose measurements. Our study enables the choice of appropriate correction factors for converting depth dose measurements into effective dose, for typical exposure situations in medical radiology. In consideration of protection of the conceptus, we give guidance on the control of the equivalent dose to the uterus.

Estimation of organ doses and the effective dose from scattered x-rays is complex because the depth dose distribution in the body is highly energy dependent. In practical radiology it is further complicated by partial body shielding with protective aprons and from variations in exposure orientation.

We developed a computer model to calculate equivalent organ doses as a function of x-ray energy spectrum, shielding parameters and exposure geometry. Our model refers to whole body exposure by broad parallel beams, in simulated exposure geometries that represent typical working conditions in medical radiology. All equivalent organ doses are calculated per unit kerma-in-air at the surface of the trunk. We adopted the appropriate organ dose conversion factors derived from Monte Carlo calculations in mathematical phantoms (4). The effective dose is calculated as the weighted sum over relevant organs with their assigned weighting factors, according to the definition by ICRP (5).

We analysed the reduction of the effective dose from use of lead aprons of different model, fit and lead thickness, all in function of tube potential and variations in exposure geometry as they occur in actual practice. The protection efficiency is strongly dependent on model and fit in combination with exposure geometry. After all, unshielded and partially unshielded organs have a dominating influence on the effective dose.

We give conversion factors for assessment of the effective dose from personal depth dose measurements. The calculations of the badge depth dose in personal monitoring refer to the depth dose equivalent at 10 mm tissue in AP-orientation (6). Our study enables the choice of an appropriate division factor (shortly called: divider) for correction of unshielded badge depth dose measurements, that always significantly overestimate the effective dose. Our divider may be applied to unshielded badge measurements "mid front" at collar or chest, outside the apron, provided that the apron suits the exposure geometry and the fit is reasonably good. We strongly advise against placing a single badge under the apron. Our calculations have clearly shown that considerable levels of effective dose cannot be assessed due to the substantial enhancement of the lower detection limit for dosimetry under the apron. For tube voltages above 100 kV, the lower detection limit is increased by a factor of 10 to 20. At lower energies, the increase of the lower detection limit is much higher. Furthermore, the degree of underestimation of the effective dose by shielded badge depth dose is highly sensitive to variations of tube voltage and lead thickness.

We analysed the protective influence of lead aprons on the uterus dose and the relationship between the uterus dose and badge depth dose measurements. We conclude that lead aprons reduce the uterus dose equivalent by over a factor of 5. Our findings show that the uterus dose can be adequately controlled by depth dose measurements with a dosemeter under the apron at the surface of the abdomen, that is especially assigned to dose control of the conceptus.

A compilation of all calculated results as reported above, is available from the authors in a database structure on a 3.5" diskette.

PROTECTION EFFICIENCY

The protective effect of aprons can be described by the protection efficiency, defined as the percentage by which the effective dose is reduced. Fig.1 summarizes the calculated protection efficiency in function of lead thickness and tube voltage. The protection efficiency for wrap-around aprons in a mixed 60% AP 30% lateral and 10% PA exposed geometry is the same as for frontal ones in predominant AP geometry with less than 20% lateral exposure. From our calculations we derived a rule of thumb that provides a reasonable estimate for the protection efficiency as a function of tube voltage [kV] in kilovolts and lead thickness [Pb] in millimetres.

$$\text{PROTECTION EFFICIENCY} \approx 85 - \frac{kV - 50}{20 \times Pb}$$

This approximation refers to lead aprons of reasonably good fit, without neck shielding. The formula is not valid for the specified lead equivalence of composite aprons. Our calculations show that well fitting frontal aprons of 0.25 mm guarantee a protection efficiency above 70%, which is adequate for working in general radiology. For tube voltages below 70 kV, a lead thickness of 0.15 mm is sufficient. For interventional radiology, which involves lateral and dorsal exposure, it is a must to use a good fit wrap-around apron. Such an apron of 0.25 mm Pb provides a dose reduction of at least 70%. For high workloads and high tube voltage, a 0.35 mm Pb apron should be considered. Further increasing lead thickness above 0.35 mm does not deliver substantial extra dose reduction to justify the extra weight of the apron. Additional neck shielding of 0.35 mm Pb however, results in a further 5% to 10% reduction of the effective dose, because of almost complete shielding of the thyroid and oesophagus. Using an apron thickness of 0.35 mm Pb in combination with neck protection of 0.25 mm brings about the maximally achievable protection efficiency of 85% to 90%.

CORRECTION FACTORS FOR PERSONAL DOSIMETRY

We have calculated the ratio between unshielded depth dose equivalent and the corresponding effective dose when using protective aprons under various conditions. Results are given in Fig.2 for wrap-around aprons in a mixed 60% AP, 30% lateral and 10% PA exposure geometry. To avoid underestimation of effective dose, we have adopted that only exposure in frontal direction contributes to the depth dose. For the same reason Fig.2 only shows lower values for the divider, which apply to poor fit. For single badge dosimetry with the badge placed outside the apron "mid front" at collar or chest, a divider of 5 gives a conservative approximation of the effective dose:

$$\text{EFFECTIVE DOSE} \leq \frac{1}{5} \times HP_{UNSHIELDED}$$

Dual badge monitoring is recommended, where the annual effective dose may amount to a substantial fraction of the occupational dose limit (7). An accurate estimate of the effective dose can be derived from the combination of the reading for the unshielded depth dose from the badge worn outside the lead apron, and the shielded depth dose recorded by the dosemeter that is worn mid-front at waist level under the apron. For dual badge monitoring we conclude a widely applicable formula to calculate the effective dose:

$$\text{EFFECTIVE DOSE} \approx \frac{1}{10} \times HP_{UNSHIELDED} + HP_{SHIELDED}$$

This empirical formula applies to aprons without neck shielding, irrespective of tube voltage and for lead thickness above 0.15 mm, if the type of apron suits the exposure geometry. For exposure conditions that occur in interventional radiology, the degree of overestimation is always less than 50%. For situations where neck shielding is used, the empirical formula for the effective dose is:

$$\text{EFFECTIVE DOSE} \approx \frac{1}{30} \times HP_{UNSHIELDED} + HP_{SHIELDED}$$

For interventional radiology, this simple approximation gives less than 30% overestimation and, again, no underestimation.

CONTROL OF UTERUS DOSE

For female workers, the dose control of the abdominal region is of special interest in radiation protection, with an eye to protection of the conceptus. We use the uterus dose equivalent as a first approximation of the dose to the foetus. The protection factor for the uterus is defined as the ratio between uterus dose without and with shielding respectively. Values for the protection factor for the uterus are given Fig.3, in function of lead thickness and tube voltage. This shows that both frontal aprons in a predominant AP orientation and wrap-around aprons in a mixed geometry guarantee a protection factor above 5, even for very high tube voltages, if lead thickness is 0.25 mm.

A special dosemeter may be worn at the surface of the abdomen under the apron, for dose control for the unborn child. We analysed the relationship between such depth dose measurements and the uterus dose. Our data demonstrate that the uterus dose is always slightly overestimated (Fig.4). In view of the uncertainties that are associated with operational dosimetry, a prudent first approach is to use the uncorrected shielded depth dose to estimate the equivalent dose to the uterus. If, for special reasons, a more precise estimate is wanted, one may divide the reading by the appropriate correction factor from Fig.4.

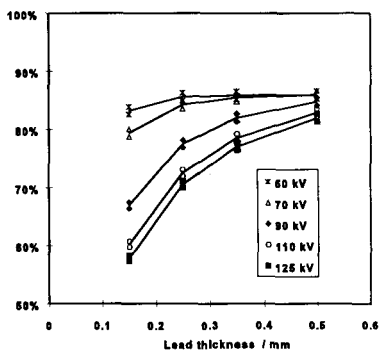

Fig. 1: Protection efficiency of lead aprons without neck shield

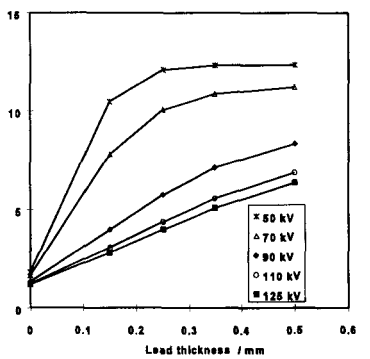

Fig. 2: Correction factor for unshielded single badge dosimetry

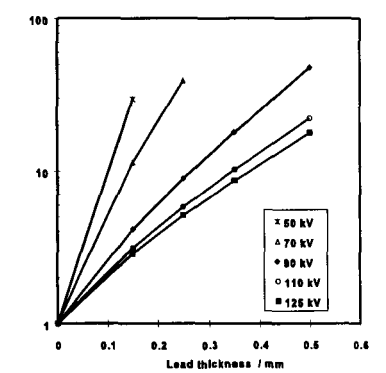

Fig. 3: Protection factor for uterus

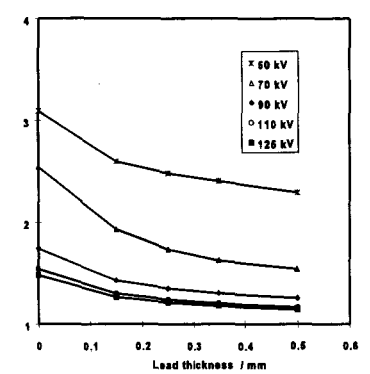

Fig. 4: Overestimation of equivalent dose to the uterus by shielded depth dose measurements

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PAPER TITLE Occupational Exposure to X - Radiation in Poland in the years 1966-1995

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MAJOR SCIENTIFIC TOPIC NUMBER 5.A... (see page 7)

ABSTRACT (See instructions overleaf)

The paper presents the dosimetric system and detail results of measurements for 1995 and overview of variation of selected doses through years 1966-95.

Actually population of people under control enumerate 26.000 employed in about 2600 X-ray departmens. Health care workers constitute about 80 % of the total monitored population. Annual dose for 98 % of people under control is below 1 mSv. A similar percentage applies to all categories of workers except those undertaking operational radiology.

The historical variation of annual dose received by the X-ray workers since 1966 is discused.

SPECTROSCOPY OF PATIENT SCATTERED X-RAYS USING GE DETECTORS

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INTRODUCTION

One of the main tasks in radiation protection is the reliable determination of risk relevant dose quantities-like organ doses, effective dose or partial body doses-for exposed persons. In cases, when a more detailed description of an exposure situation is demanded, the radiation field has to be analysed with regard to the type of radiation, the spectral energy distribution and the direction of incidence, to convert personnel dose (or any other measured dose) into a risk relevant dose descriptor. Especially in context with the increasing number of long lasting interventional examinations or treatments in diagnostic radiology this became a pressing problem in the recent past.

A catalogue of spectra was prepared (1) which contains spectra of radiation scattered by a water phantom in dependence of scatter angle and tube voltage. In addition, for rougher estimates or when measured dose values are not available, approximate air kerma values in a distance of 50 cm to the centre of the phantom are presented.

METHOD

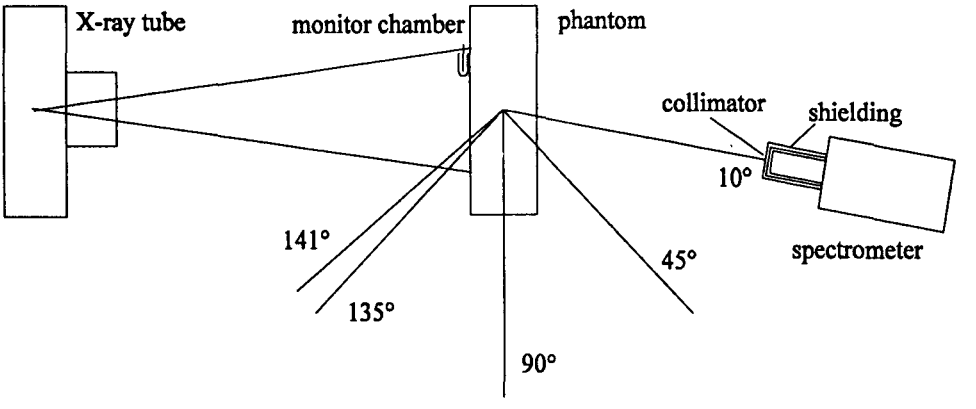

Figure 1. Experimental arrangement

A water filled perspex phantom (30 cm x 30 cm x 15 cm) was irradiated in 70 cm focus to phantom distance. Size of the central rectangular field: 16 cm x 16 cm. The measurements were performed at a diagnostic X-ray unit (Polymat 50 with a Pantex 125 X-ray tube, total filtration 3 mm Aluminium) operated in fluoroscopic mode. The spectrometer was equipped with a cylindrical (height: 50 mm; diameter: 56 mm) high purity Germanium detector (EG&G Ortec, GMX-25190 with Beryllium entrance window). To avoid distortion of the measured pulse height distributions by pile-up effect, the beam impinging onto the detector was highly collimated (beam diameter: 1.4 mm). In all other directions the detector was shielded by 3 mm copper and 12 mm lead.

Spectra were measured for the indicated directions (fig. 1) at tube voltages of 52 kV and at 60 kV to 110 kV in steps of 10 kV. Because of the small aperture of the narrow collimator it became necessary to position the spectrometer in a comparatively large distance of 212 cm to the centre of the phantom to register all the radiation scattered by the phantom into the considered direction. To achieve the desired spectra measured pulse height distributions were corrected in a stripping procedure using detector response functions calculated by Monte Carlo methods for the detector and collimator arrangement in the spectrometer (2,3). Air kerma in the phantom to collimator distance was calculated on the basis of spectral photon fluence and diameter of the collimating diaphragm.

To provide approximate values for the integral photon fluence and for air kerma in a realistic distance (50 cm) to the phantom centre dependence of air kerma from distance along the indicated directions (fig. 1) was measured by means of a 1000 cm^3 ionisation chamber (20341, PTW). Thereby a 0.2 cm^3 soft X-ray chamber (R 17927, PTW) served as monitor in the entrance plane of the phantom to compensate for fluctuations of the X-ray tube output.

RESULTS and DISCUSSION

Figure 2 shows as an example one of 35 spectra of the catalogue. The spectrum is presented in arbitrary units for energy intervals of 1 keV, normalised to unity in the peak interval. As to be expected the characteristic K-lines are shifted towards lower energies and the spectrum with a mean photon energy of 51.1 keV is in total significantly harder than the primary spectrum meeting the phantom with a mean photon energy of 43.5 keV (4).

To validate the applied methods (measurement and stripping procedure) selected primary spectra were measured in a modified experimental arrangement and compared with data from other publications (4,5). Part of the scattered spectra could also be compared with spectra calculated by Monte Carlo Methods (6). In both cases the agreement was good.

The absolute values for air kerma and integral photon fluence per unit of the product of tube current x exposure time (mAs), as presented in figure 2, depend strongly on the geometrical arrangement of X-ray tube and phantom (distance, field size), the condition of the X-ray tube and the not strict proportionality between mAs value and X-ray tube output. Consequently they can only be considered as rough approximate values.

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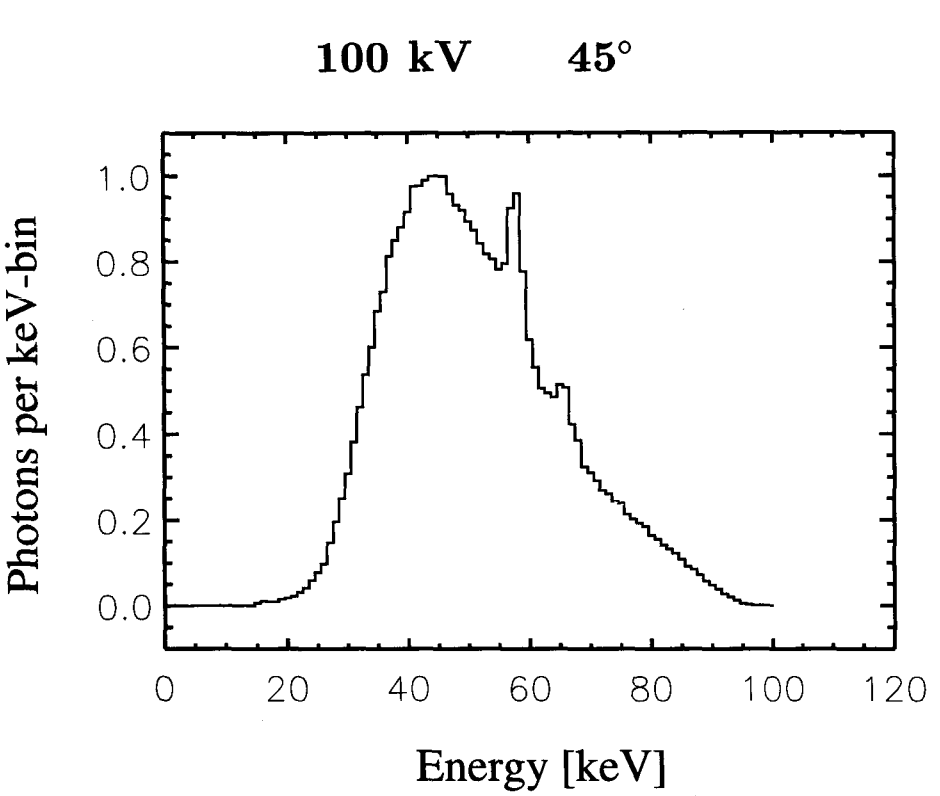

keV	1	2	3	4	5	6	7	8	9	10
0	0	0	0	0	0	0	0	0	0	0
10	0	0	0	0	0	1	0	0	1	1
20	2	3	4	5	7	9	14	19	25	30
30	38	46	53	60	68	72	81	85	88	91
40	97	97	98	99	100	99	95	93	91	89
50	87	84	81	80	78	79	92	95	77	61
60	55	50	49	48	51	50	42	38	32	31
70	29	26	26	24	23	21	20	19	18	16
80	15	14	13	12	10	9	8	7	5	4
90	3	2	2	1	0	0	0	0	0	0

Parameters during irradiation	values
Filtration	3 mm Al
Field size on phantom	16 cm · 16 cm
Mean photon energy	51.1 kV
Air kerma at 50 cm/mAs	0.158 $\frac{\mu Gy}{mAs}$
Photon fluence at 50 cm	419788 $\frac{photons}{mAs\ cm^2}$

Figure 2. Spectrum of scattered radiation for 45° scattering angle and 100 kV tube voltage.

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PAPER TITLE Estimation of doses received by interventional radiology workers in Poland

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MAJOR SCIENTIFIC TOPIC NUMBER 5.4... (see page 7)

ABSTRACT (See instructions overleaf)

The results obtained in individual dosimetry indicate that the level of exposure to X-rays in interventional radiology workers is higher than observed among other radiologists. In order to assess the actual exposure level in the personnel involved in various operations with the use of X-rays, the exposure rate on the body surface of the subjects under study was measured. These measurements were carried out in 20 hospitals during the following operations: cardiac pacemaker insertion, haemodynamics of the heart, extracorporeal lithotrypsy, transcutaneous, removal of kidney stone, radiological control of biliary routes, orthopedic surgery.

It has been found that the estimated dose equivalent depends to a large extent, on the function performed by individuals. In all hospitals the oprating physician was the most exposed person. The level of exposure depends also on the kind of the operational technique adopted, the quality and characteristics of X-ray equipment and physical parameters applied.

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PAPER TITLE The Radiation Dose Survey before and after the Shielding Improvement of the X-ray Tube Used for ERCP and Angiocardiography

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MAJOR SCIENTIFIC TOPIC NUMBER 5.4 (see page 7)

ABSTRACT (See instructions overleaf)

The Endoscopic Retrograde Cholangiopancreatography (ERCP) and Angiocardiography are standard medical diagnosis methods in these days. The commercial X-ray tube and accessories for those operations comes with typical radiation shielding for the operators i.e. doctors and nurses. This paper will present results of the survey of the operators radiation dose during the whole operation. The results shown that the hands of the doctor was completely exposure to the X-ray without any shielding while he was adjusting the table which the patient was laid upon. According to the results, we suggested some extra shielding could be added. The corresponding results after the shielding was added will be presented and discussed. We also suggested some operation procedure which could reduce some radiation dose. It will be discussed in this paper too.

SCATTERED FRACTIONS OF DOSE FROM 18 AND 25 MV X-RAY RADIOTHERAPY LINEAR ACCELERATORS

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INTRODUCTION

Over the years, measurements have been made at a few energies[1,2] to estimate the scattered fraction of dose from the patient in medical radiotherapy operations. This information has been a useful aid in the determination of shielding requirements for these facilities. With these measurements, known characteristics of photons, and various other known parameters, Monte Carlo codes are being used to calculate the scattered fractions and hence the shielding requirements for the photons of other energies commonly used in radiotherapeutic applications.

The National Institute of Standards and Technology (NIST) acquired a Sagittaire medical linear accelerator (linac) which was previously located at the Yale-New Haven Hospital. This linac provides an x-ray beam of 25 MV photons and electron beams with energies up to 32 MeV. The housing on the gantry was permanently removed from the accelerator during installation. A Varian Clinac® 1800 linear accelerator was used to produce the 18 MV photons at the Frederick Memorial Hospital Regional Cancer Therapy Center in Frederick, MD.

This paper represents a study of the photon dose scattered from a patient in typical radiation treatment situations as it relates to the dose delivered at the isocenter in water. The results of these measurements will be compared to Monte Carlo calculations. Photon spectral measurements were not made at this time. Neutron spectral measurements were made on this Sagittaire machine in its previous location[3] and that work was not repeated here, although a brief study of the neutron component of the 18 and 25 MV linacs was performed utilizing thermoluminescent dosimetry (TLD) to determine the isotropy of the neutron dose.

MATERIALS AND METHODS

A commercial drinking fountain water-bottle with a radius of 12.8 cm was used as the patient phantom for this work. A Capintec 0.076 cc chamber complete with its ^{60}Co buildup cap was inserted in a sleeve and placed at the isocenter within the phantom. The phantom was filled with water to the edge of the neck and placed at the isocenter of the accelerators (100 cm from the targets) to provide the scattered radiation. The beam for each accelerator was set to a 20 cm x 20 cm field at the isocenter.

A Victoreen model 550-3 (330cc) chamber with its ^{60}Co buildup cap was placed in front of a 26 mm sheet (25 cm x 25 cm) of polystyrene at two meters from the isocenter. An appropriate amount of buildup material was then placed in front of the chamber. The chamber was connected to a Victoreen model 500 electrometer. Approximately 30 cm in front but out of the direct path of the Victoreen chamber, a 12.7 cm diameter polyethylene moderating sphere containing TLD-600 (neutron and gamma sensitive) and TLD-700

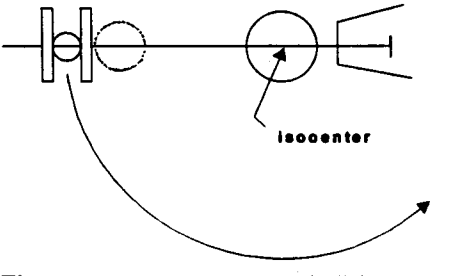

Figure 1 Experimental setup. The lighter circle indicates the location of the TLDs though they actually sat below the level of the chamber.

(gamma sensitive) chips was placed to measure the neutron component of dose equivalent throughout the angles of interest. The sphere remained in the same place for all measurements on a given accelerator, but was not necessarily in the same place from accelerator to accelerator. A top view of this setup can be seen in Figure 1.

Because there was a shift in the predominant energy of scattered photons as the chamber was moved through the angles, various thicknesses of polystyrene were placed in front of the Victoreen chamber to determine the appropriate thickness of buildup material to obtain a good photon signal reading yet exclude noise from secondary electrons. This was performed at each measurement angle. A preliminary investigation in the forward direction had shown that the effect of buildup material both in front and behind the chamber had nearly the same effect as standard build-up caps. The difference did appear to be somewhat energy dependent; however, it was consistent throughout the work on each accelerator, so was ignored.

Numerous measurements were made at angles from 0° to 140° for the 18 MeV photon beam. Because of the nature of the setup for the Sagittaire linac however, measurements were made only at a few select angles from 0° to 140° and the variation with angle was assumed to be smooth as found for the other accelerator. Repeated measurements were made on each accelerator with no phantom in the beam to check for air and wall scatter, and with the jaws in the closed position to get an assessment of head leakage contributions at each measurement location.

TLD measurements were made every 20° from 0° to 140° about the isocenter on the linear accelerators. Five chips of ^{7}LiF enriched LiF (TLD-700) and five chips of ^{6}LiF enriched LiF (TLD-600) were placed at each measurement location. Control TLDs were maintained throughout to account for any transit or extraneous exposures that may have occured. The TLDs were calibrated to ^{252}Cf for the neutron dose-equivalent calibration factor.

RESULTS AND DISCUSSION

Note that only the results from the 18 MV accelerator are presented below.

Neutron Dose Equivalent

We were interested in the neutron isotropy about the target, not the isocenter. The distance from the target along a 170 cm arc about the isocenter is:

$$R = a\sqrt{(5 + 4 \cos \varphi)}$$

where R is the distance from the target, a = 1 meter, and ϕ is the angle about the isocenter from the central beam axis. The measured dose equivalent (DE) was then corrected by $1/r^2$ to get the DE at 270 cm from the target.

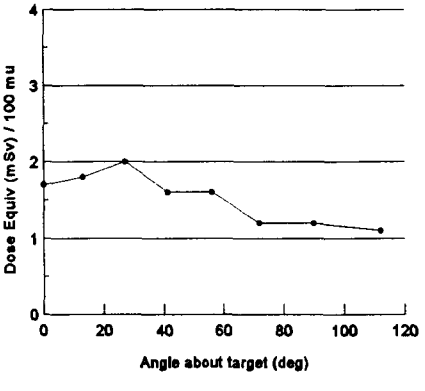

Figure 2 Neutron doses corrected to 270 cm from the target vs. angle about the target from the central beam axis

The TLDs gave expected results, namely, that the dose from neutons is relatively isotropic about the target with somewhat higher readings in the forward direction, see Figure 2. The energies of neutrons from this type of accelerator have been shown to fall predominately fall between 0.01 and 1 MeV[4], much lower than the calibration energy, so an exact determination of dose equivalent was not possible.

Photon Scattered Fractions

Figure 3 shows the scattered fractions of dose about the isocenter at 2 meters for 18 MV x-rays. The ordinate represents the ratio of the dose measured 2 m from the isocenter per unit dose at the center of the phantom. Measurements were made in the horizontal plane at angles of 10, 15, 20, 25, 30, 40, 50, 60, 75, 90, 105 and 120° from the central beam axis. The ratio is seen to drop off sharply with increasing angle which is as expected. Table 1 compares our values with those from an earlier report. Our values are smaller than those from Abrath, et al[2] by about 50% at the smaller angles to 150% at the larger angles.

Further work showed that a $1/r^2$ approximation does not hold for scattered radiation. Our measurements indicate that the ratios of dose at 1 m to that at 2 m varies from 2.5 to 3.8 and appears to be somewhat machine dependent. These measurements were made at both 30 an 90° for 6, 10 and 18 MV beams, but neither energy nor angle seemed to play a predominant role. Another finding of interest concerns the field size dependence of the scatter fraction. It is a common practice to take results from a 10 cm x 10 cm field and multiply by 4 to obtain an estimate for the result of a 20 cm x 20 cm field. Our investigation has shown that this is not the case: the result is acutally less than 4, and that the multiplicative factor is more a function of angle than of energy. Further work is being done to address this issue.

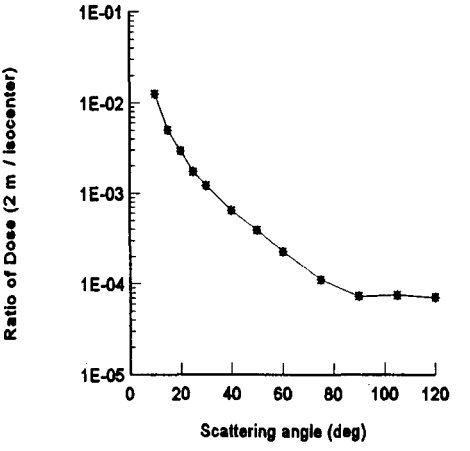

Figure 3 Ratio of radiation scattered at 2 m to that at isocenter vs. angle of scatter from central beam axis.

Differences in measurement technique lead to the inconsistencies in reported scattered fraction values found in the literature. The numerator can be affected by such variables as primary beam quality vs. scattered beam quality, distance of measurement from the isocenter, build-up thickness used, etc. The denominator can be affected by whether the value at the isocenter is recorded as measured at depth in the phantom or as computed to d_{max} for the energy of interest. Monte Carlo codes in progress should help resolve some of these issues.

Table 1 Comparison of our work with previous work. The values for Abrath *et al.* were read from their graph.

Angle	20°	30°	40°	60°	90°	120°
Our work	2.9 E-03	1.2 E-03	6.4 E-04	2.3 E-04	7.3 E-05	7.1 E-05
Abrath *et al*	5.2 E-03	2.2 E-03	9.2 E-04	3.5 E-04	1.8 E-04	1.6 E-04

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PAPER TITLE Leakage Radiation from Various Co-60 Units of Type Theratron

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MAJOR SCIENTIFIC TOPIC NUMBER 5.4 (see page 7)

ABSTRACT (See instructions overleaf)

To study the complex backgrounds of leakage-scattering-radiation of CO-60 units an area near the source loading-slot was examined, where the radiation passes through deep narrow slits producing "Hot Spots" in the thick lead-tungsten composed shielding-case of the instrument. Using a new approach, developed at the clinic in Vienna, these needle-like radiation was investigated in 3 phases: 1) photon-energy-determination of the leakage-radiation, 2) TLD-expositions to proof average absolute-doserate and 3) high- resolution-filmdensitometric determination of spikes in the 2D-dose distribution of the "Hot Spot-areas" combined with a calibration-procedure to get the correct maximum absolute doserate. Due to the results some constructive supplements had to be done to satisfy the German and Austrian regulations for radiation protection. This will be discussed in detail.

HVL and energy of the leakage radiation was determined using a calibrated ionisation-chamber. Films, exposed for a longer period, showed several well defined narrow "Hot Spots" embedded in a circular string-shaped radiation field. They were used as a positioning-aid for the arrangement of LiF-TLD-crystals. The TLDs were Co-60-calibrated with known doserate and good exposition-homogeneity. Even these results show a medium doserate caused by an integration-effect over the Hot Spot- exposed TLD-volume. A good approach to the maximum pointdoserate inside a Hot Spot was obtained by exposition of KODAK-OG-films using CAWO-U cassettes with OG8 intensifier foils. After the microdensitometric analysis of different time-exposed films the TLD-values and the optical density values were connected. The result was a calibration in relation to absolute energy-doserate.

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PAPER TITLE TYPE TESTING OF THE HARSHAW EXTREMITY AND WHOLE BODY PERSONAL MONITORING DOSEMETERS FOR PHOTON AND ANGULAR RESPONSE

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MAJOR SCIENTIFIC TOPIC NUMBER 5.4 see page 7)

ABSTRACT (See instructions overleaf)

The Harshaw 6600 personnel monitoring system can be used to measure extremity doses using ext-rads and whole body monitoring using three or four element cards. Limited performance data on these two systems is available specifically of relevance to usage in hospitals. The ext-rad comprises of a single dosemeter element, the chipstrate consists of a 3 by 3mm chip of LiF thermoluminescent material hermetically bonded to a kapton substrate to which a barcode strip has been attached. The whole body dosemeter consists of a TLD card and holder. The card consists of 3 solid LiF TL elements. The holder which is made of ABS plastic contains various filters for radiation type and energy discrimination.

Both these types of dosemeter have been type tested for their response to different photon energies ranging from 45-104keV and o°, 20°, 40° and 60° angles of incident of the photon beam. Both angular and energy response were assessed on a 30 x 30 x 15cm PMMA phantom and on a 30 x 30 x 30 cm water phantom. All irradiations were done using the ISO wide spectrum series reference radiations.

The radiation dose quantity Hp(0.07) was calculated for the extremity monitor and both Hp(0.07) and Hp(10) were calculated for the whole body card. All the results were normalised to the dosemeters response to Cs137. All measurements were compared with the true dose in the phantom using previously published conversion coefficients, with a view to assessing the performance of the dosemeters according to the recommendations of the ICRP.

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PAPER TITLE THE INTEREST OF NEW DOSEMETERS FOR OPERATIONAL DOSIMETRY IN MEDICAL FIELD

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MAJOR SCIENTIFIC TOPIC NUMBER 5.4 (see page 7)

ABSTRACT (See instructions overleaf)

The results of the individual survey of workers in medical field show that more than 99 % of the workers are exposed to levels lower than 5 mSv per year. These data, which are obtained with the legal dosemeter, a film badge, concern only the whole body exposure and more precisely the chest exposure, but they do not inform on the exposure of particular parts of the body. Moreover, with a measurement threshold equal to 0.2 mSv, low exposures are not detected with this dosemeter.

Recent improvements in detection and electronics are at the origin of new personal dosemeters well suited for operational dosimetry, especially for the most exposed workers in nuclear medicine, brachytherapy and interventional radiology.

Two dosemeters were evaluated for these medical applications : one measures whole body exposure and the other one records extremity, shoulder or forehead exposure with a remote detector. These two devices have a silicon semi-conductor detector and offer a range of capabilities : dose and dose rate measurement and readout, audible and visual programmable alarms.

Measurements were performed in various working situations thus confirming the interest of these dosemeters for a detailed analysis of working conditions, the detection of low exposures or high, localized irradiations and the training of staff. They also underline the possibility and necessity for an efficient operational dosimetry, indispensable complement to the legal survey.

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PAPER TITLE ANALYSIS OF RADIATION LEVELS FOLLOWING ADMINISTRATION OF HIGH DOSES OF RADIOIODINE IN THYROID CANCER PATIENTS

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MAJOR SCIENTIFIC TOPIC NUMBER 6.2. (see page 7)

ABSTRACT (See instructions overleaf)

The concern regarding associated safety considerations had been the issue of debate to re-examine the justification for imposing wide range of restrictions on the patients administered with large doses of radioiodine for the treatment of thyroid cancer. The present communication is therefore intended to present analysis of the radiation levels monitored in 50 thyroid cancer patients treated with high doses of radioiodine (50 - 200 mci) at our centre. The radiation levels on the surface of the thyroid, 20 cm away from thyroid surface, at lead screen, at one meter from patient and also at the entrance of the isolation room were measured. It has been observed that a patient admitted with a maximum activity of 200mCi with involvement of extra thyroidal lymph nodes take as long as 5-7 days when the criteria for discharge of patient being used as 2mR/hr at a distance of 1 meter from the patient. On the contrary the patient administered with 100mCi I-131 was found to attain the desired level within a period 3-4 days. The patients administered with smaller doses (50mCi) required the isolation of only 2-3 days. In view of the patients undergoing high degree of mental agony during isolation particularly those requiring prolonged isolation of 7-8 days, the standards of radiation levels laid down for discharge of the patient need to be reviewed and if possible appropriately changed to minimize the patient discomfort. Further analysis has revealed that this would be possible only if a method of patient counselling is evolved both prior to and after discharge of the patient.

RADIATION DOSE RATES FROM PATIENTS UNDERGOING GALLIUM-67 CITRATE STUDY

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INTRODUCTION

In 1990, ICRP widened the definition of the medical exposure[1]. When a patient is administered radiopharmaceuticals for his or her diagnosis or treatment, radiation exposure to the family attending the radioactive patients is regarded as medical exposure. Although definite dose limit to the medical exposure was not recommended by ICRP, the exposure should be kept as low as reasonably achievable. Therefore it might be necessary to set a dose constraint in nuclear medicine.

So, it is important to assess the magnitude of radiation exposure of the person who attend the patient. We have started some experimental and clinical studies on this subject. In the first stage of this studies, we chose Ga-67 citrate of a radiopharmaceutical for tumor imaging. We have performed to document the radiation exposures around the Ga-67 citrate patients and to estimate the radiation doses of the other persons who may come into close contact to them.

MATERIALS AND METHODS

In twenty three adult patients, administered Ga-67 citrate to diagnose for their examination, radiation dose rates ($\dot{X}$(mR/h)) around the patients were measured with three ionization survey meters (Aloka Model ICS-301). Records of the measurements were performed at times of 0.1 and 48.0 h after

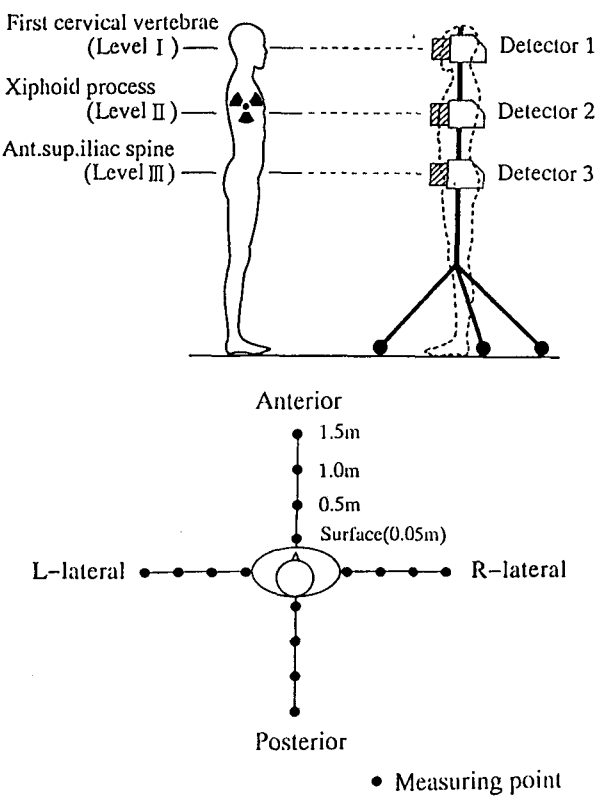

Figure 1. Detector arrangements for the dose estimation around the Ga-67 citrate patients.

injections of Ga-67 citrate.

The arrangements of these equipments were shown in Figure 1. X was normalized by the injection activity of 111 MBq and was converted into equivalent dose rate ($\dot{H}_{1cm}$ (μ Sv/h)) of 1cm in depth with a factor of 15.2 (μ Sv/mR). Biological half lives of 30 h (17%) and 613 h (83%) investigated by Evelyn E. Watson et al. [2] and physical half life (78.3 h) were used for the calculation of potential equivalent dose (H_{1cm}(μ Sv)) around the patient. The H_{1cm}s for a first week after the administrations of Ga-67 citrate were calculated from the effective decay constant (λ) of Ga-67 citrate and the initial $\dot{H}_{1cm}$s (A_0) of the measurements, using a equation:

$$H_{1cm} = A_0 (1-\exp(-\lambda \cdot t)) \lambda^{-1} \quad \cdots\cdots (1)$$

Where, the "t" is an integrating time to estimate the H_{1cm}. In figure 1, the broken line person was assumed to be a person exposed to the radiation from the Ga-67 citrate patient. To estimate the effective dose (E) of the other person who comes into close contact to the Ga-67 citrate patient, E of this broken line person at each distance from the patient was calculated by a equation:

$$E = 0.06H_a + 0.28H_b + 0.61H_c + 0.05H_{MAX} \quad \cdots\cdots (2)$$

H_a, H_b and H_c were corresponding to the H_{1cm}s calculated from the Xs of the detector 1, 2 and 3, respectively. H_{MAX} was defined as a maximum dose of H_a, H_b and H_c. Coefficients of 0.06, 0.28, 0.61 and 0.05 were obtained as partial sums of tissue weiting factors,as shown in Table 1.

Table1. Distributions of tissue weighting factors in recomendations-1990 of ICRP in areas with each detector.

	Detector1 (Head and Neck)	Detector2 (Thorax and Brachium)	Detector3 (Abdomen and Femur)	Max. of the three (Other tissue)
	0.05(thyroid)	0.12(lung)	0.20(gonads)	0.05(remainder)
		0.05(breast)	0.12(stomach)	
		0.05(oesophagus)	0.05(liver)	
			0.05(bladder)	
			0.12(colon)	
	0.012(bone marrow)	0.048(bone marrow)	0.06(bone marrow)	
	0.002(bone surface)	0.004(bone surface)	0.004(bone surface)	
	0.002(skin)	0.004(skin)	0.004(skin)	0.001(skin)
Total	0.066(~0.06)	0.276(~0.28)	0.608(~0.61)	0.051(~0.05)

RESULTS

The maximum $\dot{X}$ of 2.71 mR/h per 111 MBq was recorded at the Level II and the posterior projection. In all records, the $\dot{X}$s in posterior projection were similar to those in anterior, however, right and left lateral $\dot{X}$s were obviously lower than those. $\dot{X}$s at 0.5, 1.0, and 1.5 m from the patients were about one-fifth, one-tenth and one-twentieth of the surface one, respectively. The $\dot{X}$s in this investigation were obviously higher than the theoretical attenuation according to the inverse square law, when it was assumed that a Ga-67 citrate patient was a point source put in a free air. The retention rate calculated from $\dot{X}$s at 0.1 and 48.0 h after administration of Ga-67 citrate indicated a good agreement with that by Evelyn E. Watson et al.. H_{1cm} and E for a first week (168 h) were determined by the equation (1) and (2), as shown in Table 2. The maximum H_{1cm}

for a first week were 2216, 410, 172 and 89 μSv at distances of 0.05, 0.5, 1.0 and 1.5 m, respectively. These H_{1cm}s for a first week were equal to 82.4 percent of total radiation doses for unlimited time (∞ h).

Table2. Potential equivalent dose (H_{1cm}) and effective dose (E) around the Ga-67 citrate patient.

Area of dose evaluation	μ Sv/168h/111MBq (n=23)			
	Distances from Ga-67 citrate patient			
	0.05m	0.5m	1.0m	1.5m
Head and neck (H_{1cm})	1047	288	136	72
Thorax and brachium (H_{1cm})	2216	410	172	89
Abdomen and femur (H_{1cm})	1945	393	158	80
Effective dose (E)	1980	392	161	83

$$E=0.06H_a+0.28H_b+0.61H_c+0.05H_{MAX}$$

CONCLUSION

As far from 0.05 to 1.5 m, radiation exposure rates projected from Ga-67 citrate patients effectively reduce, however these rates are maintained for a long time. Accordingly an area around the patient is considered as a small hazard of radiations, particularly for the persons who may come into close contact to him. The patient provides only a small risk to otheres, but it should be recommend to aviod a long contact to the patient.

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RADIATION EXPOSURE AROUND PATIENTS AFTER ADMINISTRATION OF Tc-99m-DPD OR Tl-201-CHLORIDE

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INTRODUCTION

In connection with the diagnostic use of radiopharmaceuticals in nuclear medicine it often is asked if the radiation exposure by patients after administration of the radioactive substances might endanger their sorroundings and to what extent countermeasures are necessary for protection (e.g. establishment of special waiting rooms, hospitalization of the patients in special bedrooms as it is normal in nuclear medicine therapy). We examined quantitatively the radiation exposure caused by two of the most frequent nuclear medicine investigations: bone scintigraphy with Tc-99m-Dicarboxypropan-Diphosphonat (-DPD) and cardiac studies with Tl-201-chloride, respectively.

MATERIAL AND METHODS

Dose rate measurements were performed in 0.5m, 1m and 2 m distance from the surface of the bodies of two groups of patients, within the first 4 h after they had received 600±30 MBq of Tc-99m-DPD or 100±10 MBq Tl-201-chloride, in time periods of 30 minutes. In the case of Tl-201-chloride the measurements were repeated 24 hours (in some cases in addition up to 370 hours) after administration. Both groups consisted of 16 patients.

We used three calibrated dose rate detectors of the same type (Berthold LB 133®) in a stationary geometry. The patients were sitting on an chair, and the sensitive counting volumes of the detectors were positioned in about the height of the sternum (ventral).

RESULTS

In Table 1 the initial values of the dose rates are listed. Figure 1 shows the time course of the measured dose rates. Monoexponential functions were fitted. In the

Table1. Initial values of the dose rates in μGy/h (mean ± SD; n=16)

Distance	Tc-99m-DPD	Tl-201-chloride
0.5 m	19,7 ± 3,9	3,82 ± 1,04
1 m	5,66 ± 0,75	1,18 ± 0,29
2 m	1,85 ± 0,13	0,30 ± 0,08

case of Tc-99m-DPD the dose rates decreased with an effective half-life of 2.3 hours; in the case of Tl-201-chloride the effective half-life was estimated to be 60 hours. The corresponding biological half-lives thus were 3.7 hours and 330 hours, respectively. The total error in all cases was about 10%.

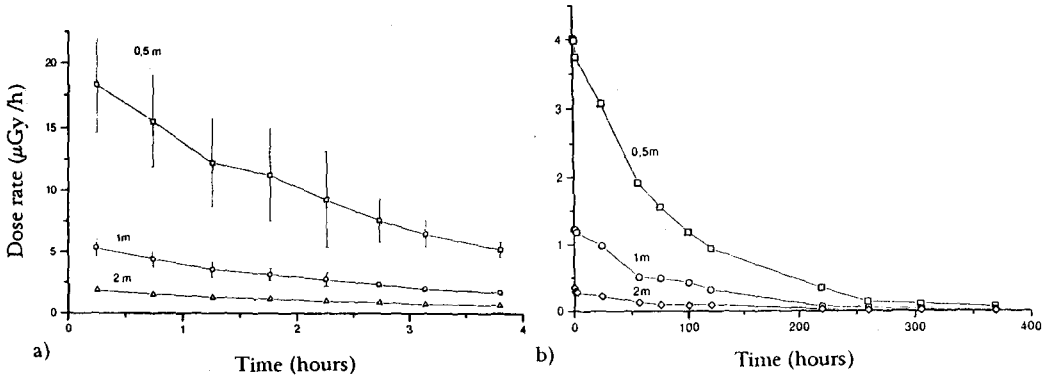

Figure1. Time courses of the dose rate (in μGy/h) in 0.5, 1 and 2m distance from patients after administration of 600 MBq Tc-99m-DPD (Figure 1a) and 100 MBq Tl-201-chloride (Figure 1b), (mean ± SD; n=16).

DISCUSSION

Integrating the dose rate functions the dose around the patients can be calculated for any time period after the administration of the radiopharmaceuticals (Figure 2). Thus the possible radiation exposure of other patients, attendents, and nurses in the surrounding of the "radiating" patients were estimated. Let us discuss five cases of possible exposure to other persons.

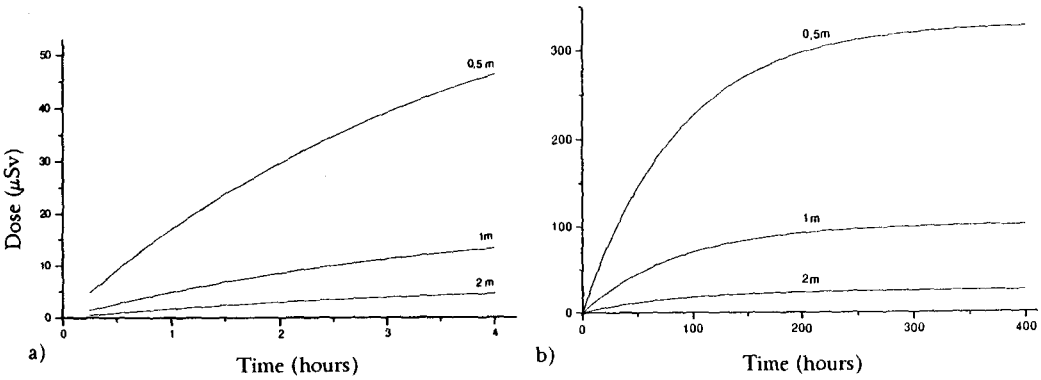

Figure 2. Courses of the radiation doses (in μSv) in 0.5, 1 and 2m distance from patients after administration of Tc-99m-DPD (Fig.2a) and Tl-201-chloride (Fig.2b), calculated from the mean values of the dose rates.

Tc-99m-DPD

The maximum dose around a patient after application of 600 MBq Tc-99m-DPD was calculated to be 67 μGy in 0.5m (19 μGy in 1m, 6.7 μGy in 2m) distance (1).

Case A: After application the "radiating" patient is sitting for 4 hours in a distance of 0.5m beside another patient. This is the longest time period a patient has to wait for bone scintigraphy. The exposure of the neighbour patient then may amount to 46 μSv, corresponding to 3% of the maximum permissible additional annual exposure to non-radiation workers.

Case B: After bone scintingraphy a patient is realeased from the hospital or is situated as in-patient in a ward. During a permanent stay in a mean distance of 1m an accompanying person or a nurse may be exposed by a dose of 7 μSv, corresponding to <0.5% of the permissible annual dose.

Tl-201-chloride

In the case of Tl-201-chloride the maximum doses in 0.5m, 1m and 2m distance from the patient are 330 μSv, 102 μSv and 26 μSv, respectively.

Case C: Due to a cardiac investigation a patient is sitting permanently from ½ to 4 hours after administration of Tl-201-chloride in a distance of 0.5m beside another patient. Then the exposure to this patient is 13 μSv, corresponding to <0.9% of the permissible annual dose.

Case D: After a cardiac investigation a patient is released from the hospital or is situated in a ward. During a permanent stay in a mean distance of 1m during the whole following day the exposure to a relative or to a nurse is <26 μSv, corresponding to <1.8% of the maximum permissible annual dose.

Case E: The patient (Case D) is hospitalized for two weeks or longer in the same sick-room. In this case from a permanent stay in 1m distance a nurse may receive a dose of 105 μSv. Considering a working time of 40 hours per week, this maximum dose ist reduced to 25 μSv, corresponding to <1,7% of the permissible annual dose.

As a result even under very restrictive assumptions the doses to the environment caused by "radiating" patients are far below the limits set by radiation protection regulations. There is no necessity to separate patients from other patients in a special waiting area after administration of the radioactive agents. This result is confirmed by studies of other researching groups (2), (3).

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OFF-AIR MONITORING FOR PET-CENTERS - PRACTICAL ASPECTS

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INTRODUCTION

During operation of a PET-Center, the air is contaminated by radioactivity. Apart from accidents, this contamination may be caused, for example, by neutron-induced air activity during operation of a cyclotron, leaks in the chemical process units and the use of radioactive gases on patients. Therefore, the room air as well as the waste air have to be monitored. Due to lack of experience, there is still uncertainty regarding the scope and the proper layout of such measuring systems.
This paper will discuss the principle of a waste air monitoring system which features adequate sensitivity as well as the required retaining capability in case the activity is exceeded.

RADIATION EXPOSURE THROUGH WASTE AIR

According to (1), the activity concentration of radionuclides, where the submersion is determinative for the limit value, must not exceed 200 $Bq.m^{-3}$ in the air on the annual average.
In the borderline case of an infinitely expanded activity cloud and on the basis of dose factors according to (2) and (3) this will result in an effective dose of 0.31 mSv per year for C-11, N-13 and O-15, and 0.35 mSv for F-18, and 0.43 mSv for Ar-41. For infants (1 year) these values have to be multiplied by a factor of 1.2. The indicated dose values can be attributed to the γ-submersion by nearly 100%. The share of the effective dose due to inhalation of 0.05 $mSv.y^{-1}$ is only perceptible with F-18. The exposure of skin and eye lenses caused by β-submersion is only 3 $\mu Sv.y^{-1}$ effective skin dose or 14 $\mu Sv.y^{-1}$ for the lenses, even for O-15 with its high β-energy of 1.73 MeV.
The actually occurring radiation exposure, however, is always below 0.01 $mSv.y^{4}$ (4), due to the activity cloud which is always limited in scope and the dilution factor. Keeping this in mind, one can, therefore, define a higher limit value - in accord with (1) - on the basis of propagation calculations and the estimation of the expected dose of the population in order not to have to restrict the operation of a PET center.

TASKS AND POSITIONING OF THE MEASURING SYSTEMS

The measuring and monitoring system for PET nuclides is designed to

- prevent the release of high activity concentrations from the respective working area, so-called advance measuring systems
- determine the emitted activity relative to the approved limit values or limit values stipulated by law, the integration measuring system.

Advance Measuring Systems (5)

These measuring systems are used for the rapid measurement of radioactive PET nuclides which, when released from the waste air channel, would exceed the permissible emission values. At the same time, these measuring systems control, for example, blocking flaps or initiate a storing process to allow the short-lived PET nuclides to decay. The time required for measuring these activities is dependent upon the selected detector configuration, and in particular upon the ratio of the partial waste air flow being monitored relative to the total waste air quantity. To prevent that activity can leak out during the measurement, a delay volume that is dimensioned according to the requirements is installed into the partial waste air path before the blocking flap.

These measuring systems are used, e.g., in the partial waste air paths of the cyclotron, in hot cells, the PET room, etc.
Figure 1 shows the schematic layout of an advance measuring system.

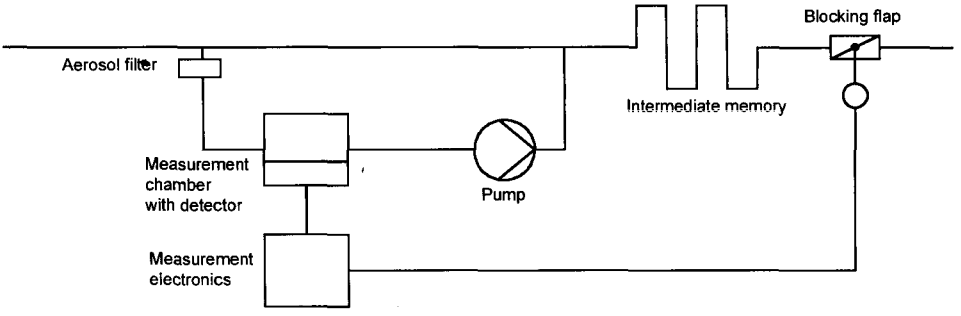

Figure 1: Waste air monitoring by an advance measuring system

Integration Measuring System (5)

This measuring system is designed to monitor and register the activity emission relative to the fixed limit values in Bq/m^3 as well as integral in Bq/time. The detector configuration is installed close to the blocking flap in order to get representative data on a possible activity emission. This measuring system has to be designed so sensitive that any exceeding of the activity concentration of 200 Bq/m^3 will safely be detected at any time.
We recommend using setups featuring bypass sampling in a partial air flow, since due to the defined measurement volume and the chance to shield off gamma interference radiation one will obtain a clear calibration value and thus a representative measurement value.

Figure 2 shows such an integration measuring system.

DETECTOR CONFIGURATIONS

Since PET nuclides are positron sources, all detectors that are used for beta measurement can be used. In addition, one can also perform a measurement via the 511 keV gamma destruction radiation.
The drawback of gamma measurements is, however, that all important PET nuclides are energetically identical, but Ar-41 is taken into account as an air activation product with 1.29 MeV with different calibration factors, and a defined measurement geometry without gamma interference by the 511 keV radiation is hard to achieve. Large-area proportional counter tubes used for beta measurement have the drawback that one gets about the same calibration factors for the radionuclides O-15, C-11, N-13 and Ar-14. For F-18, the calibration factor is only about 30% less sensitive than the reference nuclide C-11. This means: on the basis of the German Radiological Protection Ordinance (1), which defines the same limit value for all radionuclides (except F-18), the radioactivity emission in waste air will not be overrated to the disadvantage of the user.
Large-area proportional counter tubes can be regarded as the optimum detector for positron sources and other gaseous β-activities. They have a lower γ-sensitivity (< 1%) and allow adjustment to large measurement volumes. For the integration measuring system, we recommend using an 8 liter measurement chamber with flanged proportional counter tubes (active area: 900 cm^2). This ensures that one will get a calibration factor of less than 80 $(Bq.m^{-3})/cps$ and one can monitor the limit value of 200 $Bq.m^{-3}$ in a measurement cycle of less than 4 minutes. For the advance measuring systems one can use, for example, 1 liter measurement chambers with flanged proportional counter tubes (active area: 200 cm^2). In this case, one can obtain a calibration factor of less than 600 $(Bq.m^{-3})/cps$.

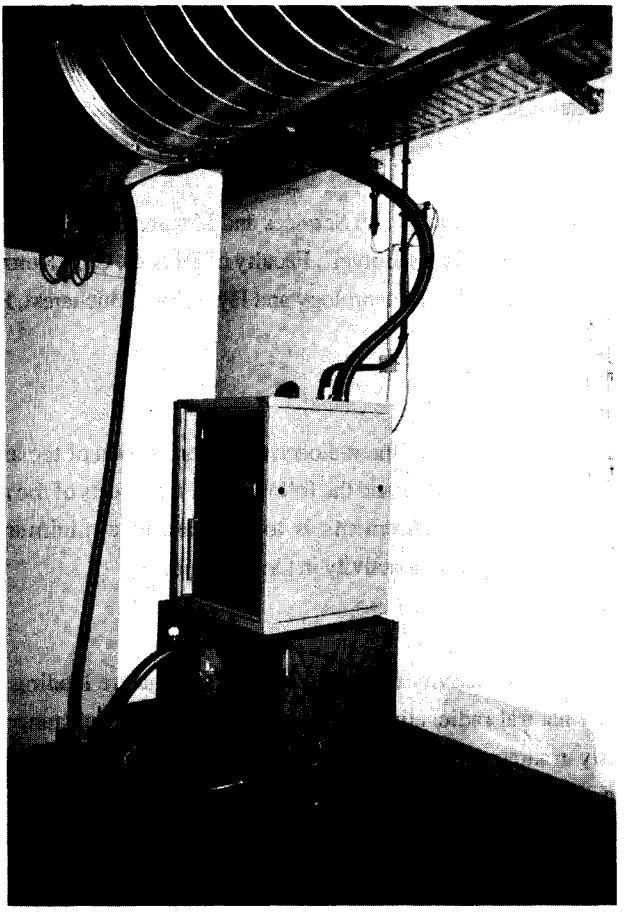

Figure 2: Integration measuring system

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STUDIES OF RADIOACTIVITY OF THE AIR IN CLUJ-NAPOCA, ROMANIA

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SUMMARY

We present measurements of the radioactive contamination of the air in the laboratories of the some nuclear medicine centers and the follow-up of the effects of the nuclear activity.

The purpose of the measurements is to decrease to a minimum the occupational exposure and the monitoring of radioactivity in the atmosphere.

INTRODUCTION

The human nuclear activity contributes to the radioactive loading of the atmosphere, beside the initially natural radioactive loading. The radioactive substances in the aerosols or gases, may directly irradiate the human body by inhalation, ingestion or immersion, or indirectly through water and food (1, 5, 8).This paper presents the results of the verifications of air contamination in the laboratory rooms of the Nuclear Medicine Units in comparison with the atmosphere of Cluj-Napoca town (4).

MATERIALS AND METHODS

Radioiodine was selected as an indicator of contamination of the occupationally exposed workers because of its noxious effects (1). The air samples were collected in different workrooms e.g. laboratories for the first dilution, administrative and measuring offices.

The principle of the working methods is the pumping of known air quantity through special filters and measuring the filter radioactivity (2, 8, 9).

The filtering cartridge for ^{131}I is made of active coal impregnated with $AgNO_3$ solutions (2) and was measured with the well type NaI(Tl) crystal connected to an impulse-meter having an efficiency of 20% to 40%.

The beta radioactivity of the SYNPOR membrane situated in the front of he active coal was measured in anticoincidence counter connected to an impulse-meter. The efficiency is about 9,8% and the counting times were of about 2 hours.

In 1991 the filtering cartridge made of active coal impregnated $AgNO_3$ solution was replaced by the SLF 2I-50 special filter for radioiodine and the SYNPOR membrane filter was replaced by the FPP filter.

The results have been determined with the following formula:

$$A = \frac{R - F}{E \cdot e \cdot V}$$

where A=sample activity (Bq/m^3 or mBq/m^3); R=filter counting rate (imp/s); F=fund (imp/s); E=counting system efficiency; e=filtering efficiency; V=volume of filtered air (m^3).

In 1994 the used filters were searched for ^{131}I, ^{137}Cs and ^{60}Co by measuring them with a GeLi detector connected to a ICA-80 multichannel analyzer.

RESULTS AND DISCUSSION

Table 1 resumes the results obtained in 1979 of the average concentration of ^{131}I in the laboratory air where this isotope was used.

Table 1. Average values for ^{131}I in air in 3 medical laboratories in 1979.

Labor. No.	Dates	^{131}I (Bq/m^3)
1	May 21	1,34±0,15
1	May 21	3.78±0.44
1	May 28	3.03±0.33
1	May 28	under MDC
2	October 15	1.67±0.18
2	October 15	2.34±0.24
2	October 16	1.67±0.18
2	October 16	2.18±0.22
3	June 28	1.67±0.18
3	June 28	2.07±0.27
3	July 3	5.37±0.56
3	July 3	5.56±0.56
3	July 6	2.37±0.26
3	July 6	1.93±0.20

MDC=1.1 Bq/m^3

Analyzing the results we see that the values are under the occupationally maximum allowable concentration (1), but in the 3rd laboratory they are increased because of the high number of patients diagnosed and treated there.

The ^{131}I concentration in the atmosphere was determined simultaneously. For these determinations a larger air volume was collected, the sampling was up to 18 hours for diminishing the MDC value. The ^{131}I concentration is under the minimal detectable concentration and the results for beta radioactivity of aerosols varied between 4 mBq/m^3 and 48 mBq/m^3.

On May 1986 the maximum concentration of ^{131}I was of 170 Bq/m^3 and after 20 May all the values were under 3,3 Bq/m^3.

The values registered in 1995 for the studied workrooms of the ^{131}I concentration range between 1,11 Bq/m^3 and 29,63 Bq/m^3 and are situated on an average below the admissible levels of the occupational exposure norms.

CONCLUSION

In the nuclear medicine laboratory the radio protection norms have been respected, the studied workrooms do present safe work conditions.

Although the air quality in Cluj-Napoca town for overall beta radioactivity and ^{131}I, ^{137}Cs and ^{60}Co measurements is not harmful, the air radioactivity must be kept under control.

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NUCLID PRODUCTION WITH CYCLOTRONS AND RADIATION PROTECTION PROBLEMS

DIPL.ING.ERICH L. HUBER, VIENNA

With cyclotrons it is possible to produce nuclids used for medical and biological examinations with short halftime in a simple manner. The short halftime is the benefit for the mean irradiation of patients but it has the disadvantage that the production rate of the nuclids has to be very high. The time between production and administration of the nuclid to the patient is responsible of the produced activity of the radioactive agens. The longer this time and the shorter the half time is, results the irradiation risk. It shall be calculated the dependence of these factors and measures will be layed down to optimize the risks of the production and use of the radioactive agens.

DIE AUSWIRKUNG DER PRODUKTION VON NUKLIDEN IM ZYKLOTRON AUF DEN STRAHLENMSCHUTZ

Das Zyklotron bietet die Möglichkeit Radionuklide herzustellen, die für nuklearmedizinische Untersuchungen neue Methoden und dies bei wesentlich geringerer Strahlenbelastung des Patienten ergibt. Die hiebei erzeugten Radionuklide besitzen eine kürzere Halbwertszeit, müssen also in höherer Konzentration bzw. Aktivität angewendet werden, um die erforderlichen Resultate zu liefern. Die Aktivität, die im Zyklotron erzeugt werden muß, hängt davon ab, nach welcher Zeit die Nuklide verabreicht werden und die Messung erfolgt. Je länger diese Zeit ist und je kürzer die Halbwertszeit des Nuklids ist, desto größer ist das Risiko einer Strahlenbelastung. In welchem Ausmaß dieses Risiko durch die Erzeugung erhöht und bei der Anwendung vermindert wird,soll berechnet werden, und wie weit zusätzliche bzw. andere Strahlenschutzmaßnahmen erforderlich sind, soll angegeben werden.

PRINCIPLE

The conventional cyclotron consists of a circular, evacuated chamber situated in a uniform magnetic field. Inside this chamber are two hollow electrodes with an electrostatic difference of potential where a positive ion is accelerated to high energy E = 4.8x10e-11(HRZ)2/A, where E is expressed in MeV,H in gauss,R in cm and A,Z are the mass and the atomic number of the accelerated positive ion. The magnetic field strength of the order of tens of kilogauss limits the design of the cyclotron. Protons and Alfa particle are accelerated with a magnetic field strength of 15 kilogauss and a dee diameter of 76 cm to about 15 MeV. For the installation in medical institutions smaller compact cyclotrons are used, where the magnetic field varies with azimuthal angle, which are called AVF azimuthally varying field cyclotron. There are more than 80 short-lived radionuklides used by medical institutions, most with physical half-lives less than a few days, produced approximately 50% by cyclotrons. Most of interest /2

for medical purposes for instance for the Positron Emitter Tomography (PET) are the „physiological" radioisotopes (11C,13N,15O,18F). The estimated radionuclidic production with a 40 MeV cyclotron is as following:
14N(p,4He)11C, 20.30 m Half-Life, 45.5 TBq Activity, 11 h/w Production
16O(p,4He)13N, 9.96 m Half-Life, 4.8 TBq Activity, 3 h/w Production
14N(d,n)15O, 2.02 m Half-Life, 6.8 TBq Activity, 3 h/w Production
18O(p,n)18F, 109.80 m Half-Life, 34.6 TBq Activity, 13 h/w Production
20N(d,4He)18F, 109.80 m Half-Life, 2.1 TBq Activity, 2 h/w Production
124Xe(p,2n)123I, 13.00 h Half-Life, 2.3 TBq Activity, 4 h/w Production
Other Radioisotopes, 43.0 TBq, 8 h/w Production
The main areas of the cyclotron are the vault of the cyclotron and two target bunkers with automatic pneumatic transportation systems to manipulation and labelling laboratories and the PET area and also with shielding wagon running on a monorail to connect bunkers with hot radiochemical laboratories.
The generation of neutron and photon fluxes into the vault and the bunkers needs ordinary concrete shielding with an addition of 1% of borate materials with a thickness of about 250 cm. Lower shielding levels of about 50 cm of concrete has to be projected for the hot radiochemical laboratories. The calculation of the shielding walls will be made with an overestimation of about tenfold to protect the workers. The nuclei present in the air and in the dust are responsible for radioactive gaseous effluents mainly from the interaction of primary and secondary particles.Therefore an air circulation system of 0.5-1 time/h during the run and 8-10 after the decay time is necessary to protect workers and population. The following main isotopes contribute the airborne radioactivity:
14N(n,2n)13N, Half-Life 9.96 m, Beta+(1.25 MeV)
16O(n,2n)15O, Half-Life 122 s, Beta+(1.7 MeV)
16O(n,p)16N, Half-Life 7.20 s, Beta+,Gamma
40Ar(n,4He)37S, Half-Life 5.06 m, Beta-(4.8 MeV)
40Ar(n,p)40Cl, Half-Life 1.40 m, Beta-(7.5 MeV)
40Ar(n,Gamma)41Ar, Half-Life 1.83 h, Beta-,Gamma
It was found that the proportion of the gamma dose near the accelerators is generally negligible considering the neutron dose equivalent. This means,that the control of the gamma dose alone is insufficient for the accidental dosimetry around cyclotrons.
Possible sources of radioactive liquid waste are from accidental releases from the cooling water circuit, from the radiochemical laboratory, from the PET area and their treated patients.
Possible sources of radioactive liquid waste are from accidental releases from the cooling water circuit, from the radiochemical laboratory, from the PET area and their treated patients.

CALCULATIONS
The production of radioactive sources for medical diagnostic methods with a cyclotron needs a very different way compared with using long-living nuclids. To have the same measurement effects with short-living nuclids you must use /3

much higher activities. Additionally you need yet higher activities produced by the cyclotron. This means that the level of radiation protection must be higher to protect workers on the one side and lower to protect the population on the other hand and vice versa.
To estimate the necessary protection, the activities of cyclotron produced radionuclides will be calculated and compared with the activities using long-living nuclides. Finally the dose ratio will be calculated dependent from the half-lifes for workers and patients.
The activity of the nuclid originated in the cyclotron is Ao. After the time To the intake of the nuclid of the patient is beginning and continues until Td.
After To is the activity Ad=Ao.exp(-To.ln2/T), with the Half-life T. From this results the proportional dose D= Ao.(-T/ln2).(exp(-(To+Td).ln2/T)-exp(-To.ln2/T))
The dose comparison between two nuclids with different half-lifes after the Td leads to the unknown activity Ax=A.(T/Tx).exp(-Td.ln2/T.(1-T/Tx)).
The activity at the time of the production can be calculated as follows:
A=Ao.exp(-To.ln2/T) and Ax=Axo.exp(-To.ln2/Tx). Therefore is
Ao=A.exp(To.ln2/T) and Axo=A.(T/Tx).exp(-Td.ln2(1/T-1/Tx)).exp(to.ln2/Tx)
Important is the ratio Axo/Ao=(T)TX(-ln2.(To+Td).(1/T).(1-T/Tx)).
Also important is the proportional dose ratio DX/D=exp(-Td.ln2.(1(T).(1-T/Tx)).
wherein is
Ao the activity of the known nuclid at the production in the cyclotron
Axo the activity of the nuclid produced in the cyclotron
T the half-life of the nuclid used without cyclotron
Tx the half-life of the cyclotron produced nuclid
Td the time of the measurement and
To the time between the production of the nuclid in the cyclotron and the begin of the measurement.
The exposure factor of the patient result from following equation if you use the effective half-life: EFP=Ax(Txeff)/A(Teff), 1/Teff=1/Tphys.+1/Tbiol.
For Txbiol.>>Txphys. and Tphys.>>Tbiol. is EFP -- Txphys./Tbiol.
Also you can define the exposure factor of the worker EFW which result from the time between the cyclotron production or the delivery of the nuclid intil the measurement.
With the manipulationtime Tm is EFW=T/Tx.exp((Td+Tm)(1-T/Tx)ln2/T)
This gives the result for T>>TX and Td+Tm=Tx of EFW --(T/2)/Tx.

CONCLUSIONS

The activity of the nuclids produced in a 40 MeV cyclotron is about 100 TBq / week.
The medical examinations with 99mTc, 131J use activities from 1.5 til 740 MBq dependent from the radiopharmakon. From the standpoint of the patient the radiation protection is much higher using cyclotron produced nuclids because of the short half-life. On the other hand the shorter the half-life the higher is the required activity production and also the risk of irradiation. It is also more difficult to control the intake of workers. Therefore the radiation production expense will increase very intensively with decrease of risk.

MEASUREMENT OF THE EXPOSURE OF THE SWISS POPULATION TO MAGNETIC FIELDS OF 50 Hz POWER FREQUENCY AND 162/3 Hz IN RAILWAYS

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INTRODUCTION

All installations that generate, transmit, or use electric power cause electric and magnetic fields. Common to all types of sources is a strong dependence of the magnetic flux density on the distance to the source. However, this information is not sufficient to know to what degree various parts of the population are exposed to magnetic fields during the different periods of the day. For this reason a study was carried out to assess the typical exposure of the Swiss population to the magnetic fields of 50 Hz power frequency and to 162/3 Hz magnetic fields in railways. A method of data reduction that allows for the determination of frequency distribution and percentiles for any selection of measurements was applied.

50 Hz MEASUREMENTS

These measurements were made with EMDEX-II instruments which register the resultant of the magnetic flux density up to about 300 μT. In the broadband mode the frequency response is flat from 40 to 800 Hz. Thus, for 50 Hz fields the magnetic flux density is measured correctly, while fields from railways (16 2/3 Hz) are underestimated.

Data from 464 employees were collected on weekdays (24 h, sampling interval (sa) = 10 s). To complete the time range and the demographic spectrum, 45 of the employees collaborated also for weekend measurements (48 h, sa = 30 s). In addition, data from 43 house keepers were included as well (24 h, sa = 10 s). The collected data were transferred to a notebook and the time series were evaluated with the EMCALC V2.12 software. The information of a simple protocol served to split the time series into 5 environmental categories: at work, commuting, at home, at rest and miscellaneous. For each environmental category of every person the frequency distribution was calculated with the EMCALC software. For further analysis the numerical output data were transferred to a large EXCEL table.

162/3 Hz MEASUREMENTS

Measurements in railways were made with HFR-1200 instruments. The instruments register the resultant of the magnetic flux density in the frequency range from 16 to 500 Hz with a sampling interval of 2 seconds, 1 minute or 1 hour.

Magnetic field measurements were made in fast trains from Zurich to Geneva and also from Zurich to Chur. More data were collected in double-decker S-trains around Zurich. In all cases the magnetic flux density was measured at floor-level, at waist-height and in the luggage rack. Most of the time the sampling interval was 1 minute, only short measurements were made with a sampling interval of 2 seconds. The collected data were transferred to a notebook and the time series were evaluated with the HFR-1200 software and the graphic program EXCEL V5.0.

RESULTS

50 Hz fields

The evaluation of the complete data set gives the following results: Median 0.05 μT, geometric mean 0.06 μT, geometric standard deviation 3.8, arithmetic mean 0.21 μT. 69% of the recorded values are below 0.1 μT, 3% are above 1 μT, and only 0.17% are above 10 μT. The results for "commuting" are intermediate, but these values may represent only a lower limit. 162/3 Hz fields from railways, which are likely to be an important source for this category, are not included in these results.

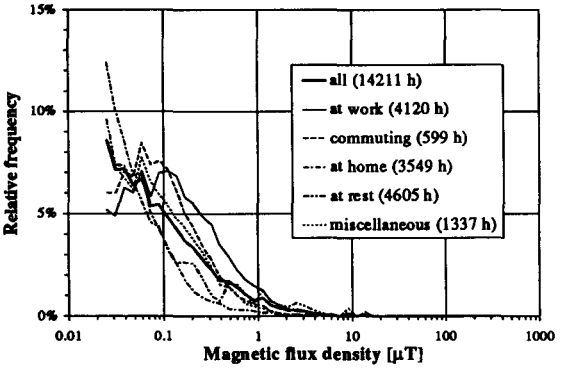

Figure 1:

50 Hz magnetic fields: Relative frequency distribution curves for the complete data set and the main categories. Total measurement times were put in parenthesis.

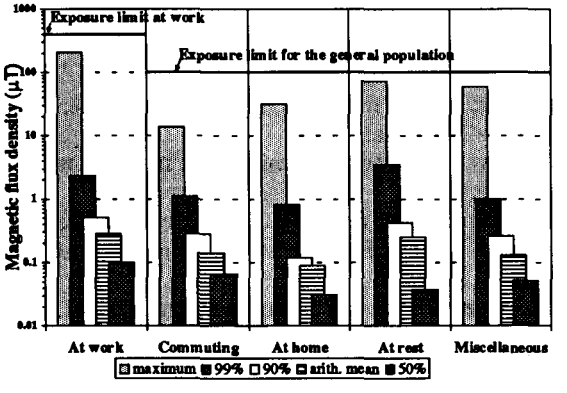

Figure 2:

50 Hz magnetic fields: Maximum, arithmetic mean, 99%, 90%, and 50% percentiles. The exposure limits for the general population (100 μT [3]) and at work (400 μT [4]) which are based on acute biological effects, were never exceeded.

As in similar studies [1, 2], the data of the 50 Hz measurements for all 5 main categories show an approximately log-normal distribution. As expected, the highest exposures were found in the category "at work" and the lowest magnetic flux densities were recorded "at home". Quite surprisingly for the category "at rest" the measured magnetic flux densities are partially higher than those "at home" (see Figure 1) and the arithmetic mean is almost as high as "at work" ($mean_{at\ work}$ = 0.28 μT; $mean_{at\ rest}$ = 0.25 μT; see Figure 2). This is mainly due to a small number of cases with fields in the range from 1 to 15 μT. The main reason for these high readings was, that some people put the EMDEX meter during the night very close to a clock radio or a similar electric appliance. For this reason the measured magnetic flux densities during the night may not be representative for the exposure of the person. Therefore, detailed measurements were made in bedrooms. With the exception of two cases the arithmetic mean within the area of the bed had been close to the arithmetic mean in the category „at home" of earlier

measurements (< 0.1 μT). In one of the exceptional cases an electrical mattress caused arithmetic means between 0.08 μT and 0.45 μT. In the other case higher values (arithmetic means between 0.18 μT and 1.95 μT) had been measured around the pillow only because of a nearby clock radio.

162/3 Hz fields

In trains the arithmetic means of the magnetic flux density varied in most cases between 3 μT and 6 μT with a maximum of 20 μT. A maximum single value of 75 μT was measured in a double-decker train on the lower floor at ground-level. In contrast to the 50 Hz magnetic fields the readings for 162/3 Hz in trains show almost symmetric distributions. The exposure limit for the general population for 162/3 Hz magnetic fields of 300 μT [3] has never been exceeded.

CONCLUSIONS

In residences a range from 0.02 to 0.04 μT can be considered as background level for 50 Hz power frequency magnetic fields. A significant enhancement of magnetic field exposure may be generated by electrical appliances. This is the case in particular if the distance to appliances containing transformers or motors is less than 1 m for an extended period of time. In trains (162/3 Hz) mean magnetic flux densities of 3 μT to 6 μT can be considered as typical. The recommended exposure limits of 100 μT for 50 Hz and 300 μT for 162/3 Hz have never been exceeded.

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ACKNOWLEDGEMENT

We are grateful to the help by the participants of the field studies, and in particular to the staff of Swiss Rail, Zurich. The Forschungsgesellschaft für Energie- und Umwelttechnik (FGEU), Berlin has made the HFR-1200 instruments available. The study was partly financially supported by the Federal Office of Environment, Forest and Landscape (BUWAL) and the Projekt- und Studienfonds der Elektrizitätswirtschaft (PSEL).

STUDY OF THE MUTAGENIC POTENTIAL OF LOW POWER MICROWAVES BY DIRECT DNA ANALYSIS

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INTRODUCTION

Man is constantly exposed to electromagnetic radiation which is a constituent part of the biosphere. Due to the technological progress and numerous man-made sources such as those used in industry, traffic, medicine, defence, communication network as well as some home appliances, the character of the natural electromagnetic field has altered significantly. These sources of nonionizing radiation cause atmospheric pollution similar to the pollution from various industrial sources. Unambiguous detection of the mutagenic potential of these radiations is necessary for genetic well being of future generations.

DIRECT DNA ANALYSIS

DNA was isolated from brain and testis tissues of male Swiss albino mice exposed to low power microwave ($1 mW/cm^2$, 2.45 GHz continuous wave) for 2 hrs daily for a period of about 200 days. A variant of the restriction fragment length polymorphism (RFLP) approach was developed to study the effect of at the DNA sequence level using a single restriction enzyme digest and hybridization with a short synthetic oligodeoxynucleotide probe specific for a simple repetitive sequence $(GACA)_9$.

RATIONALE: Eukaryotic genome harbors two principally different classes of DNA: a) single copy DNA present once per haploid chromosome set; b) repetitive DNA existing from a few to a few hundred thousand (or million) copies. The whole array of repetitive sequences may comprise from less than 10% to more than 90% of the genome in different animal and plant species (in man it is about 30%). Short sequence motifs (less than 10 bases long) are ubiquitous repetitive components of eukaryotic genome (1). The number of repeat units of such simple sequences vary considerably causing allelic variation. When sequences complimentary to these repeat units are used as hybridization probes, polymorphism at several loci is simultaneously detected and individual specific hybridization patterns are generated (2). Of the various types of hybridization patterns obtained, multilocus monomorphic band profile is particularly useful because any loss or gain of band due to sequence rearrangement or sporadic mutations can be easily detected by the technique.

CYTOGENETIC ANALYSIS

The sensitive analysis of sister chromatid exchange (SCE) induction , which represents a sensitive and qualitative index of mutagen-carcinogen-induced DNA damage in eukaryotic chromosomes, was used as a cytogenetic end point to assess the mutagenic potential of microwaves. Bromodeoxuridine (BrdU) tablets (25 mg) were implanted subcutaneously. 19 hrs posttablet implantation, mitotic arrests were established by administration of colchicine (25 µg/animal). Metaphase plates prepared from bone marrow, were stained by modified fluorescent plus Giemsa (FPG) technique (3).

RESULTS

A multilocus monomorphic band profile was demonstrated with Southern blots in both the tissues of brain and testis studied. Hybridization profile of the brain DNA showed a sharp band (mol. wt 8.2 kb) in both the control and the exposed group of animals (Fig.1, arrow). In the exposed animals, an additional band at 7.7.kb appeared below this particular band. Gel track analysis showed a sharp peak marked 1 (Fig. 3 a,b) in both the control and the exposed animals and the appearance of a second peak marked 2 (Fig. 3 d-g) in all the exposed animals.

The hybridization profile in the testis DNA is not a reflection of sharp band difference in this region between exposed and control animals but rather a broadening of the band width. Both control and exposed animals revealed a sharp band in the testis DNA in the region of 8.1. kb (Fig. 2, arrow). The track histogram of both control and exposed animals showed a sharp peak marked 1, corresponding to 8.1 kb (Fig. 4, a-c). In the exposed animals, there was a change in the peak profile with the appearance of a second peak marked 2 corresponding to 7.7 kb (Fig. 4, d-g).

It is interesting to note that amidst a large number of bands at identical positions in DNA of both controls and exposed group of animals, rearrangement of DNA is consistently observed in all the exposed animals in the same region between 7-8 kb, irrespective of the duration of the exposure.

No statistically significant differences in the SCE frequency were recorded between the control and the irradiated animals exposed for various intervals of time. The mean value of SCEs in both control and exposed group of animals fell between 2.47+1.80 to 3.53+1.76; range 0-7 . The distributions were reasonably continuous and unimodal so that the difference in mean value could be evaluated in terms of standard statistical parameters of normal distribution. One level of ANOVA test was applied for the evaluation of the SCE frequencies using epistat software programme.

DISCUSSION

In the present study, the DNA analysis from exposed animals with nine repeat 5'-GACA-3' probe show similar profiles as those produced from the control animals with the notable exception of a prominent 7.7 kb Hinf I fragment specific to all exposed animals. Since this particular fragment is not present in the control animals but appears after microwave exposure, it is suggested that probably in the unexposed animals, the copy number of these repeat sequences is not sufficient to form a distinct band. Microwave exposure may have led to the amplification of these tandem sequences generating more copies of GACA sequences in this particular region. Although it is not known at present whether exposure to a mutagenic agent or a specific class of mutagens increases the mutation rate in the region of these tandem repeat units, it is known that stress induces amplification by extra replication of DNA segments in the non coding repeat sequences (4). The observed change of DNA rearrangement could be attributed to some sort of non-specific stress created by low intensity microwave field.

However low power microwave did not induce increased incidences of SCE formation in the exposed animals compared to the controls. This may be explained on the basis that though both SCEs and amplification phenomenon form a part of recombinational events, mechanisms of their formations are different. Interestingly, a parallel can be drawn at this juncture.

between microwaves and ionizing radiation (IR), which although a powerful mutagen (and a carcinogen), is a poor inducer of SCEs [5,6].

The technique of direct DNA analysis, which seems to monitor even minute sequence variation in the genome, therefore, appears to be a powerful technique for the assessment of the nonionizing radiation effects. Furthermore, based on the present study and other recent documents (7), what seems imperative is the (re)evaluation of the recommended safe limits of microwave exposure from the personnel and people who are exposed.

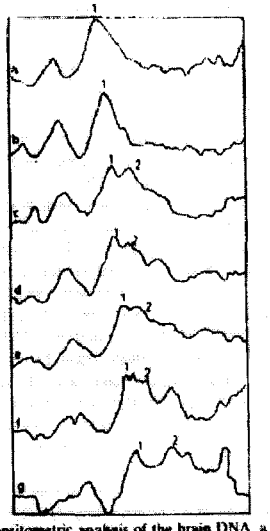

Fig. 3. Densitometric analysis of the brain DNA. a and b are control DNA; c–g are DNA from exposed animals. Peak 1 is present in both control and exposed animals while peak 2 appears in all the exposed animals.

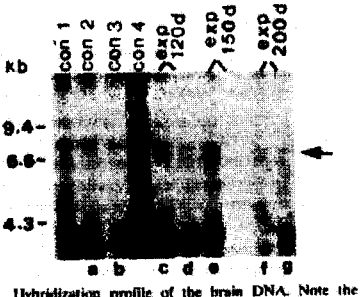

Fig. 1. Hybridization profile of the brain DNA. Note the appearance of a band below the position marked with an arrow in the exposed groups of animals. Lanes marked a–g are the ones which have been traced densitometrically as shown in Fig. 3.

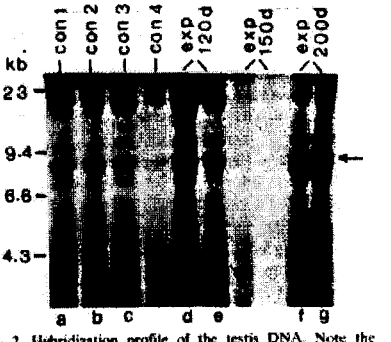

Fig. 2. Hybridization profile of the testis DNA. Note the broadening of the band width in the position marked with an arrow in the exposed animals. Lanes a–g are the ones which have been traced densitometrically as shown in Fig. 4.

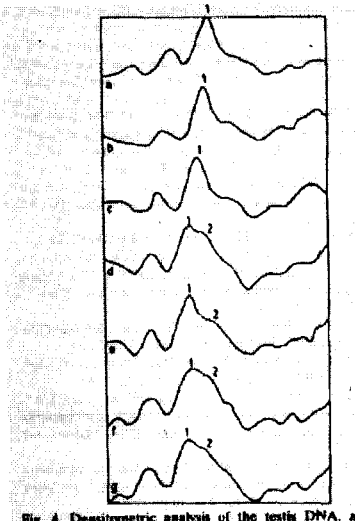

Fig. 4. Densitometric analysis of the testis DNA. a–c are control DNA; d–g are DNA from exposed animals. Peak 1 is present in both control and exposed animals while peak 2 appears in all the exposed animals.

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A REVIEW OF THE POSSIBLE EFFECTS OF 50/60 HZ ELECTRIC AND MAGNETIC FIELDS ON MELATONIN SECRETION

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INTRODUCTION

The possible relationship between exposure to extremely low electric and magnetic fields (EMF) and biological effects has become a public concern and a very active research topic. Many in vitro and in vivo studies have been carried out to explore the possible mechanism by which those fields can influence the fonctionning of living cells and whole animals. One of the ways in which electric or magnetic fields may effect animals and men is via melatonin.

ANIMAL DATA

In Rodents:

Electric fields: three kinds of reactions to 60 - Hz electric field have been described: reduction of pineal melatonin synthesis [(1)], reduction of serum melatonin due to an increase of degradation or of tissue uptake of melatonin [(2)], absence of effect of chronic exposure [(3)].

Magnetic fields:

	SPECIES	FIELD	FREQUENCY AND ORIENTATION	DURATION	EFFECT
YELLON (4)	Adult Djungarian hamsters (male and female)	100 µT	60 Hz	15 min 2 hours before darkness 'long days'	. 1st experiment - Nocturnal melatonin rise in pineal and serum: delayed January - Nocturnal peak: shifted and blunted . 1st replicate - Nocturnal melatonin rise in pineal and serum: reduced (6 months later) June - Nocturnal peak: blunted . 2nd replicate - No statistical differences between control and exposed hamsters January
YELLON (5)	Djungarian hamsters (male and female)	100 µT	60 Hz	15 min 2 hours before darkness 'short days'	. Nocturnal melatonin rise in pineal and serum: delayed
WILSON (6)	Djungarian hamsters	100 µT	60 Hz	15 min 2 hours before darkness	. Nocturnal pineal melatonin rise: decreased . Medial basal hypothalamus norepinephrine: increased
KATO (7)	Adult rats Wistar-King albino (male)	1 µT 5 µT 50 µT 250 µT	50 Hz circularly polarized	6 weeks	. Nocturnal pineal and plasma melatonin levels: decreased . No dose-response relationship
KATO (8)	Adult rats Wistar-King albino (male)	0.02 µT 1 µT	50 Hz horizontal or vertical	6 weeks	. No significant differences in pineal or plasma melatonin levels
KATO (9)	Adult rats Long-Evans (male)	0.02 µT 1 µT	50 Hz circularly polarized	6 weeks	. 0.02 µT : pineal and plasma melatonin levels: decreased at 12.00 h pineal melatonin level: not decreased at 24.00 h . 1 µT : pineal and plasma melatonin levels: decreased at 24.00 h and at 12.00 h
LOSCHER (10)	Adult rats Sprague-Dawley (female)	0.3 µT 1 µT	50 Hz vertical	12 weeks	. Nocturnal plasma melatonin level: decreased
SELMAOUI (11)	Adults rats Wistar (male)	1 µT 10 µT 100 µT	50 Hz Vertical	12 hours (short term) 4 weeks (long term)	. Short term exposure : 100 µT : plasma melatonin level and NAT activity : decreased (30 %) 1 µT-10 µT : no difference in serum melatonin NAT activity . Long term exposure : 1 µT : no difference 10 µT-100 µt : plasma melatonin decreased NAT activity decreased no dose-response relationship

60 Hz electric and magnetic fields in non rodent mammals

Lee [12] exposed female lambs aged 8 weeks to a mean electric field of 6 kV/m and a mean magnetic field of 40 mG (4 μT) for a year, by keeping them below electric transmission lines. He was unable to show a change in the nocturnal secretion of melatonin or any change in the age of the onset of puberty or estrus. This study have been replicated providing the same negative results.

Baboons [13] were exposed first to an electric field of 6 kV/m, with a magnetic field of 50 μT, and then to an electric field of 30 kV/m and a magnetic field of 100 μT. There appeared to be no alteration in the profile of the plama melatonin concentration during the experiment. However, when the type of exposure was changed to one with « rapid » field onset/offset, there was a reduction of about 15 % of the levels of melatonin previously observed during pre-exposure. These initial results require confirmation .

HUMAN DATA

The data for human subjects is not more consistent than the animal data. Wilson[14] fitted with conventional electric bankets, studied on volunteers the excretion of the principal urinary metabolite of melatonin, 6-sulfatoxy melatonin. He found no change in the urinary excretion of this metabolite. When continuous polymerwire (CPW) electric bankets, generating a magnetic field 50 % greater than conventional electric bankets, were used, 7 of the 28 volunteers showed a drop in their 6-sulfatoxy melatonin excretion during the exposure period, followed by an increase when exposure stopped.

Graham[15] exposed healthy volunteers to a 60-Hz magnetic field of 20 μT, resulting from an ON/OFF effect every 15 secs for one minute, repeated every other hour. The results indicate that there was no statistically significant difference between the exposed and control groups. The exposed subjects were separated, post hoc, into two sub-groups, one with a high, and the other with a low basal melatonin secretion. The response of the exposed subjects with the lowest basal melatonin secretion (< 60 ng/ml) was a reduction in the level of serum melatonin. A replicate experiment failed to find the results reported in the original one : there was no difference in the response to a magnetic field exposure between subjects with a lower basal level of melatonin and the subjects with a higher level. Touitou [16] exposed healthy volunteers to a continuous and intermittent 50-Hz magnetic field of 10 μT. The levels of serum melatonin and urine 6-sulfatoxy melatonin in exposed men did not differ significantly from those in control subjects.

CONCLUSIONS

A diminution and/or a retardation of the nocturnal peak of melatonin have been reported in rodents exposed to electric and/or magnetic fields. Experimental date from primates and man showed insufficient evidence of such a change.
Modification of melatonin secretion cannot be considered at the moment as a verified biological explanation for the epidemiologic studies that find an association between breast cancer or depressive disorders and occupational or residential EMF exposure.

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PAPER TITLE Analysis of mutagenic effect of low frequency electromagnetic fields fluorescence in situ hybridization (FISH)

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1 (see page 7)

ABSTRACT (See instructions overleaf)

Fluorescence in situ hybridization (FISH) has been shown to be a feasible technique of detecting chromosome rearrangements, being currently used for analysis of induced chromosomal damage.

The experimental evidence on genetic effects with 50-60Hz electromagnetic fields (EMF) indicate contradictory results; however, the majority of the reports failed to demonstrate adverse effects.

Blood peripheral lymphocytes from normal donors, phytohemaglutinin stimulated, were in vitro exposed to a 50 Hz EMF at a flux density of 300, 115, and 28 μT during the time of cultures for cytogenetic analysis (72h). After harvesting, following "chromosome painting" FISH procedures, microscope slides with chromosome spreads were hybridized with two whole-chromosome DNA-libraries and detected with fluorescein-labeled avidin. Under fluorescence microscope, stable chromosome aberrations such us deletions, and translocations, insertions, rings, and duplications were easily detected.

The FISH aberration scoring, under progress, are compared with the conventional light microscope scoring previously done which include, both, stable and unstable chromosomal mutations. With the current results we could conclude that even the use of more accurate techniques as FISH, continuous 50 Hz EMF are not able to produce chromosomal damage at the detectable level.

Acknowledgment: This work was made by grant (93/662) from Fondo Investigaciones Sanitarias.

EFFECT OF COMBINED IONIZING AND NON-IONIZING RADIATIONS ON THE EXPRESSION OF THE C-JUN ONCOPROTEIN

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INTRODUCTION

The question of the effects of extremely-low frequency electromagnetic fields (ELF-EMF) on public health remains very controversial. Epidemiological studies (1, 2) indicated a correlation between ELF-EMF exposure and the incidence of childhood leukemia. Nevertheless, laboratory investigations regarding ELF-EMF genotoxic effects in *in vitro* biological systems failed to evidence any genomic damage (3-6). Recently however, attention has focused on the potential role of electromagnetic signals in cellular promotion implied in carcinogenesis process.

For instance, the transcription of oncogenes (7,8) has been reported to be responsive to ELF-EMF and the number of transformed foci in cells treated with a phorbol ester have been demonstrated to be enhanced after magnetic field exposure (9). Furthermore, cells irradiated with ^{60}Co showed increased chromosomal aberrations when they were exposed to ELF-EMF (10). Recently it was demonstrated that the number of ionizing radiation-induced micronuclei raised after ELF-EMF exposure in two of the three rat tracheal cell lines tested (11).

The aim of this study was to determine if ELF-EMF may induce the expression of the oncoprotein c-JUN, involved in cell transformation, either alone or after preexposition to gamma rays.

MATERIAL AND METHODS

Cell culture

Both cell lines used were obtained from gamma rays-irradiated rat tracheal cells in primary culture or from spontaneous immortalized clone.

Cells were maintained in HD medium (3v. Ham F12/ 1v. DMEM) with 1% of decomplemented fetal bovine serum at 37°C in a 5% carbon dioxide atmosphere.

Radiation exposures

Electromagnetic fields were generated with 2 pairs of Helmholtz coils put in a plastic box shielded with mu-metal and placed in a 5% CO_2 incubator. The electromagnetic signal consists of a sinusoidally varying field with a frequency of 50 Hz and a 0.1 mT r.m.s. amplitude, combined with a 0.05 mT static geomagnetic-like field (horizontal and vertical components).

EMF sham-exposures were performed in a similar mu-metal shielded box, without Helmholtz coils and placed in another incubator where measured background sinusoidal field was < 0.001 mT.

Confluent tracheal epithelial cells were gamma-irradiated (^{60}Co, 6 Gy, 0.8 $Gy.min^{-1}$) and control cells were mock-exposed. A uniform time of 1 hour elapsed from the irradiation time to the initiation of continuous incubation in the sham- or ELF-EMF exposure system.

Western blotting

Adapted from (12), cells were lysed 1.5, 4, 6 and 10 hours after ionizing irradiation, extracted and 10 µg protein samples were separated by SDS-PAGE on a 12% polyacrylamide gel and blotted onto a PVDF membrane. c-JUN protein was assayed using a polyclonal c-JUN/AP1 antibody (Santa Cruz Biotechnology) and the enhanced chemiluminescence (ECL) detection procedure (Amersham).

RESULTS

Experiments were done in triplicate and typical results are shown below. Two bands are usually observed and the major one was taking into account for data interpretation.

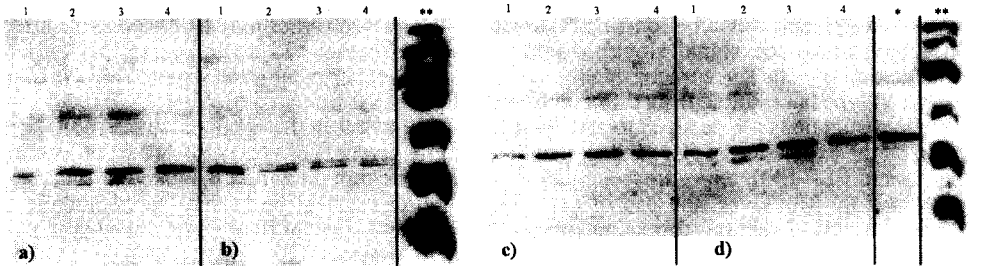

Figure: Expression of c-JUN in an epithelial tracheal cell line, after ionizing and/or non-ionizing irradiation. Cells were harvested 1.30h (**a**), 4 h (**b**), 6h (**c**) and 10h (**d**) after γ rays exposure.
(lane **1**): 0 Gy+Sham EMF; (lane **2**): 0 Gy+EMF;
(lane **3**): 6 Gy+Sham EMF; (lane **4**): 6 Gy+EMF.
*: positive control; cells were treated by Phorbol 12 myristate 13 acetate (PMA, 100 ng/ml, 1 hour).
**: molecular weight markers (97.4 - 68 - 49 - 31 - 20.1 - 14.4 kDa).

The preliminary results, obtained with a spontaneous immortalized cell line, seemed to indicate that the level of JUN protein was significantly increased in cells exposed continuously during 6 and 10 hours to ELF-EMF after a gamma rays irradiation (6 Gy) versus untreated and only gamma-irradiated cells. In this previous experiment, cells were not exposed to ELF-EMF alone, because of material constraints.
Data concerning another cell line failed to reproduce these results. Indeed, no synergistic effect of ionizing and electromagnetic fields was observed regarding the induction of JUN protein. However, under our experimental conditions, ELF-EMF alone enhanced the induction of this oncoprotein in a quite similar way as gamma rays. The figure shows that c-JUN is early induced (30 minutes after ELF-EMF exposure and 1.30 hour after ionizing irradiation) and durably expressed until 10 hours of exposure.

DISCUSSION

ELF-EMF have been recently suggested to act as promotor agents in carcinogenesis processes (9-11,13). As overexpression of protooncogenes and their products is involved in epigenetic mechanisms (14), we examined the ability of two epithelial tracheal cell lines to synthetize the c-JUN protein in response to ELF-EMF exposure, with or without a previous ionizing radiation acting as an initiator.

c-jun belongs to the early response genes family and is known to be induced until several hours after ionizing radiations (15) and enhancement in the transcription of oncogenes including c-jun was previously reported in T CEM-CM3 lymphoblastic cells after 1 hour exposure to a very similar electromagnetic signal, excepted the absence of the static field (7). The same group recently indicated that the same signal could increase the transcription of the ras oncogene (16). EMF were also demonstrated to alter polypeptide synthesis in salivary gland cells, with an increased expression of several non-identified proteins (17). Furthermore, messengers such as Ca^{2+} or inositol triphosphate, involved in signal transduction pathways and leading to the transcription of genes such as c-jun, have been reported to be responsive to ELF-EMF (18,19).

Rat tracheal cell lines are usefull in transformation assays using chemical agents (20) and have been demonstrated to be responsive to ELF-EMF after gamma irradiation. A synergistic effect of the combination of gamma irradiation and ELF-EMF exposure, depending however on the cell line tested (two of the three cell lines used were responsive), has been observed concerning micronuclei induction but no effect of ELF-EMF alone was detectable (11). In this study, data obtained in a spontaneous immortalized tracheal cell line suggested such a synergism on the delayed induction of c-JUN expression. In a second cell line isolated from irradiated primary epithelial cell culture, no synergistic effect was detectable but ELF-EMF exposure alone appeared having an effect on the

early induction and delayed expression of JUN protein. Being assumed that both cell lines used behaved in a different way after ionizing treatment, data reported earlier may be linked to the effect of the electromagnetic signal alone. Moreover, these effects could be linked to the level of transformation of cells since tracheal cell lines used in this experiment were issued from the same primary cell type but resulted from different treatments.
Further investigations are needeed to ascertain ELF-EMF effect in the spontaneous immortalized cell line on the expression of JUN and to elucidate the effectiveness of synergistic effect of ionizing radiation and ELF-EMF exposure in induction of oncogene protein which is related to act in cell transformation. Moreover, the use of primary epithelial tracheal cells may be of interest to determine the role of cell transformation level in the biological effect of ELF-EMF alone which may emphasize the hypothesis of ELF-EMF promotor effect.

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PAPER TITLE Resonance effect of weak ELF electromagnetic field on E. coli cells and its dependence on the genome structure

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1 (see page 7)

ABSTRACT (See instructions overleaf)

The cellular effects of weak electromagnetic fields of extremely low frequency (ELF, 1-300 Hz) have been extensively investigated. Some effects of ELF are questionable due to poor reproducibility. One possible reason for this could be dependence of resonance frequencies on genome structure of cells under investigation. We tested whether the resonance ELF frequencies can be different for mutant and wild-type strains of *E. coli* K12. The effect of weak ElF, 30 μT magnetic flux density, on the genome conformation was studied by the method of anomalous viscosity time dependencies in the 6-68 Hz frequency range. We observed four frequency windows with resonance frequencies of 8.9 Hz, 15.5 Hz, 29.4 Hz, and 62 Hz when exposing the K12 AB1157 cells which had a few genetic markers. In the same frequency range the wild-type K12 EMG2 cells had only three effective windows with resonance frequencies 8.3 Hz, 27 Hz and 56 Hz. The resonance frequencies were different significantly ($p<0.001$-0.000001) in studied strains. It was shown, that at the 8.9 Hz resonance frequency the ELF affected the rate of cell division which was measured by means of dilution plating and protein/DNA synthesis, measured by incorporation of ^{3}H-Lucite/^{3}H-thymidin. Using electrophoresis, we observed the 8.9 Hz-induced changes in the number of proteins tightly bound to DNA. Conclusions: 1) Weak ELF fields which were comparable in magnitude with those of typical electrical devices affected *E. coli* cells at the selected frequencies. 2) These effective frequencies were different for two *E. coli* strains which distinguished genetically.

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Mini-Presentation

PAPER TITLE ELECTROMAGNETIC HYPERSENSITIVITY

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MAJOR SCIENTIFIC TOPIC NUMBER 7.2.... (see page 7)

ABSTRACT (See instructions overleaf)

The discussion on the existence and relevance of hypersensitive reactions to weak electromagnetic fields suffers from the lack of a generally accepted definition of the phenomenon. Based on the assumption that relevant electromagnetic biological interaction is mainly based on induced intracorporal current densities, perception thresholds of directly applied 50 Hz currents have been measured on a statistical sample of the general population by a mobile measurement setup and double blind experiments. This results have been compared with magnetic field provocation experiments on normal and hypersensitive persons. The results are presented and discussed.

SICKNESS ABSENCE FROM 1978 TO 1992 OF ELECTRICITE DE FRANCE (EDF) WORKERS EXPOSED TO ELF-EMF : The substation group

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INTRODUCTION

Electromagnetic fields are suspected of many health effects, including cancers, immunity and fertility disorders, congenital malformations, suicides and depression...(1-2). Although the results of all studies are contradictory and no biological mechanism has been found, populations are concerned about this matter.

This study analyses the absence from work of some EDF employees who are *a priori* the most professionally exposed to ELF EMF. Absence from work is taken as an health indicator. Although it is not specific, it can be predictive of serious health problems (3). The specific context of EDF has made it possible to collect and analyse the medical aspect of absence from work, that is rare in France.

We were interested with employees working with high voltage (over 90 kV). Three job categories have been selected : live line workers, substation maintenance workers and substation exploitation workers (called in the company the ATEX). The first group concerned the live linemen and the results were exposed previously (4,5). It appeared they had more accidents at work, but this was more related to the job of lineman rather than to the EMF exposure. Nevertheless, substation workers have an exposure less intensive but longer than live linemen. We present here the result about the absenteeism of the substation workers, maintenance and exploitation.

OBJECTIVES

The objective of this retrospective longitudinal study is to describe the profile of absence from work for medical reasons for the substation workers, maintenance and exploitation, over a period of 15 years (1978-1992), and then to compare it with a representative sample of EDF GDF employees, of same age and sex, and for the same period (reference group).

Cancer pathology is not the aim of the study, because of its rarity in this young selected population. But we were particularly attentive to the psychiatric disorders.

POPULATION

The EDF generation and transmission department, which deals with all the employees concerned by high voltage, have 34 operational sub-groups. Each of them includes :

- a maintenance substation team, also about 15 persons. That is about 500 persons each year concerned by the study,
- four or five substation grouping, each with 7 or 8 ATEX (substation exploitation workers). That is about 1100 persons each year concerned by the study.

The substation maintenance workers are exposed to EMF, with an average occupational exposure of about 35 µT. The ATEX may be particularly exposed, especially in term of duration, when they live at their place of work. This was the case for two-thirds of them in 1992.

METHODOLOGY

To be included in the study, workers must be men, having worked for at least one year accumulated either in a live line team, a substation team or as ATEX, in the EDF generation and transmission department, between 1978 and 1992. A subject remains in the exposed group even if he leaves the exposed job. The absenteeism is taken into account up to 31 December, 1992 or to his departure of the company, by retirement or by resignation (but this event is rare, 0.05% per year). All employees who have left the exposed job before 1978 have been excluded from the study.

The substation group included the substation workers (maintenance and exploitation) and subjects having worked either simultaneously or consecutively on both lines and substations.

A reference group had been chosen, at random, from all the EDF GDF non-management male employees, one reference subject for each exposed worker, matched on the first year of employment.

For each person included in the study, his job history and his socio-demographical data for 1993 were extraxted from the employees data base. The epidemiological data base of the Service Général de Médecine de Contrôle, who manages the social security of the company, supplied the absence data : number of absences with the date, duration of absences, and diagnoses if they are known. Mortality data (death during occupational life), long term illness and invalidity data were also available.

The socio-professionnal data were analysed using *Epi Info* software in the Service des Etude Médicales, to compare the exposed and the reference groups. The absenteeism data were analysed using *SAS* software in the Service Général de Médecine de Contrôle, with the method of person x years. The tests used are *Chi2* and *Student*.

RESULTS

The substation group is composed of 2327 persons exposed from one year or more, divided into 685 maintenance substation workers, 1121 ATEX and 521 mixed workers (maintenance and/or exploitation). Among these exposed workers, 243 (10.4%) no longer had an exposed job in 1992, and 289 (12.4%) had left the company.

Their average length of work in the company was 17.8 years (from 1 to 44 years) and their average length of exposure was 11.8 years (from 1 to 37 years). These 2327 persons correspond to 22168 persons x years.

In 1993, the socio-demographical data, which could influence the absenteeism, were comparable between the exposed and the reference group : they had the same mean age (41 vs 41.3 years old) and a similar distribution into the different matrimonial situations (unmarried, married or co-habited, separated or divorced, widowed). The exposed workers had more children than the reference workers, but this factor concerns more the female absenteeism than the male absenteeism.

The absenteeism of the exposed group is significantly less important than that of their referees ($p<1‰$) (table 1) : there are less substation workers who stop work (34.5% vs 38.9%), they are absent less often (1.46 vs 1.67 absence in a year) and their absences are shorter in duration (21 days vs 23.3 days). But the absenteeism is different if substation maintenance or substation exploitation workers are considered : substation exploitation workers (ATEX) have a less significant absence rate than their referees (1.45% vs 2.33%); substation maintenance workers have an absence rate over their referees (3.03% vs 2.6%) because there are more exposed people who take sick leave but they take it less often than their referees.

The medical diagnoses (table 2) are also significantly different : sick leaves for psychiatric (2.1% vs 4.1%) and respiratory (13.4% vs 15.3%) diseases are less frequent in the exposed group than in the reference group. On the other hand, sick leaves for accidents at work are more frequent in the exposed population (7% vs 4.5%).

CONCLUSIONS

The substation workers have better health indexes than their referees, and it is not very surprising because they are carefully selected to do this job. But the differences are less important between the substation workers and their referees than those between the live linemen and their referees.

No pathology, which has been described as possibly linked to electromagnetic fields, has appeared through the medical absenteeism surveillance in our study.

Tables 1 : Absences variables

	exposed	referees	p
Absence rate	1.98%	2.49%	p<1‰
Percentage of employees absent at least once	34.5%	38.9%	p<1‰
Average annual number of sick leaves per employee absent	1.46	1.67	p<1‰
Average duration of an sick leave	14.4 days	13.9 days	
Average annual duration of sick leave per employee absent	21 days	23.3 days	p<1‰

Table 2 : Medical conditions (number of absences during the study period)

	exposed (%)	referees (%)
Psychiatric	**239 (2.1)**	**588 (4.1)**
Respiratory	**1490 (13.4)**	**2205 (15.3)**
Digestive	436 (3.9)	627 (4.3)
Cardiovascular	168 (1.5)	201 1.4)
Osteoarticular	743 (6.7)	1032 (7.2)
Accidents at work	**778 (7)**	**653 (4.5)**
Accidents outside work	875 (7.8)	1060 (7.4)
Urinary and Genital	125 (1.1)	189 (1.3)
Other diagnoses	675 (4.9)	1049 (6)
Diagnosis unknown*	5745 (51.5)	6966 (48.4)
Total	11149 (100)	14381 (100)
		p < 1‰

*Sickleaves non controlled by the doctors of the Service Général de Médecine de Contôle

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WEAK MAGNETIC FIELDS INJURIOUS TO HEALTH, STRONG MAGNETIC FIELDS HARMLESS ? RADIATION PROTECTION BY THE PRESENT OF MAGNETIC FIELDS.

DIPL.ING.ERICH L. H U B E R, VIENNA

Usually magnetic fields are part of the environment without making injuries to health. Only when limits in standards were fixed the certainty become conscious that electromagnetic fields in their various forms must be hazardous. The effects of the pure magnetic fields cannot be found out easy because it is difficult to screen the magnetic fields, especially the magnetic field of the earth. This analysis shall also find out how to hold limits by using extremly high magnetic fields in medicine and research. The results show that screening is not the only method when the praxis requires behaviour where screening is not possible.

GESUNDHEITSSCHÄDLICH BEI GERINGEN, UNGEFÄHRLICH BEI STARKEN FELDSTÄRKEN ? STRAHLENSCHUTZ BEIM AUFENTHALT IN MAGNETFELDERN

Immer häufiger treten Magnetfelder in unser Gesichtsfeld, ohne daß eigentlich Schäden in gravierender Weise auftreten, von denen Menschen unmittelbar betroffen sind. Erst durch normative Festlegung von Grenzwerten wurde zwingend bewußt, daß gewisse gesundheitliche Beeinträchtigungen auch durch elektromagnetische Felder hervorgerufen werden können. Inwieweit das magnetische Feld daran beteiligt ist, läßt sich nur für bestimmte Effekte begründen. Wie man den Forderungen nach Einhaltung dieser Grenzwerte in extrem hohen Magnetfeldern gerecht werden kann, soll dieser Beitrag untersuchen, wobei die Anwendung von Untersuchungsmethoden in Medizin und Wissenschaft an erster Stelle stehen soll. Es ergibt sich dabei, daß die Abschirmung nach außen keineswegs ausreicht, um die Forderungen von Normen und Richtlinien zu erfüllen, sondern daß es zu einem großen Teil auch darauf ankommt wie Untersuchungen durchgeführt werden.

To realize the hazards of magnetic fields first you must know the parameters of the magnetic fields which can cause harms. Magnetic fields are not directly observable, but their effects are very evident. The main magnetic term is the magnetic induction B measured in tesla (T) / SI unit and gauss (G) /CGS unit,where 1 nanotesla (10E-9 tesla) = 1 gamma (10E-5 gauss). These dimensions are first used in the measurement of the Earths magnetic field, whose attributes mainly touch peoples health. But because nobody can escape from the Earth magnetic field we must try to know the hazards of artificial fields. In any case the whole electromagnetic spectrum is responsable for the welfare of the human being. The question is whether the ionizing energy is only responsible for stochastic changes which leads to cancer deseases or if the nonionizing part of the electromagnetic spectrum also can be made responsible for such health defects. The primary modes of action of non-ionizing radiation is either photochemical or thermal. Photochemical reactions

primarily occur upon absorption of radiant energy in the ultraviolet and visible portions of the spectrum. Thermal effects are produced primarily upon absorption of visible, infrared, and microwave radiation. In biological systems the energy transfer produces electron excitation, which can result in dissociation of the molecule if the bonding electrons are involved, dissipation of the excitation energy in form of fluorescence or phosphorescence, the formation of free radicals and degradation into heat.
Looking at radio frequency and microwave radiation between 10 MHz and 300 GHz we learn that microwaves interact with matter primarily through the conversion of electromagnetic energy to potential molecular energy in the absorbing medium, where dissipation of the potential energy results in the production of heat. The depth of penetration of microwaves into the body is approximately equal to 0.1 of the wavelengths between 3 cm and 200 cm.
But the avascular organs as the eye, the hollow viscera and the testes are of primary interest, because they do not have a high capacity for removing heat through the circulation of blood. The interaction with the central nervous system and the phenomen of „pearl chain formation" become also aligned in a specific pattern in a microwave field.
The electromagnetic waves in this sphere result from electric fields, measured in Volt/ Meter and magnetic fields, measured in Ampere / Meter. Being in the far field region we can use the power density S measured in Watt/m2 to characterize the quality of the electromagnetic field of antennas. The intake of electromagnetic energy into biological systems is given through the specific absorption rate SAR in W/kg or the specific absorption SA in Joule/kg for pulsed radiation. Limits are given in new standards for instance ÖNORM S1119 and S1120 over the range from electro- and magnetostatic fields until 30 kHz (ELF Extremely low frequencies and VLF Very low frequencies),and from 30 kHz to 3000 GHz.
The classification of electromagnetic waves often overlap and the discription of a wave in terms of energy, wavelength, or frequency differs from region to region. Gamma rays and x rays generally are discribed in terms of photon energy; ultraviolet, visible light and infrared radiations in terms of wavelength; and radio frequencies in terms of frequency. The term „ionizing radiation" emphasizes the minimum photon energies capable of producing ionization, dislodging orbital electrons and produce ion pairs. Therefore is the lower limit for ionization effect from the standpoint of biological systems 12 electron volts. Lower frequencies or higher wavelengths characterize the term of „non-ionizing radiation".
A very important point of view is the perception of radiation risk especially if events with high consequences and low probability occur. Effects from ionizing radiation represent around 1% of the global burden of disease. This small contribution is dominated by natural radiation, medical care of patients, and fallout from nuclear weapons test. Industrial radiation sources contribute less than one-hundredth of 1%. Nevertheless it is not possible to say with cercainty that a particular cancer victim has died for instance from radon exposure. Therefore the public generally believes the risks are lower than scientific estimates would substantiate. The correct scientific method is the a priori

investigation of putative sources of risk and this carried out for example that studies of childhood leukaemia around nuclear installations reveal little,if any, evidence for an increased risk of it. But this does not help much when I want to know if the probability if the cancer is caused by radiation effect is more than 50 % or not. Only on this condition there is a chance of compensation. This would be legal either for ionizing radiation and for non-ionizing radiation.
The first step in the discussion between cancer disease and electromagnetic fields is done. But the chance to use the results of epidemiologic studies as an exhibition is very small, nearly zero. The advocates of this cancer-field theory examinated 50 Hz- magnetic fields with inductions of 0.3 mikrotesla. Known is only the muscle excitation with magnetic fields about 0.8 tesla and timegradient of about 120 tesla/second. However the causal connection between cancer and magnetic field strength of about 0.3 mikrotesla could not be found hitherto.
The development to examine biological effects of very weak magnetic or electric field with extremely and very low frequencies similar to those of natural background as o.o3-0.077 mikrotesla and 2.5 V/m does not lead to useable statements but gives assumptions of further work.
Of more interest is the biological effect of static magnetic fields over a long period, but the exposure in 3-10 millitesla over 10 years could not demonstrate any health effect. For magnetic resonance imaging there are used magnetic field strength of about o.5,1 and 1.5 tesla and the surrounding line inside the examination room has to be lower than 0.5 millitesla. So we see that outside of the examination room will no dangerous static magnetic field occur.Although the praxis of examination shows that physicians will be exposed inside the 0.5 millitesla line close to the patient the propability of health risks will be small. How far 1.5 tesla are injurious to the health of the patient is unknown till now. The same must be said of other magnetostatic facilities used for research with field strength up to 19 tesla for instance. But in this case measurements can be handled in the manner that scientists are not touched by dangerous influences.
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SURVEY OF NON-IONIZING RADIATION LEVEL IN THE JRR-3M

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ABSTRACT

Measurements of the free space levels of magnetic field in the Japan Research Reactor No.3 Modified (JRR-3M) and computer simulation of generating electromagnetic fields are presented. Extremely low frequency (ELF:50Hz) and very low frequency (VLF) field levels in the JRR-3M were fairly lower than applicable guidelines or standards. Simulation of generating electromagnetic fields would suggest that the generators were electrical devices, such as semiconductor power converter and video display terminal (VDT). These devices could produce not only ELF but also VLF electromagnetic fields.

INTRODUCTION

There has been a considerable development of science such as accelerator, superconducting magnetic energy storage (SMES) and nuclear fusion. Now VDTs are widespread among the general public in Japan. These man-made sources have increased chance of occupational and public exposure to non-ionizing radiation (NIR). The International Radiation Protection Association (IRPA) created the International Non-Ionizing Radiation Committee (INIRC) in 1977, and published some reports as to NIR(1).

In Japan, public concern has been growing about possible adverse effects of low frequency electromagnetic fields on human health. The Ministry of Posts and Telecommunication (MPT) indicated guidelines on protection standards for exposure to NIR in 1990(2). However, there are not many scientific data in Japan for the Japanese guidelines on limits of exposure to NIR. It is very important to measure magnetic field levels in workplace and home. There are also concerns about electromagnetic interference and compatibility (EMI/C) of electromagnetic fields. In the nuclear facilities, electronic personal dosimeter is adopted for personal dose measurement for the workers at radiation controlled areas, but the electronic personal dosimeter is not always independent of electromagnetic fields. For the purpose of personal dose monitoring for workers, it is necessary to measure the normal levels of magnetic field.

In the present work, measurements and simulation were carried out so as to obtain knowledge of NIR risk.

METHODS

Measurements of magnetic flux densities were made in the JRR-3M while it was not in operation. The JRR-3M attained its first criticality in March 1990 as a high-performance multi-purpose research reactor and started operation with maximum output of 20MW in November of the year. There are various utilization facilities installed in the JRR-3M, magnetic fields caused by these facilities influenced not only radiation workers but also instruments.

Two types of instruments were used for measuring ELF magnetic flux densities and VLF magnetic field strengths. One is ELF/power frequency EMF survey meter MODEL HI-3604 (HOLADAY) which has 6.5 inch diameter 400 turn electrically shielded magnetic field sensing coil. Its sensitivity for magnetic fields ranges 0.1mG-20G. The other is VDT/VLF radiation survey meter MODEL HI-3603 (HOLADAY) which has 8 inch diameter magnetic field sensing loop. Its

sensitivity for magnetic fields ranges 1-1999mA/m. Magnetic fields were measured along three orthogonal axes, since both instruments did not have three orthogonal field sensors. The magnetic flux densities and magnetic field strengths were expressed by the root of the sum of these mean squared mutually orthogonal components. All the measurements were taken at a height of 1m above the floor except for some cases. Temperature and relative humidity in the JRR-3M were 18.8℃ and 54.0%, respectively.

Recently there has been widespread VDT in workplaces and homes. A survey was made of VDT. This model is 15 inch cathode ray tube (CRT). The vertical deflection frequency is 56.4Hz, the horizontal deflection frequency 24kHz.

Root mean square (RMS) electromagnetic fields were determined by measurements as variation of the distance from centre of the VDT's surface. In order to reduce the proximity effect, the distance between the electromagnetic field strength meter and operator had to be at least 2.5m. Measured time period for ELF electromagnetic fields was over 1 second, for VLF fields 6 minutes. VLF electromagnetic field strengths were averaged over 6 minutes. The VDT's brightness was adjusted to an average level. The survey meter and the VDT were mounted on each non-conductive tripod.

There has been a tendency for fluorescent light and air conditioner etc. to be equipped with inverter which would produce electromagnetic wave. A computer simulation was performed for a double-way rectifier circuit using PSPICE (P Simulation Program with Integrated Circuit Emphasis, MicroSim Co.) so as to make clear the sources of electromagnetic wave. Simulated circuit consisted of four semiconductor controlled rectifiers, two FETs and SMES, and was controlled by pulse width modulation (PWM). Electromagnetic fields generated by semiconductor were estimated by Fast Fourier transform (FFT) of alternating-current.

RESULTS AND DISCUSSION

As shown in Figure 1, the magnetic field levels measured in the JRR-3M are low. The maximum magnetic flux density at the near surface of the electrical devices is fairly lower than applicable guidelines. These data were found by measurements at the distance 10cm from devices' surface. There are relatively high levels of magnetic field of vacuum device for a adiabatic tub and automatic voltage regulator which contain semiconductor elements. Observation of the magnetic fields using oscilloscope indicated that 50Hz frequency and higher harmonics of the magnetic flux densities were generated from electrical devices. As for high frequency, harmonics of the magnetic flux densities due to frequency-conversion at the electrical devices were observed.

Figure 2 shows ELF electric field strengths generated by VDT. The distances between the survey meter and the VDT are 0.1,0.2,...,1.0m. Table 1 shows the electric and the magnetic fields at 10cm from the VDT's surface and IRPA guidelines. Using these field strengths value, induced current density in human head was found from calculation. Its current density was 0.6mA/m^2 as worst case assuming a 7.5E-2m radius loop and conductivity 1S/m. Normally,, the spontaneous endogenous current density in body is about 10mA/m^2. The health risk caused by electromagnetic fields of VDT is not necessarily considered.

Computer simulation indicated that electronic circuit consisted of nonlinear elements and controlled by PWM emitted not only ELF but also VLF electromagnetic wave.

CONCLUSIONS

[1] The magnetic field levels measured at JRR-3M were fairly lower than IRPA and Japanese guidelines. VDT electromagnetic fields were also lower than these guidelines.

[2] Electrical devices, such as semiconductor power converter and VDT could produce not only ELF but also VLF electromagnetic fields.

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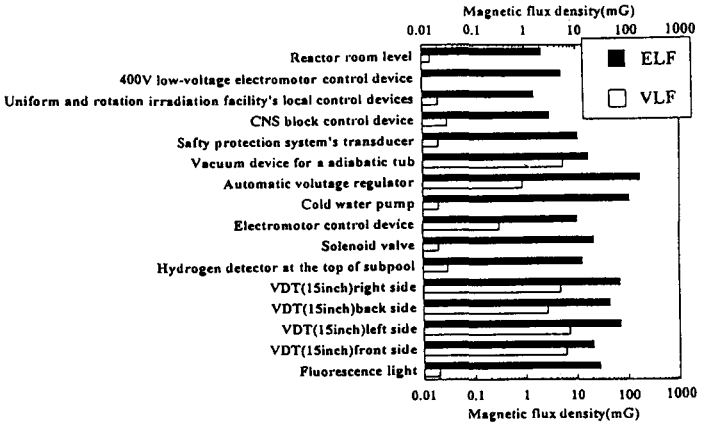

Figure 1. Magnetic flux density produced by electrical devices.

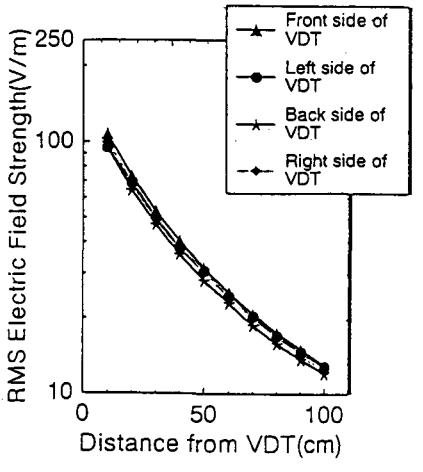

Figure 2. Electric field strength(ELF) variation with distance from VDT.

Table 1. Electric and magnetic field levels, IRPA Guidelines and Japanese Guidelines

	measured levels*)	IRPA	JAPAN(P)
ELF electric field strength (V/m)	9.5E+1~ 1.1E+2	1.0E+4	
ELF magnetic flux density (mG)	2.1E+1~ 7.2E+1	5.0E+3	
VLF electric field strength (V/m)	3.3 ~ 6.0E+1		6.1E+2
VLF magnetic field strength (mA/m)	2.1E+2~ 5.8E+2		1.6E+5

*) at 10cm from VDT.

NON-IONISING ELECTROMAGNETIC FIELDS ON OFFSHORE INSTALLATIONS

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INTRODUCTION

The concern over the effects of occupational exposure to non-ionising electromagnetic fields (EMF) has greatly increased in recent years. A great deal of knowledge is known about the thermal effects of radiofrequency EMF's and at the moment, many epidemiological and laboratory studies are being performed on extremely low frequency (ELF) and very low frequency (VLF) EMF's. Some studies have reported an increased risk of leukaemia and other cancers in children living close to overhead power cables and power industry electrical workers.

Wertheimer and Leeper reported cancer links in children residing near overhead power cables as early as 1979 (1) and many subsequent studies have continued to make similar associations (2,3,4,5 and 6). These studies suggest that prolonged exposure to higher than normal magnetic fields increases the occurrence of certain cancers in both children and adults. The most common associations are between EMF's and leukaemia, other haematopoetic cancers, brain cancers, central nervous system cancers or melanomas.

Studies of adults living near overhead lines by Youngson et al (7)) and working in the electricity industry by Armstrong et al (8) and Savitz & Loomis (9) have also shown associations with certain cancers. The epidemiological studies are incomplete in several areas and many have been openly criticised. As yet, there is no conclusive laboratory evidence but studies are ongoing (10,11,12). The Hendee and Boteler study (12) suggested that " EMF's might be cancer promoters but are unlikely to be cancer initiators".

In addition to ELF studies, there have been many reports investigating exposure to EMF's from visual display units (13) with equivocal results. Laboratory studies have reported conflicting results and as yet the hazard, if any, is still uncertain (14). Reports have also recorded exposure levels of operators in broadcast radio stations showing a variety of levels dependant on the occupation (15).

In December 1992, the Commission of the European Communities proposed a council Directive on the minimum safety and health requirements regarding the exposure of workers to the risks arising from physical agents including electric and magnetic fields at frequencies up to 300 GHz (16). The proposed Directive contains a set of ceiling levels expressed as fundamental dosimetric quantities and action levels expressed in terms of electric field strength, magnetic field strength and power density.

On offshore oil producing installations there are many sources of electromagnetic fields operating at a variety of different frequencies. These include transformers and generators capable of large power generation, numerous switch rooms, radiofrequency communications including microwave and satellite links. Due to the confined work areas, the switch rooms and transformer rooms are frequently used as office areas by engineers and technicians. This paper the results of these fields on several offshore platforms and compares the measured fields with the proposed CEC directive.

METHODS

Three different field strength meters and a VDT adapter were used for the measurements to cover the broad frequency range available on offshore installations. These include the Holaday HI-3600-02 power frequency field strength meter used for measuring both electric and magnetic fields in the frequency range 30-300 Hz. The Holaday HI-3600 VDT Radiation Survey adapter fits onto the aforementioned field strength meter and is sensitive to electric fields between 2 - 300 kHz and to magnetic fields between 8 - 300 kHz. The Holaday HI-3000 broadband isotropic field strength meter uses two probes. The electric field probe is sensitive to frequencies between 0.5 MHz - 5 GHz and the magnetic field probe between 5 - 300 MHz. The higher frequency RAHAM Model 4 Isotropic Wideband Electromagnetic radiation hazard meter measures the electromagnetic field in terms of power density in the 10 MHz - 26 Ghz range.

RESULTS

The results clearly fit into three different categories, including power generation and distribution system measurements, VDU measurements and telecommunication equipment measurements. The maximum electromagnetic fields from the power production were summarised and are tabulated below :

Area	Maximum Magnetic Field (μT)	Maximum Electric Field ($V.m^{-1}$)
Platform A	181.95	120.0
Platform B	227.98	651.0
Platform C	2590.43	31.0
Platform D	816.15	522.0
Platform E	532.67	1400.0

Table 1 Summary of maximum levels of magnetic and electric fields at 60 Hz found on offshore platforms.

The maximum levels measured were typically found close to the transformers and the incoming supplies from those transformers to the distribution boards and were all found in the switch rooms and transformer rooms of the oil processing package. From the table, the maximum level of magnetic field found throughout the platforms measured was 2.6 mT on Platform C. This exceeds the first action level of 333 μT recommended by the CEC for this frequency as do levels on Platform D and Platform E. The levels found on Platform C not only exceeds the first action level but also the hazardous activities level of 1 mT for this frequency. The maximum electric field of 1.4 $kV.m^{-1}$ found on Platform E near a VDU screen, is within the recommendations by the CEC (10 $kV.m^{-1}$).

At the time of measurements, the load on the drilling package was minimal on all but one of the platforms (platform C) and the subsequent levels were lower than those in processing. The maximum magnetic field found in the drilling package was found on platform D and was 129.92 μT. Power is produced at 6.8 or 11 kV for all the processing applications and at 600 V for the drilling applications. Induction welding is used frequently on offshore installations and levels were found to be no greater than 9.24 μT.

The maximum ELF electric and magnetic fields near VDT's found on the platforms are summarised and presented below in table 2.

Position	Electric Field Strength ($V.m^{-1}$)	Magnetic Field Strength ($mA.m^{-1}$)	Magnetic Flux Density (nT)
Platform A	120	439	553
Platform B	138	2014	2530
Platform C	16	444	558
Platform D	102	2411	3030
Platform E	1400	1852	2327

Table 2 Summary of maximum ELF measurements in front of VDU screens on offshore platforms.

Throughout the platforms, the maximum magnetic field measured near a VDU was found to be 3030 nT (3.03 μT) and the maximum electric field was 1.4 $kV.m^{-1}$. The maximum magnetic field measurements are much lower than those found in the switch rooms and are well below the recommendations by the CEC of 333 μT for the magnetic flux density.

Similarly for VLF magnetic and electric fields, the CEC draft proposal recommends that the magnetic flux density does not exceed 20 μT and the electric field does not exceed 614 $V.m^{-1}$ for the frequency range 1-300 kHz. Again the results are summarised and presented below in table 3.

Position	Electric Field Strength ($V.m^{-1}$)	Magnetic Field Strength ($mA.m^{-1}$)	Magnetic Flux Density (nT)
Platform A	94	250	314
Platform B	37	411	516
Platform D	81	358	450
Platform E	8	46	58

Table 3 Summary of maximum VLF measurements in front of VDU screens on offshore platforms.

Clearly the VLF levels found near VDU's are well within the recommendations by the CEC even those measurements taken close to the screen. The maximum levels from the telecommunications equipment on the four platforms are presented below in table 4.

Platform	Electric field strength ($V.m^{-1}$)	Magnetic field strength ($mA.m^{-1}$)	Magnetic flux density (nT)	Power Density ($mW.cm^{-2}$)
Platform A	N	77	97.33	20
Platform B	N	N	N	2
Platform D	141.42	510	640.74	N
Platform E	316.20	390	486.68	N

Table 4 Summary of maximum radiofrequency measurements found on the platforms where N is negligible and approximately equal to the background.

The majority of the maximum levels were found near the outside aerials but on platform E, maximum levels were found near some of the handheld radiofrequency sources. The maximum electric field strength was found to be 316 $V.m^{-1}$ at a frequency of 156 MHz. At this frequency, the CEC recommendation is 61.4 $V.m^{-1}$ and therefore the electric field found at this frequency is much greater than the first action level and is even greater than the hazardous action level (3 times the first action level) of 184.2 $V.m^{-1}$.

The higher frequency measurements of power density produced maximum levels of 20 $mW.cm^{-2}$ near a VHF aerial on Platform A. None of the other sources produced power density levels exceeding the CEC action level of 1 $mW.cm^{-2}$.

CONCLUSION

The majority of the levels measured were found to be below the recommendations by the CEC for all frequencies measured although some did exceed them. On three platforms, levels at ELF were found to exceed the recommendations with a maximum of 2.6 mT found on one platform which is approximately 8 times the CEC recommendation of 333 μT. Levels measured near the telecommunications equipment also exceeded recommendations on isolated incidences. Throughout the measurements, the time dependency of the exposure has not been taken into account and clearly this greatly reduces the overall exposure.

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METHODS OF DEVELOPING AN EMISSION CATASTER FOR LOW FREQUENCY ELECTRIC AND MAGNETIC FIELDS

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Introduction

In the common environment each individual is exposed to a large variety of power frequency or extremely low frequency (EFL) electric and magnetic fields. Traditionally these fields have been thought to be harmless but recent scientific studies reveal, however, that at least effects do exists. Therefore further investigations have to be carried out in order to obtain a greater data base. As a major point new ways have to be established that allow an effective collection of all relevant data concerning the exposition in this fields. In our contribution for the first time methods that consider simultaneously all major field emittants in the range of 0-500 Hz spread over the area as great as a town district. These methods have been developed and successfully tested in the district Buch/Karow of Berlin during the research study *"Emissionskataster der elektrischen und magnetischen Feldexposition im Stadtgebiet Berlin - Eine vergleichende Pilotstudie in ausgewählten öffentlichen Bereichen" (Emission cataster of electric and magnetic field exposure in Berlin - a comparative study in selected public areas)* [1][2][3]. In this study the following field sources have been taken to consideration:

- 110-kV, 220-kV, 380-kV transmission lines
- 110-kV power transmission cables
- relay station Karow
- 10-kV medium voltage level
- 1-kV distribution level
- power stations
- railway
- urban railway

General methods

In contrast to conventional environmental factors electric and magnetic fields are characterised by the following special properties:

- strong spatial fluctuation
- fast temporal variations

The latter are concerning mainly magnetic fields witch are able to penetrate almost every material whereas electric fields are strongly distorted near vegetation or buildings what in most cases is leading to near field reduction. In order to enable a collection of field strength data over a large area it is necessary to make further simplifying assumptions:

- mean or maximal (worst case) current values
- undisturbed electric fields

Based on these assumptions it is possible to extrapolate measured data within acceptable errors. Measurements therefore are only necessary to classify objects and to normalize calculated data. In the case of power transmission lines i. e. it is possible to bring calculated and measured data into congruency within 95% [4]. This can be done via measuring traverse profiles on well defined positions and knowledge of all important parameters, particular transmission line currents and exact distance between conductors and ground that depends on operational and environmental conditions.

A suitable way for determining the catenary wire current and the backwards directed current of railway lines or urban railway lines can be achieved by simultaneous measurement of magnetic field strength at different distances (i. e. 5, 10, 15 and 20 m) over longer time periods. Subsequential simulation calculations carried out with the current as input parameter have to be varied as long as their profile fits with the measured values. Also in this case an exact knowledge of the railway line configuration is required.

Computational methods and graphical processing

In the above mentioned project a newly developed software [5] was used that enables simulation of whole units of energy distribution facilities and allows calculation of large areas in a straightforward manner. Based on the construction documents the corresponding objects as super grids etc. have been transferred to the computer. Exact positioning could be achieved easily by taking the coordinates out of digitalized maps which have been obtained by vectoring analogue maps in the *"Landeskoordinatensystem Soldner Berlin"*. After computation calculated isolines of the field strengths together with object geometry's could be transferred back to the digital maps using the CAD program AutoCadLT™.

Example: Power Transmission Lines

The investigated area is divided by three transmission lines of 380-kV, 220-kV and 110-kV respectively. The latter is partially installed as an earth cable. The corresponding magnetic flux (Fig. 1) was calculated assuming medium currents and air temperature of 10°C (according to DIN VDE 0210).

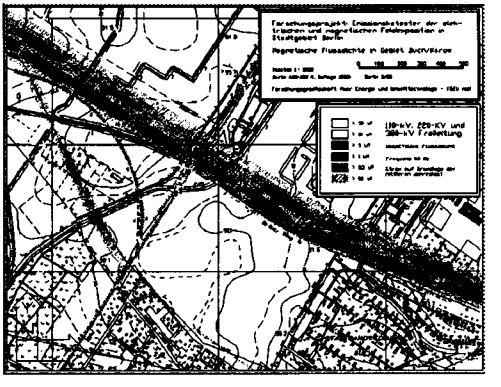

Fig. 1: Calculated magnetic field strength of power transmission lines in Berlin-Buch/Karow 1 m over ground (maximum field strength: 4.2 µT).

Calculations are based on the documents and data provided by the energy supply companies. Errors in height data, especially the minimal rope-ground distance, case distinct variations in calculated ground field strength as demonstrated in Fig. 2.

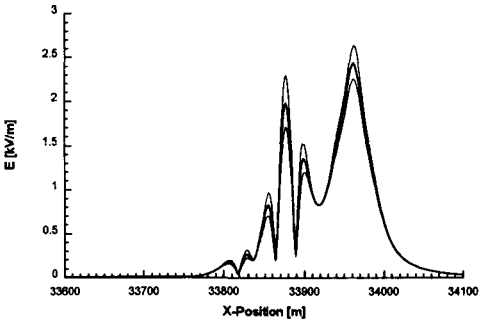

Fig. 2: Example of transverse profiles of the electric field strength under the 3 transmission lines in Berlin-Buch/Karow calculated for a variation of the minimal rope-ground distance of ± 1 m (x-position in the *"Landeskoordinatensystem Soldner Berlin"*).

An example for the good agreement that can be achieved between measurement and calculation is shown in Fig. 3, were the profile along the trace axis of the 380-kV transmission line is displayed. Even the drastic decrease of the electric field strength near the super grids is perfectly matched by the calculation curve.

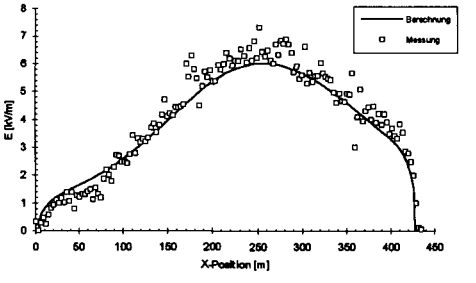

Fig. 3: Measured and calculated electric field strength on the centre line of the 380-kV transmission line.

Example: Railway

In the area under investigation is also running an electrificated railway line with two tracks. Computation was based on a model of three conductors (two rails and the cat wire), which general configuration is given in the literature. currents were assumed to be maximal (2x500 A, 16 2/3 Hz) and the part of the backward current in the rails was set to 95%. These values have to be seen as upper limits and thus the calculation could be seen as a worst case approximation. Although in Berlin-Buch/Karow railway as well as urban railway lines are running on an embankment of a height of 4 m, the reference height for the magnetic field strength was set to 1 m, since this height is relevant for persons in the environment of the railway line.

The results obtained with these approximations are shown in Fig. 4, but one has to keep in mind that the depicted field strengths represent maximum values that only be expected during short time intervals.

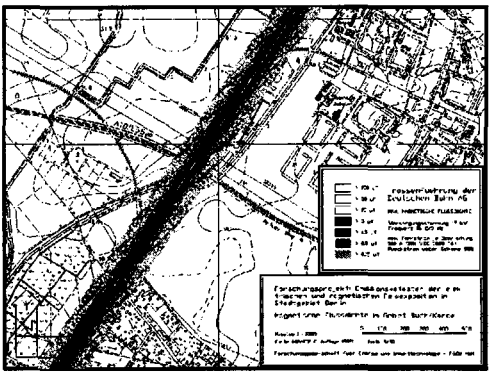

Fig. 4: Magnetic field strength near the railway embankment in Berlin-Buch/Karow calculated for maximal current (2x500A, 16 2/3 Hz), at 1 m over ground (maximum field strength: 20 µT).

In general mean values of the magnetic field strength are about one order of magnitude lower as can be seen in Fig. 5, which shows a measurement over a longer time period.

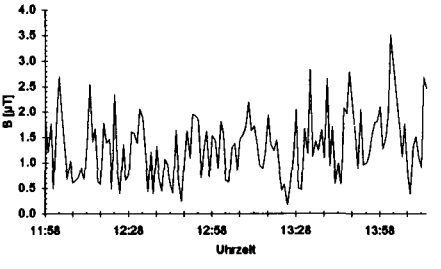

Fig. 5: Typical long-time measurement of magnetic field strength along a railway line in Berlin-Buch/Karow measured 1 m over ground at a distance of 12 m.

General results

Further major results of magnetic field strengths are shown in the following spreadsheet:

	B [µT]	current	f [Hz]
power transmission lines			
110-kV	2.0	med.	50
220-kV	4.2	med.	50
380-kV	2.6	med.	50
110-kV cable	0.24	med.	50
10/1-kV cable	0.67	med.	50
power stations (max.)	3.45	med.	50
power stations (value in 1.7 m)	0.3	med.	50
railway	20	max.	16 2/3
urban railway	29	max.	DC

One should keep in mind that some field strengths represent medium values while others base on maximum currents. A complete discussion of the obtained data is given in [6].

Conclusion

The main conclusions of the reported examples can be summarized as follows:

a) An isolated consideration of a single field source often not is possible since other sources, as other transmission or railway lines, do contribute to the total field strength in a non neglectable amount.

b) Excellent agreement between measurement and calculation is achievable under exact knowledge of all relevant parameters. This even holds for the more complex calculation of electric field strength, although in practice the electric field is often influenced by the environment.

c) Computation of electric and magnetic fields with acceptable errors and reasonable effort is possible also for large areas. High precision, however, requires exact determination of operational parameters and environmental conditions. Simple adoption out of data sheets always leads to great inaccuracies.

d) Magnetic fields of 10-kV and 1-kV cables in the investigation area could not be separated.

e) No relation was found between the number of underground 1-kV cables and the corresponding magnetic field strength in streets. An explanation is the compensation of single fields, as verified by computational simulation.

f) The current asymmetry on the 1-kV distribution level is in the range 1-3 A and contributes the major part to the magnetic field.

g) Magnetic field strength near the inspected power stations exceeded the street level only in the local environment (distance below 1 m).

h) In the case of railway lines the problem of strong time variations of cat wire currents is up to now only be accessible assuming maximum currents.

The research project *"Emissionskataster der elektrischen und magnetischen Feldexposition im Stadtgebiet Berlin - Eine vergleichende Pilotstudie in ausgewählten öffentlichen Bereichen"* therefore contributes to a great part to the clarification of the following points:

- realizability of low freq. electric and magnetic cataster
- creation of a real environmental data base
- estimation of the actual effort
- testing of computational methods
- reasonableness of extended field methods

Since the calculated field strengths in most cases remain below critical values [7][8] or don't case technical troubles even at this early point the conclusion can be drawn that a standardized extensive evaluation of low frequency fields is not top priority. Additional investigations of selected areas, however will yield further important insight.

The project was supported by:

- Berliner Kraft- und Licht (Bewag) - Aktiengesellschaft
- Senatsverwaltung für Gesundheit
- Senatsverwaltung für Stadtentwicklung und Umweltschutz
- VEAG - Vereinigte Energiewerke Aktiengesellschaft

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SENSORS FOR LOW FREQUENCY ELECTROMAGNETIC RADIATION

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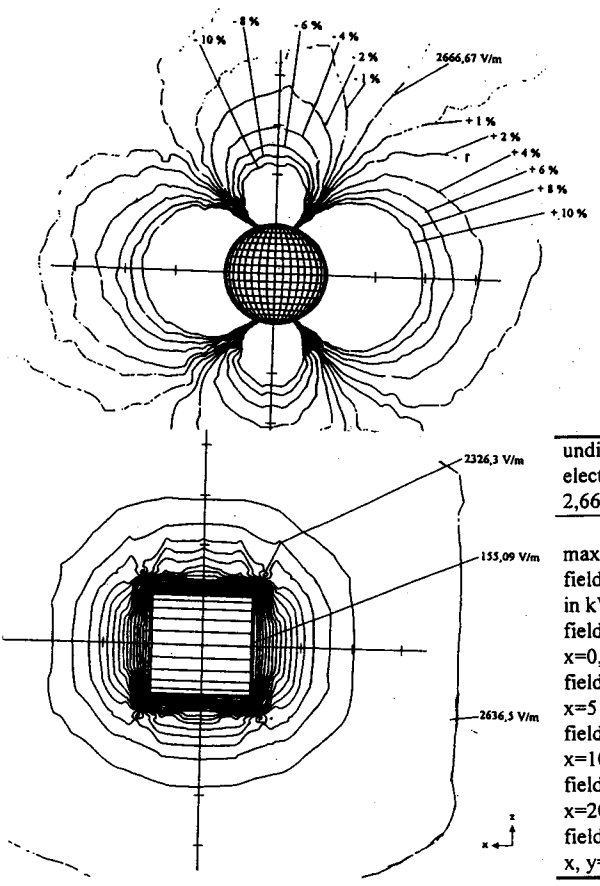
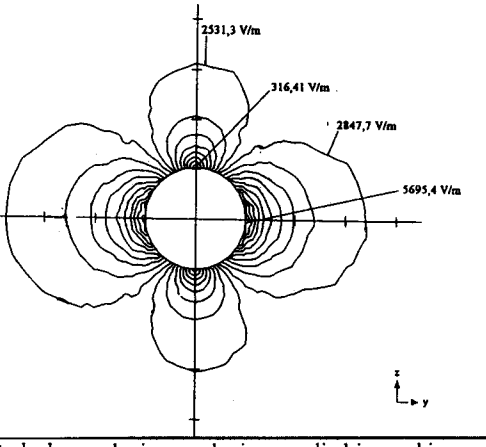
INTRODUCTION

For estimating the personal exposure from low frequency electric and magnetic fields suitable measurement systems are necessary. Investigations of appropriate magnetic and electric field probes and sensors have been done as a basis for a pocket sized personal dosimeter. There exist many different sensors, which can be used for the measurement of static and low frequency magnetic fields, such as inductive transducers, hall generators, tunnel diode oscillators, flux gate sensors and magnetoresistive sensors. In contrast to the magnetic sensors there are less different sensor types based on different physical detection principles available showing a sufficient sensitivity for low frequency electrical fields. Important is the knowledge on the feed back of the sensor itself on the detected field strength. Whether a sensor is suitable for a certain application is depending on its frequency related sensitivity, its signal to noise ratio, linearity and detection range, sensitivity to environmental influences as temperature, humidity etc. This presentation will focus on some important basic aspects of the measurement of low frequency electrical and magnetic fields for radiation protection purposes.

ELECTRICAL FIELD: DETECTOR GEOMETRY INFLUENCES ON THE UNDISTURBED FIELD

By means of analytical and numerical (1) investigations feed back effects on the undisturbed electrical field have been investigated. The results are presented in the following examples of three different sensor geometries, spheres, cylinders and cubes (Fig.1)



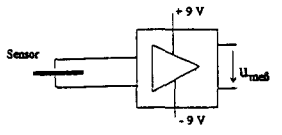


undisturbed electrical field 2,667 kV/m	spheric geometry r = 1 cm	spheric geometry r = 2 cm	cylindric geometry r = 1 cm	cubic geometry a/2 = 1 cm
	analytical	numerical	numerical	numerical
maximum field in kV/m	8,0	9,1	10,9	9,8
field in kV/m x=0, y, z	8,0	9,1	10,9	9,8
field in kV/m x=5 mm, y, z	6,9	5,9	6,0	5,4
field in kV/m x=10 mm, y, z	3,5	3,3	10,9	9,8
field in kV/m x=20 mm, y, z	2,8	2,9	3,0	3,0
field in kV/m x, y=0, z	2,7	3,1	3,1	3,0

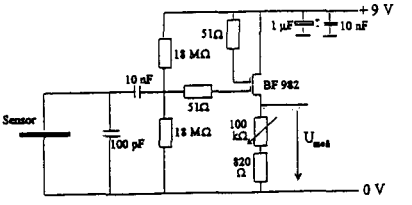
Fig. 1: Feed back of different sensor geometries on the undisturbed electrical field (the origin of the x,y,z - coordinate system is the geometric center of the object)

ELECTRICAL FIELD: EXAMPLE FOR THE MEASUREMENT, INFLUENCED CHARGE METHOD

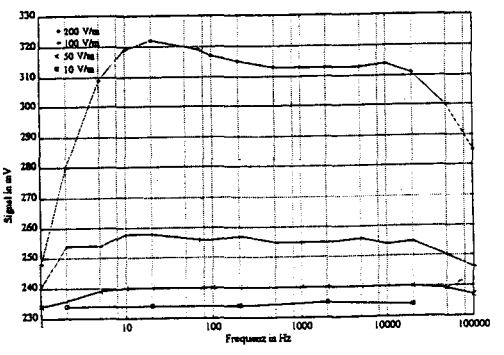
There are different detection principles for electrical fields such as the measurement of induced electrical charge, induced coulomb force on charges, deviation of electron beams, piezo-effect, electro-optical effects (Pockels-, Kerr-Effect), modification of the gate to source voltage of field effect transistors by induced charges etc. There is a big spectrum of different electronical implementations resulting in different sensitivities and frequency responses of a sensor. The following examples show two possibilities for the measurement of induced charges on a capacitive sensor, a differential amplifier circuit and a field effect transistor circuit (2, modified). They show differences in detection efficiency and frequency dependence (Fig. 2).



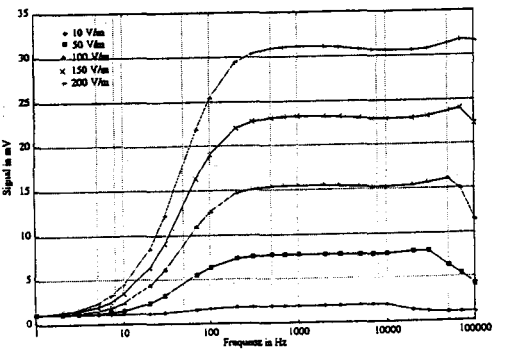
capacitive sensor 72 mm diameter, 125 pF



capacitive sensor 72 mm diameter, 125 pF



without electrical field: 1,1 mV



without electrical field: 233 mV

Fig. 2: Frequency dependent sensitivity of given sensors detecting induced charges on a capacitive sensor using
a) a differential amplifier circuit (left) and
b) a field effect transistor circuit (right)

Electrical field strengths of 0, 10, 50, 100 and 200 V/m and frequencies varying from 1 Hz to 100 kHz have been applied. There is a small lower frequency detection limit for the differential amplifier version, which can be varied by modification the amplifier entrance circuit. The entrance capacity reacts as a frequency dependent short circuit resulting in a frequency depending loss of sensitivity. The linearity of the sensor signal is poor. The absolute sensitivity of course depends on the effective area, where electrical charge is induced.

When field effect transistor circuits are used (Fig. 2 b), there is a higher detection limit at lower frequencies. The limiting factor for the lower frequency limit is the resistance R_E, which was chosen to be 18MΩ. Higher nominal values for R_E often have been found to show lower resistance's as indicated. In addition air humidity and resulting current losses become important and result in instable circuit conditions. To increase the sensitivity the parallel capacitance, which is 100 pF in Fig. 2b, has to be increased. As a consequence the lower frequency limit increases also. It is a question of optimisation on the application conditions, what is best.

MAGNETIC FIELD: DIFFERENT SENSOR PRINCIPLES AND THEIR APPLICATION RANGE

The following sensors for the measurement of static and low frequency magnetic fields have been investigated: inductive transducers, hall generators, tunnel diode oscillators, fluxgate sensors and magnetoresistive sensors. The results in terms of detection range for the magnetic field strength and the detectable frequency range are presented in Fig. 3.

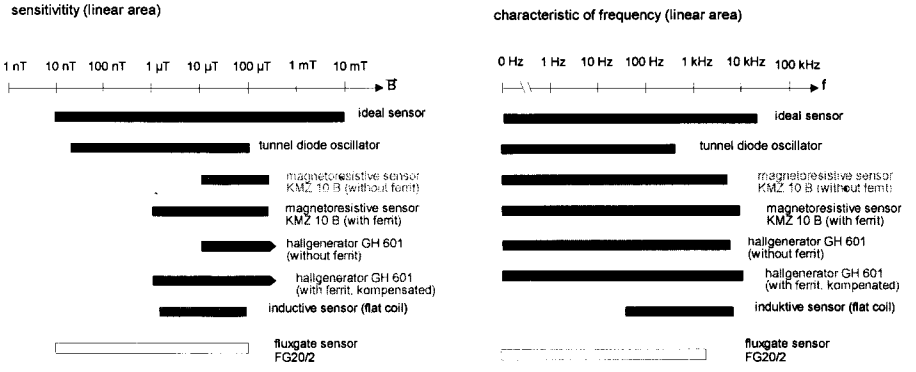


Fig. 3: Comparison of different sensor types, the linear range of their sensitivity and the detectable frequency range

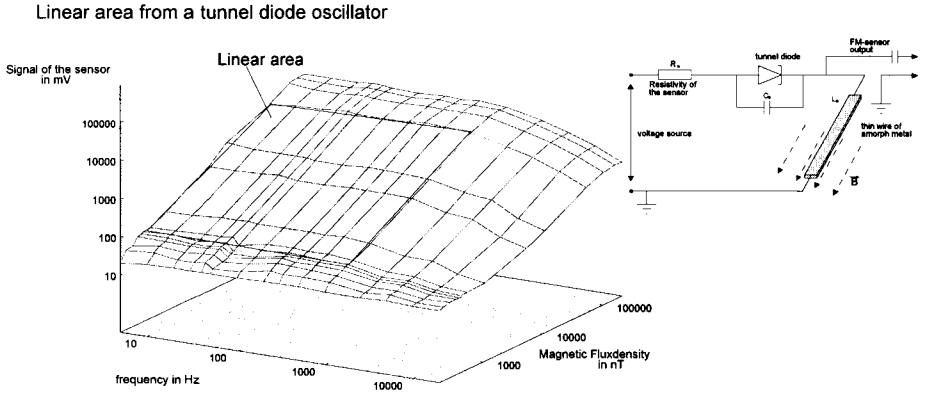


Fig. 4: Linear range from a tunnel diode oscillator

For radiation protection purposes an ideal sensor should cover the sensitivity rage from 10 nT up to at least 10 mT and the frequency rage from DC up to 30 kHz. Compared to that conditions, none of the tested sensors covers all. In consequence a combination of different sensor types will be necessary. Such a combination could be a fluxgate sensor combined with a coil. The mentioned coil has to cover the „upper" frequency range. An example for the linear range of the sensor signal of a tunnel diode oscillator over frequency and the magnetic field is presented in Fig. 4. Also for magnetic field sensors the choice, which is best, depends strongly on the application.

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NUMERICAL EVALUATION OF SAR IN PREGNANT WOMEN DURING MRI EXAMINATIONS

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INTRODUCTION

Magnetic Resonance Imaging (MRI) tomography, although among the safest diagnostic techniques presently exploited, presents some risks for patients. Among the major causes of risk is heating of tissues elicited, through the Joule effect, by the electric currents induced by the radiofrequency (RF) electromagnetic field used in MRI. International guidelines from radiation protection organizations such as the International Non-Ionizing Radiation Committee of IRPA (IRPA/INIRC) have been published in order to protect the patients from RF electromagnetic fields during MRI examinations (1): limits are given in terms of Specific Absorption Rate (SAR), i.e. the power absorbed per unit mass, which is regarded as the most suitable physical quantity to describe the thermal load induced in the body by RF fields.

Special caution in the use of this technique is recommended by IRPA/INIRC in the case of pregnant women: the MRI examinations should be postponed until after the first trimester, and limited to cases in which the diagnostic information cannot be obtained by ecography. This caution results from the scarce available information on the safety of MRI during pregnancy. An important step to obtain more information on this subject is the theoretical determination of SAR in the mother and in the fetus. To evaluate the SAR distribution in the different tissues of models that we have developed on the basis of information from ecographic scanning, we have employed a numerical technique, the impedance method, previously used to determine SAR in a model of the human torso subject to different procedures of MRI tomography (2,3).

Preliminary results on SAR distribution are presented and discussed.

THE IMPEDANCE METHOD

In the frequency region used in MRI diagnostic examinations, the absorption of RF energy in the body is mainly associated with the magnetic field, depending on several physical properties of the field, like frequency, strength, polarization (relative to the body), and on the geometrical and physical properties of the body. In the impedance method the properties of the body are summarized in a numerical model composed of a given number of cells, each one characterized by its mass and its electric impedance, calculated from the mass density and the electric permittivity and conductivity of the tissues composing the cell. The body is replaced by a three-dimensional network of impedances that can be subjected to a mesh analysis: calculating by the Faraday's law of induction the electromotive force due to the time-varying magnetic field in each loop, it is possible to evaluate the electric current circulating in each cell, and then the SAR relative to each single cell, and the SAR averaged on the various tissues.

This technique, which uses a quasi-static approximation, has some limitations in frequency: it was shown in fact, by means of comparisons with analytical calculations of SAR in cylindrical models, that the impedance method provides reliable estimates of SAR up to 30 MHz, while at higher frequencies the estimates are systematically greater than the analytically evaluated SARs (3).

THE NUMERICAL MODEL

On the base of transverse sections of the pregnant woman obtained from ecographic examinations, we have made some cell models of the mother-fetus system at different stages of pregnancy. The anatomical resolution of these sections is 5 mm, while, to increase the precision of results (as suggested by comparisons between analytical and numerical solutions obtained on a multi-layer cylinder), in the numerical models we have used cubic cells of 1 mm edge, obviously at expense of computation time and memory resources. Each transverse section is

contained in a 32 cm x 21 cm rectangle, corresponding to 67200 cells.

Models refer to the 12th week, the 24th-26th week, and the 38th-40th week. In Figure 1 a transverse section of the pregnant woman at the end of the first trimester is shown, corresponding to the horizontal plane 10 cm above the base of the uterus.

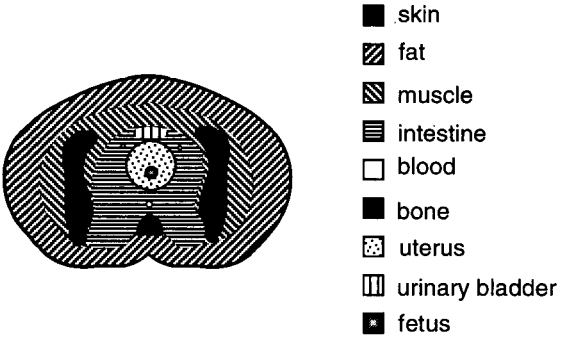

Figure 1. Transverse section at 10 cm above the base of the uterus at 12th week of pregnancy.

RESULTS AND DISCUSSION

Our preliminary results refer to the SAR deposited in the woman at the 12th week of pregnancy by an RF magnetic field directed along the axis of the body, at the frequencies of 13 and 30 MHz. In this paper, we only consider the 30 MHz frequency, which corresponds to the higher values of SAR, as it could be foreseen by simple theoretical considerations, synthesized as follows:

$$SAR \sim \sigma(f)\, E^2 \sim \sigma(f)\, (dB/dt)^2 \sim \sigma(f)\, f^2\, B^2$$

where $\sigma(f)$ is the electric conductivity, increasing with the frequency f, E is the induced electric field and B is the magnetic flux density.

For exposure to continuous waves, and in the case of magnetic flux density B = 1.26 μT at 30 MHz, the calculated distribution of SAR in different tissues and organs is that shown in Figure 2.

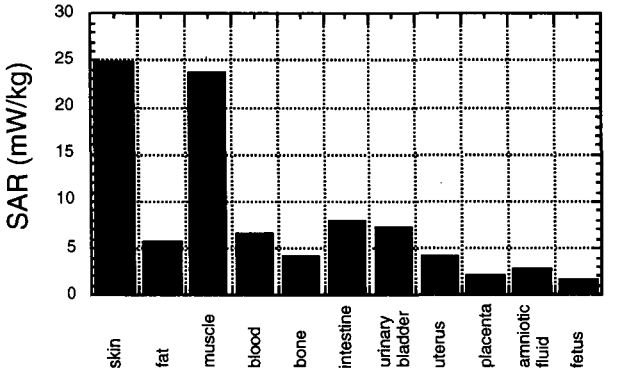

Figure 2. SAR distribution in a pregnant woman at 12th week. (B=1.26 μT, continuous wave, f = 30 MHz)

By means of suitable correction factors, we can obtain SAR values in more realistic situations. These correspond to various MRI procedures which use different values of the magnetic flux density and pulsed fields with duty-cycle $D = N_i N_s \tau / T$, where N_i is the number of pulses in a sequence, N_s is the number of excited slices during a sequence, t is the pulse width and T is the repetition time of sequences: in Table I data are shown relative to SAR deposited in the fetus and in the amniotic fluid for some typical MRI examinations.

Table I. SAR in the fetus and in the amniotic fluid for some typical pulse sequences.

MRI procedure (f = 30 MHz)	**Correction factor**	**SAR (mW/kg) in the fetus**	**SAR (mW/kg) in the amniotic fluid**
Continuous wave: B = 1.26 μT	1	1.6	2.7
Spin Echo: 1 echo, 1 π pulse τ = 1 ms, T = 1 s, B = 23.6 μT	0.35	0.57	0.94
Fast Imaging: 1 echo, 1 π /2 pulse τ = 5 ms, T = 0.1 s, B = 2.36 μT	0.175	0.29	0.47
Inversion Recovery: 4 echoes 5 π pulses, 12 slices τ = 0.5 ms, T = 1 s, B = 47.1 μT	41.9	68.5	112.4

Owing to the specificity of the "target" organ (the fetus in the amniotic fluid), the results are not easily comparable with the recommended SAR limits (1) which refer to individuals with thermoregulatory system. However, it is possible to evaluate the temperature rise in the amniotic fluid and in the fetus supposing the absence of a mechanism to get rid of the heating produced by electromagnetic energy absorption. Considering the specific heat of amniotic fluid and fetus equal to that of water, and imposing an upper limit of 0.5 °C to the temperature rise, we obtain a limit of 2093 J/kg for the product SAR x Δt, where Δt is the exposure duration. In the case of an MRI examination lasting 1 h, the limit for SAR is 0.6 W/kg, much higher than the values shown in Table I.

CONCLUSIONS

Preliminary results from the application of the impedance method to a realistic numerical model of the pregnant woman seem to indicate that SAR deposited in the fetus and in the amniotic fluid, at the end of the first trimester of pregnancy, is not such to cause too high rises in temperature.

Work is in progress relative to different polarizations of the RF magnetic field and to the other stages of pregnancy.

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MANUSCRIPT CUT IN LENGTH !

FOR FURTHER INFORMATION PLEASE CONTACT AUTHOR

Occupational exposure to radiofrequency radiation from (4 - 7 MHz) RF dielectric heat sealers.

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Introduction

We have conducted radiation measurements on 12 welding machines which were operated in a factory in the Netherlands. These machines were welded through electromagnetic fields, used to weld plastic parts through rapid heating. High temperatures are reached through application of frequencies between 5 and 7 MHz. The present investigation includes 3 different machine types: EA20 (5 machines), EA50 (4 machines) and HD500 (3 machines).

The production process

RF heaters are usually operated by women in their fertile ages. They perform their work standing upright in front of the RF sealers. Three machines however were operated by two women, who alternatively performed an operation on one of the machines. This regards the machines numbers 6, 7 and 8.

The purpose of the measurements was to determine the radiation level. It was to be investigated whether the RF radiation doses received from the RF sealing apparatus was acceptable according to the international regulations provided by the International Radiation Protection Association/International Non-Ionizing Radiation Committee (abbreviated as IRPA/INIRC). These regulations are strongly recommended by the Netherlands Institute for Radiation Technology (NIFRT).

A good assessment of the exposure levels is required because the operators of the RF welding apparatus are in close proximity, i.e. within two wavelengths from the radiation source. Both the electric as well as the magnetic field should be measured. For each, the strength is measured on seven locations on the body. During the measurements, the operator was in his normal working position. In cases where the readings went off the scale, a second (control) measurement was carried out at a larger distance. However, these did not affect the outcome of the investigation.

Description of equipment

Field strength measurements were done using a "Holaday Broadband Isotropic Field Strength Meter" type HI-3012. This is an analog instrument which displays Field Strength Units squared ($|FSU|^2$). Both electric as well as magnetic fields can be measured. The instrument has been manufactured in the United States in 1991. It come with two probes:
an MSE E-field probe with a range between 10^2 and 10^9 V^2/m^2;
and the HCH H-field probe with a range from 0.05 to 10.0 A^2/m^2.
The working range of the HI-3012 is between 0.5 MHz and 5 GHz (for electric fields) and between 0.5 and MHz and 300 MHz (for magnetic fields).

For both probes the accuracy is ±0.5 dB.
The apparatus has to be calibrated before use. The calibration is done by a specialised institution which provides a calibration factor (CF). This factor can be used to correct possible deviations. This factor (a number) is determined separately for each probe and is reported in a calibration report. Furthermore measurements were corrected for the time (during an operation) that the machine was not active. For this purpose a so-called Duty Cycle (DC) was determined. This is an effective average.

Results

The results have been combined in the following two tables. The strengths of the electric and magnetic fields provide an indication of the radiation levels to which the operator is exposed during the regular production process. Measurements have been corrected with the Duty Factor and the Calibration Factor. Comparison of the relative strengths of radiation exposure on different areas of the body shows that the eyes and the abdomen are more exposed to both E and H-field than knees and feet. In other words, those body parts that are at the same height or slightly above the welding spool are more exposed.

The table shows that excessively high electric fields are present near machines 2, 4, 6, 9, 11 and 12. The remaining 6 machines do not appear to deviate substantially from the accepted exposure-value as regards the E-field. The norm is derived from the peak-frequency. This results in a more rigorous value. Using an average frequency would result in a less stringent norm. However, this makes hardly any difference as regards the number of machines where the norm is exceeded.

As regards the different types of machines this means the following:

for the HD-500 and the EA-50 the norm is established using a frequency of 7 MHz. This results in a norm of 87.71 V/m (electric) and 0.23 A/m (magnetic).

for the EA-20 the norm is established using a frequency of 5 MHz. This results in a norm of 122.80 V/m (electric) and 0.32 A/m (magnetic).

Table 1 Electric Field Strength [V/m]

#	Eyes	Neck	Chest	Abdomen	Pelvic	Knees	Feet	Norm
1	64.42	91.10	128.83	78.89	40.74	14.40	12.88	122.80
2	225.89	338.84	>357.17	>357.17	>357.17	87.49	56.47	122.80
3	4.40	6.22	11.35	8.03	5.08	0.00	0.00	122.80
4	>53.52	107.04	>169.25	>169.25	65.55	16.93	15.14	122.80
5	8.28	5.23	6.92	5.85	4.14	3.70	2.48	122.80
6	>107.60	131.78	263.56	284.67	152.16	48.12	48.12	87.71
7	2.07	1.46	6.54	4.63	4.14	1.46	0.65	87.71
8	25.44	25.44	31.15	17.99	17.99	1.80	0.00	122.80
9	89.89	>201.00	0.64	>201.00	142.13	34.81	49.23	87.71
10	60.56	54.16	66.34	60.56	46.91	13.54	23.45	87.71
11	>156.08	78.54	99.35	86.04	60.84	19.24	14.90	87.71
12	112.84	138.20	9.03	338.52	178.42	30.90	0.00	87.71

Magnetic field measurements are less favourable than those for the electric field.
This follows from the data in the table.
In fact, only machines 5 and 7 are within the established norm.

Table 2 Magnetic Field Strength [A/m]

#	Eyes	Neck	Chest	Abdomen	Pelvic	Knees	Feet	Norm
1	1.52	1.07	1.32	1.07	0.29	0.17	0.15	0.32
2	1.33	1.33	1.33	1.33	1.33	0.52	0.21	0.32
3	0.42	0.30	0.37	0.42	0.60	0.04	0.01	0.32
4	>0.63	>0.63	>0.63	>0.63	>0.63	0.35	0.12	0.32
5	0.14	0.23	0.31	0.22	0.22	0.04	0.02	0.32
6	>1.27	>1.27	>1.27	>1.27	>1.27	1.06	0.57	0.23
7	0.04	0.05	0.15	0.17	0.09	0.02	0.01	0.23
8	0.56	0.21	0.30	0.30	0.30	0.21	0.03	0.32
9	>0.75	>0.75	>0.75	>0.75	>0.75	0.17	0.17	0.23
10	0.55	0.32	0.64	>0.71	>0.71	0.23	0.07	0.23
11	0.56	0.49	>0.59	>0.59	>0.59	0.26	0.02	0.23
12	1.33	1.11	>1.33	>1.33	>1.33	0.73	0.16	0.23

As regards exposure numbers for operators the following remarks are in order:

It can be seen that there is no relation between E-field radiation and H-field radiation
The radiation emitted varies with size, shape and thickness of the material to be processed
There is no proportional increase for exposure levels for either E-fields or H-fields with machine power (in kilowatts)

Conclusion

The present investigation shows clearly that RF-welders 2, 4, 6, 9, 11 and 12 very significantly exceed the standard norm. For some machines the readings were off scale, therefore an extra measurement was done at a larger distance.

It follows that for the machines mentioned certain provisions are necessary. People who are carrying certain medical aids such as pacemakers and other electronically regulated life-supporting equipment should not be allowed to enter the area where the RF welders are located if the welders are in operation. Pregnant women should not participate in this stage of the production process.

Recommendations

Machines where the norm is exceeded should be insulated in order to minimize the radiation exposure of the operator. Metal foil can already reduce the RF radiation.
The area where the machines are situated should be marked with special warning signs.
It is recommended that a medical file is created for each of the operators. This file should be updated twice yearly.

RADAR OCCUPATIONAL EXPOSURE: INTERFERENCES WITH THE FUNCTION OF THE NERVOUS SYSTEM

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INTRODUCTION

In the modern life and work environment the electromagnetic pollution is obviously increased and therefore the knowledge of its level and effects constitutes a major priority and challenge for the scientific community (4,11,12). In this international context the estimation of the health status of the microwaves exposed people colligated with the evaluation of the exposure have an important role not only for the scientific knowledge but especially for the improvement of the protection standards (7,13). Therefore the case of pulsed microwaves which seems to have peculiar biological effects (1,3), especially on the nervous system (10), has determined us to try to study in a clinical and epidemiological approach the occurrence of such effects upon chronic exposed humans.

OBJECTIVE

To assess the possible impairment of the nervous system function due to the chronic occupational exposure to low and medium levels of pulsed microwaves.

STUDY DESIGN

Cross-sectional study in 60 occupationally exposed people in radar maintenance activity, compared to matched controls, followed by a long-term survey of the main findings.

METHODS

Assessment of microwaves exposure (frequencies, power densities, SAR estimations, ergonomic investigations, questionnaires).

Assessment of the nervous system function few days after the cessation of the work: questionnaires, careful psychiatric and neurological examinations, psychological tests and electrophysiological investigations (EEG, short-latency evoked potentials, electroneuromyography - EMG, excitability, H-index as a measure of the sensitive conduction velocity, motor conduction velocity). Medical investigations in order to eliminate other causes of central or peripheral neuropathia have also been done. Yearly assessments of microwaves and of the nervous system (NS) function have subsequently been performed.

The exposed lot has included 60 electronic and electrotechnical technicians and engineers with the mean age of 39.7±3.6 years and the length of service in radar maintenance of 15.9±5.3 years. The control lot has comprised 36 people with the same electronic and electrotechnical training and activity, from the same factory, but which were never been exposed to microwaves. The age, the length of service and the other characteristics of the controls are matched with those of the exposed.

RESULTS

The investigation of the occupational exposure to pulsed microwaves has shown frequencies in the range 0.2-10 Ghz, average power densities of 0.01 - 5 - 10 mW/cm^2 and great variations of partial and total exposures to complex field distribution, with extremely various power densities and frequencies, generated in a close and small workspace by multiple unshielded generators. By using the methods and diagrams from the "Radiofrequency Radiation Dosimetry Handbook (Fourth Edition)", CH Durney, 1986, (5,11), we have estimated the Specific Absorption Rates (SARs). Various whole body SARs have been estimated: 0.01 - 2 W/kg. The local SARs, even more difficult to estimate, were greater for the head (neck) and hands (wrists).

The NS investigations has shown neurasthenia at 80% of the exposed versus 8.3% at the controls, the difference being statistically significant ($p<0.0005$); the relative and the attributable risk calculations have shown that pulsed microwave exposure is a risk factor and even a determinant factor for the neurasthenia. Significant changes of psychological tests (increase of the reaction time, decrease of the intellectual efficiency, impairments of the recent memory, attention, mood and behaviour) that support the higher and significant prevalence of neurasthenia have also been found. The mean age of the neurasthenic subjects is equal with the age of the whole lot, but the length of service of these subjects is statistically different (16.3±5.2 years vs. 11.6±5.7 years, $p<0.001$). This also means that the exposure in the above mentioned conditions could be a determinant factor for the neurasthenia. The prevalence of this finding is the same at the engineers and at the technicians, although the first group has a much more intense intellectual work. This also could be an evidence for the microwaves role in neurosis. The frequency and the severity of the neurotic symptoms are more intense when the length of service and/or the exposure are greater. Amplitude and frequency changes of EEG has revealed hypovoltages at 67% of the neurasthenic subjects and also decrease of the alpha waves index and increase of the theta and delta waves percentage. The auditory evoked-potentials have especially shown an increase of the latency of the second wave, which may reflect a possible long term stimulation of the Varolio pons as a structure of the acoustic way.

At the peripheral nervous system level we have found decreases of the peripheral nerves conduction velocity, especially sensitive, in 46.7% of the exposed people vs. 16% at the exposed, the difference being statistically significant ($p<0.001$). Slight neurological signs of polineuropathia have also been found. The relative and the attributable risk calculations have shown that pulsed microwave exposure is also a risk factor and even a determinant factor for the impairment of the conductibility on the studied nerves. The analysis of these changes has shown the followings: 30% of the exposed have a decrease of the sensitive conduction velocity (20% having only an aspect of incipient sensitive polineuropathia and the others - 10% - having also a decrease of motor conduction velocity, so they display an aspect of incipient sensitive-motor polineuropathia). 13.33% presents nervous conduction speeds at the lower limit of the normal and 3.33% have typical changes of motor polineuropathia. We have calculated the correlation index between the length of service and the H-index (sensitive conduction velocity) in all the exposed lot and we have found an inverse correlation: $r = -0.68$, which shows that the sensitive conduction velocity significantly decreases when the length of service increases. The calculation of the regression has found that an increase with

0.88 years of the length of service decrease the H-index with 0.52, relation considered as a signal and a stimulus towards further studies.

It is to be mentioned that the yearly revaluation of these findings showed similar aspects, but the people which worked less showed a slight improvement of some parameters.

CONCLUSIONS

The long time exposure at pulsed microwaves with average power densities rather small in comparison with the IRPA guidelines (7,11,13) on limits of exposure to RF seems to determine various degrees of neurasthenia, behavioural changes and peripheral polineuropathia aspects.

These changes with statistical significance, correlated with the length of service and exposure levels, the persistence at the exposed people of these impairments as well as, the absence of other neurotrop agents, have determined us to state the possible persistent interference of the pulsed microwaves with the electroneurophysiological phenomena.

It is to be mentioned that the findings concerning the long term pulsed microwaves effects on the peripheral nervous system in humans as well as the chronic changes of the auditory evoked potentials may contribute, if confirmed, to a better understanding of the interaction mechanisms of the electromagnetic fields and radiations with the nervous system and, of the significance for the human health and well-being of this interaction (2,6,8,9,14). So, we consider that this kind of studies must be continued as a complement to the experimental ones and that they could be really useful for the scientific knowledge.

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EFFECTS OF 415 MHz FREQUENCY ON HUMAN LYMPHOCYTE GENOME

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INTRODUCTION

The continuously increasing use of artificial sources of electromagnetic radiation in industry and medicine has been accompanied in everyday life with telecommunication systems which is followed with great interest in possible hazardous effects of this type of radiation. The interesting applications of mobile telecommunications and the use of cellular phones are of topic interest.

Numerous cytogenetic investigations are focused on the effects of microwave radiation from mobile communications frequency of 450 and 950 MHz on isolated cells in vitro (1).

The aim of this work was to investigate the effects of microwaves from mobile telephone freqeuncies on human peripheral blood lymphocytes cultured in vitro.

MATERIAL AND METHODS

To investigate the effects of microwaves from mobile telephone frequencies on human cells cultured in vitro, blood samples were taken from two healthy female donors, between 25 and 35 years of age (mean age 30 years). They were not smokers, and in the preceding year, they were not occupationally or for the purpose of diagnostics or therapy exposed to radiation. Whole blood samples were irradiated in semipermeabile membrane 24/32 cm (the Scientific Instrument Centre, Ltd).

The source of microwaves was a mobile telephone apparatus, standard NMT-450*. The samples of blood were irradiated in the real, everyday environmental conditions of a standard telephone user, at room temperature of 24 °C.

Four ml of heparinized blood was injected in 10 cm long semipermeabile membrane and each such prepared sample was located on unconductive surface 10 cm far away from the middle part of the telephone antenna. Orientation of samples was parallel to the antenna (in the direction of electrical field vector). The samples were exposed to microwave frequency of 415 MHz , at time intervals of 10, 20 and 30 minutes, output power 15 W.

Calibration and frequency measurement were performed on STABILOCK 4015 (Schlumberger) and during the experiment output and reflected power were continuously measured using power metre PM 400-1000 (PROCOM).

Following exposure, 72-h cultures were set up and slides were prepared according to standard methods for analysis of micronuclei (2). The method involves cultivation of 0.3 ml of whole blood, 4 ml of F-10 medium (Gibcco) supplemented with bovin serum (20%) and phytohaemagglutinnine (Murex). After 44 hours of cell cultivation, cytochalasin B (Sigma) at a final concentration of 3 μg/ml was added. The method is adapted from Fenech and Morley (2).

Five hundred binuclear lymphocytes were scored per sample. Results were expressed as total number of micronuclei per cell and their distribution.

**NMT (Nordic Mobile Telephone) is a standard of analogous (frequency modulation) mobile telephone network.*

RESULTS AND DISCUSSION

In order to show possible cytogenetic damages caused by exposure to 415 MHz waves, the micronucleus test was used as a very sensitive method which give possibility to detect micronuclei which contain lost chromatid/chromosome fragments in the form of "micronuclei".

The data obtained on the frequency of micronuclei in human lymphocytes after in vitro exposure to microwaves of 415 MHz with varied exposure intervals, is shown in Table 1.

Table 2. gives measured and calculated values of relevant parameters during the experiment.

The output power value of the antenna was a constant, and reflected power value was 0 W.

Exposure (min)	*No. of CB cells*	*No.of micronuclei*	*Micronuclei distribution/cells*					*Cells with MN*	*Micronuclei/cells (±SD)*
			0	*1*	*2*	*3*	*4*		
10	500	22	481	16	3			19	0.044±0.23
20	500	29	479	15	4	3		21	0.062±0.32
30	500	31	476	20	2	1	1	24	0.064±0.33
control	500	9	491	8	1			9	0.020±0.15

Table 1. The frequency and distribution of micronuclei in human lymphocytes exposed to 412,.950 MHz

Output power (W)	*Frequency of continuous electromagnetic wave (MHz)*	*Calculated mean value of power density in the sample** (mW/cm^2)*	*Exposure time (min)*
15	412.950	19.709	10, 20, 30

Table 2. Experimental conditions

The number of micronuclei in non-irradiated control samples was 6 per 500 analysed cells. On the contrary, in the cells exposed for 10 min, the number of micronuceli was 22 per 500 binuclear cells. After 20 min the number of micronuclei increased to 29, and after 30 min to 31 micronuclei per 500 cells. Beside the increased number of cells with micronucleus, various distribution of micronuclei per cell was also noted. The cells with 1, 2, 3 and 4 micronuclei observed in the exposed samples, indicated the effect of 415 MHz waves on mitotic spindle or formation of DNA breaks. The data obtained pointed also to an increase in the number of micronuclei in relation to the length of exposure, whereas other authors reported that effect of 954 MHz waves disappeared with increasing distance from the source (1).

One of the most important priorities in research of electromagnetic field effects is investigation of its synergistic effects with chemicals (3).Significantly higher mitomycin C induced SCE frequencies were observed in lymphocytes previously exposed to 954 MHz waves (4).

***Calculation was based on ideal circumstances, with ceertain approximations. The samples has assumed the free space impendance*

DNA damage caused by electromagnetic fields has been recently determined by identification of Single Gel Electrophoresis (SCGE) or comet tail lengths using comet assay described by Singh et al. (5). According to results reported by Verschaeve et al. in human peripheral blood lymphocytes exposed in vitro to frequency of 954 MHz, increase of single strand DNA breaks were observed compared to control value (1).

Since studies of other authors as well as results obtained in our investigation indicate the presence of damage of human lymphocytes genome caused by exposure to microwaves from mobile communication, further cytogenetic investigations are necessary in order to estimate the effects of exposure and to determine permissible levels of exposure.

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DETERMINATION OF THRESHOLD VALUES OF ELECTROMAGNETIC NEAR-FIELDS FOR PATIENTS WITH IMPLANTED PACEMAKERS IN THE FREQUENCY RANGE 30KHZ - 50MHZ

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Abstract

In order to protect persons with implanted pacemakers, threshold values of electromagnetic fields should be prescribed. In the German standard DIN VDE 0848 threshold values in the frequency range 30 kHz - 50 MHz are only given for far-fields. However, near-fields have usually higher densities and their behavior is much more complicated than that of far-fields. Therefore, it is significant to determine threshold values of near-fields. In this work, we follow the following procedure: 1. The near-field is characterized by using a Taylor series with a few terms. 2. The induced field in the human body is calculated for each single Taylor term. 3. The coupling mechanism of each single Taylor term is investigated for a number of different pacemaker models at different locations. 4. Threshold values of the near-field are determined regarding the maximum allowed disturbing voltage at the entrance of the implanted pacemaker. This procedure is also applied to far-field problems and exhibits a very good agreement with the known threshold values.

1. Introduction

In the field of electromagnetic compatibility and protection of persons, it is desirable that the danger probability for people in a certain area can be obtained directly from local measurements of electromagnetic fields [1]. It is possible to define threshold values of electromagnetic fields for persons with pacemakers if the relation between the field at absence of the human body and the disturbing voltage induced by this field at the entrance of the pacemaker is known. The maximum allowed disturbing voltage is prescribed in the German Standard DIN VDE 0848. In order to achieve this goal, one has to characterize electromagnetic fields with the help of a few parameters. Attempts have been made with conventional methods e.g. the multipole method and the plane wave expansion. But those methods do not lead to a substantial reduction of the number of parameters with which near-fields can be described.

In Section 2, a Taylor series is introduced to characterize electromagnetic fields in a region, whose size is much smaller than the wavelength and the distance between this region and the sources. It is also shown, how the Taylor coefficients can be determined from the measured field densities in this region considering the constraints due to Maxwell's equations. The total field is then a sum of the products of the dimensionless Taylor coefficients and the fields of single Taylor terms. In Section 3, the induced field in the human body is calculated for each single term. Then, the disturbing voltage at the entrance of the pacemaker is investigated in Section 4 for a number of different pacemaker models. Considering that the total disturbing voltage should not exceed the maximum allowed disturbing voltage, threshold values can be given in concrete cases.

2. A General Taylor Series and Constraints

A general expression of the electric and magnetic field strength in a source-free area can be written in the following way:

$$\vec{E} = \sum_{l+m+n=0}^{\infty} \vec{E}_{lmn} x^l y^m z^n , \quad \vec{H} = \sum_{l+m+n=0}^{\infty} \vec{H}_{lmn} x^l y^m z^n . \tag{1}$$

We concentrate our attention to the magnetic field density because of the similarity of the two fields. By substituting the equation into the wave equation and the equation $\nabla \cdot \vec{H} = 0$, one obtains

$$(l+2)(l+1)\vec{H}_{l+2mn} + (m+2)(m+1)\vec{H}_{lm+2n} + (n+2)(n+1)\vec{H}_{lmn+2} + k^2\vec{H}_{lmn} = 0 , \tag{2}$$

$$\vec{e}_x(l+1)\vec{H}_{l+1mn} + \vec{e}_y(m+1)\vec{H}_{lm+1n} + \vec{e}_z(n+1)\vec{H}_{lmn+1} = 0 , \tag{3}$$

where k is the wave number. In a small area, one needs only a few Taylor terms to describe the magnetic field:

$$\vec{H}_t(x,y,z) = \vec{H}_1 + x\vec{H}_2 + y\vec{H}_3 + z\vec{H}_4 + x^2\vec{H}_5 + y^2\vec{H}_6 + z^2\vec{H}_7 + xy\vec{H}_8 + xz\vec{H}_9 + yz\vec{H}_{10} . \tag{4}$$

Equations (2) und (3) can be represented by 7 constraints,with which the coefficients $\vec{H}_1 \ldots \vec{H}_{10}$ are determined from the measured field data. In order to compensate for the errors in the measurements and to maintain the physical properties of the field, one must consider the 7 conditions determining the coefficients. For this purpose, the least squares method of Gauss and the method of Lagrange's multipliers are employed [2]. Denoting the measured fields and the fields described by the Taylor series with the index m and t, respectively, one has only to find the coefficients which minimize the expression

$$S = \sum \left[(H_{xm} - H_{xt})^2 + (H_{ym} - H_{yt})^2 + (H_{zm} - H_{zt})^2 \right] \tag{5}$$

under the constraints, where the sum is over all points at which the field is measured. It follows a linear algebraic equation system from which the coefficients can be determined with a sufficient number of measured data.

As an example we show a log-periodic antenna with 10 elements located in the xz-plane (Fig 1). Between two adjacent elements there are relations $h_{n+1}/h_n = d_{n+1}/d_n = 0.83$. The first element has a length of 5 m and a radius of 5 mm. The elements are crossways connected by a double transmitting line of a wave impedance 50 Ω and fed at the middle of the shortest element with $1V$ at 30 MHz. The near field is computed by using the method of moments. In a cube of $(2.4\ m)^3$ with its center at the point (0.2 m, 6.2 m, 0.2 m), the Taylor coefficients are calculated by the method described above. The field distribution reconstructed with the Taylor coefficients by using Equation (4) only differs a little from that calculated using the method of moments. The error is less than 1%.

3. Calculation of the electromagnetic field in the human body

The electromagnetic field in the human body induced by the single terms should be calculated in this section. As the single terms do not satisfy all Maxwell equations, we replace the single terms by single Huygens' sources [3], whose fields satisfy Maxwell's equations. With the help of the Huygens' sources, we can construct fields in the volume V_A which are very similar to the original Taylor terms. For example, if one places an electric current sheet $\vec{J}_A^{2y} = \vec{n} \times \vec{e}_y\, x\, [A/m^2]$ immediately inside a cuboid, whose outer space is filled with ideal magnetic material, one obtains a magnetic field in the cuboid which differs hardly from $\vec{e}_y\, x\, [A/m^2]$.

The numerical calculation of the field in the human body is performed by using the Finite Integration Technique [4]. Fig. 2 shows the body model and the pattern of the magnetic field in the volume with the human body at 30 MHz in the plane $y = 0$ m for the electric current sheet $\vec{J}_A^{2y}$. One can see that the magnetic field in the volume is almost linear in x like that in the volume without the human body. Similarly, the electromagnetic fields in the human body for other terms are calculated [5].

4. Coupling in the pacemaker and threshold values of electromagnetic near-fields

With the knowledge of the field in the human body, the disturbing voltage for all single terms can be calculated by using the method of moments. The case and the electrode of the pacemaker are modeled by a grid, in which the influence of the entrance impedance and isolation of the electrode can be investigated. For a set of Taylor coefficients, it is then easy to obtain the total disturbing voltages by summing the voltage for all single terms. In order to determine threshold values of electromagnetic fields, it is necessary to consider different positions of the pacemaker and electrode because of different implanting techniques employed. Therefore, the disturbing voltage was calculated for 36 different positions and configurations of the pacemaker and the electrode. The maximum value of the total disturbing voltage for the magnetic field $U_{St\max}^H$ and the electric field $U_{St\max}^E$ can be written as

$$U_{St\max}^H = \sum_{n=1}^{10} \sum_{\alpha}^{x,y,z} \left[\left| \frac{H_{n\alpha}}{[H_{n\alpha}]} \right| \left| U_{n\alpha}^H(k_{\max}) \right| \right],\ U_{St\max}^E = \sum_{n=1}^{10} \sum_{\alpha}^{x,y,z} \left[\left| \frac{E_{n\alpha}}{[E_{n\alpha}]} \right| \left| U_{n\alpha}^E(k_{\max}) \right| \right], \tag{6}$$

where $H_{n\alpha}/[H_{n\alpha}]$ and $E_{n\alpha}/[E_{n\alpha}]$ are the dimensionless coefficients and $U_{n\alpha}^H(k_{\max})$ and $U_{n\alpha}^E(k_{\max})$ the disturbing voltages for the single terms in the worst case. We obtain an estimate of the disturbing voltage independent of the position of the pacemaker system. The safety of persons with pacemakers in antenna fields is guaranteed if

$$U_{St\max}^H \le U_{SS}/2,\ U_{St\max}^E \le U_{SS}/2, \tag{7}$$

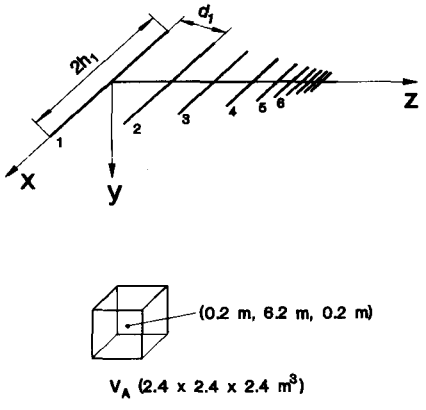

Fig. 1: Geometry of the logarithmic-periodic antenna

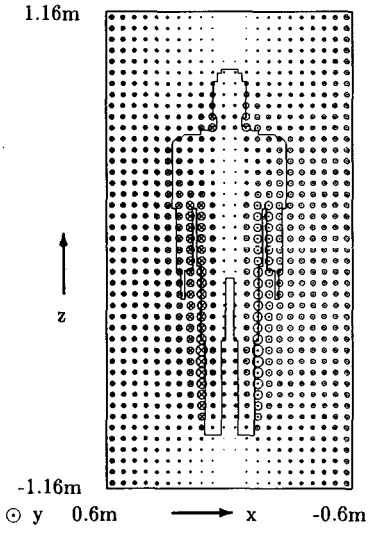

Fig. 2: Real part of the magnetic field $\vec{H}_{J_m}^{(x,y)}$ in the volume with the human body, generated by Huygens' current sheet $\vec{J}_A^{2y}$ (max. 1,46 A/m)

where $U_{SS}/2$ is the maximum magnitude of the disturbing voltage defined in Section 4.2.2.1.1 of Part 2 of the German standard DIN/VDE 0848.

At 30 MHz the maximum top magnitude is 0.145 V. With the coefficients for the field in the volume V_A near the logarithmic-periodic antenna calculated in Section 2, we conclude that the feeding voltage of the logarithmic-periodic antenna should not exceed 244 V.

The concept was also applied to far-field problems. The comparison of our results with those in DIN VDE 0848 shows a very good agreement in the frequency range up to 10 MHz (see the following Table). At 3 MHz, for instance, we have a threshold value of 14.6 V/m, whereas that in DIN VDE 0848 is 16.1 V/m.

Threshold values	$\hat{E}$/0.3MHz	$\hat{E}$/3MHz	$\hat{E}$/30MHz	$\hat{H}$/0.3MHz	$\hat{H}$/3MHz	$\hat{H}$/30MHz
DIN VDE 0848	48.4 V/m	16.1 V/m	4.1 V/m	0.128 A/m	0.043 A/m	0.011 A/m
Taylor Series	38.4 V/m	14.6 V/m	7.7 V/m	0.114 A/m	0.036 A/m	0.018 A/m

Aknowledgement

This work was financially supported by the Federal Institute for Occupational Safety and Health (Bundesanstalt für Arbeitsschutz), Germany.

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RADIO AND TV BROADCASTING IN ITALY: A NATIONAL PROJECT FOR HEALTH AND ENVIRONMENT IMPACT ANALYSIS

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INTRODUCTION

The development of radio- and TV-broadcasting in Italy has been characterised in the last decades by an almost complete deregulation on the siting and power of emitters, the only licensing procedure being related to the assignment of transmission frequencies. That has led to a number of antennas which is probably among the highest in the world.

At the same time, the adoption of a European provisional standard issued by CENELEC (1), as well as the development of a national law on the safe exposure of the general public to radiofrequency electromagnetic fields (which is expected to be enforced within a short time), give special importance to the assessment of the health and environmental impact of each plant. A national project for such analysis has recently been launched in Italy, based on a census of all broadcasters operating in the Country.

THE NATIONAL PROJECT FOR HEALTH AND ENVIRONMENT IMPACT ANALYSIS

The project, involving both national and regional health and environment authorities, was launched by the National Institute of Health in 1993 and was expected to develop in the following stages:

1) implementation of a national archive of broadcasters;
2) organisation of regional archives, to be managed by local authorities;
3) check and update of the data;
4) local and global analyses.

Phase 1 was completed in 1994. The data, provided by the Ministry of Telecommunications, were loaded on a data base designed for this project. The archive was officially recognised by the Italian National Institute for Statistics (ISTAT) and included within the National Statistical Program.

Phase 2 is in progress. Subsets of the general data base have been extracted for each Italian region and are being distributed to individual health authorities, which have full responsibility for their management.

Phase 3 has been started only in a few regions, or even in smaller areas (e.g. provinces). The reliability of data is checked by cross-links with registers of local technical services of the Ministry of Telecommunications and by direct inspections. Technical data which may be missing (e.g. characteristics of the antenna, radiation pattern etc.) are also collected and inserted.

Phase 4 will be extensively exploited upon completion of the previous ones. However, preliminary evaluations have been performed, both at national and local level, whose results are summarised later in this paper.

CHARACTERISTICS OF THE ARCHIVE

The national database is installed on a computer Apple Macintosh and is managed by a special software, user-oriented, based on the application program FileMaker Pro. Regional archives are managed by the same software, either in Macintosh or in DOS-Windows environment.

Broadcasters are classified by region, subdivided by province and further on by municipality. Each data set is divided in turn into two parts, for radio and TV emitters, respectively. The database is linked with an archive of demographic data, allowing analysis of the distribution of emitters relative to the population.

Due to the features of the application program, which allows any field of the database to be used as a key, selection or sorting of broadcasters is possible in a great number of ways, e.g. by site, by frequency, by radiated power, etc.

At present, a limited number of fields are available in each record of the data set to describe the space distribution of electromagnetic radiation around the emitter. The possibility of linking an archive of different types of antennas and related characteristics is being explored, to allow a more refined and complete picture of the radiation pattern.

For this purpose, computational programs are also being developed to calculate the space distribution of the electric and magnetic fields around the plant, based on the technical characteristics of the antenna. The calculations could be run from inside the application program, to directly provide the diagrams of exposure levels.

PRELIMINARY ANALYSES

A global evaluation of available data indicates that 56,101 broadcasting plants have been licensed, almost equally divided into TV (21,769) and radio (27,332) stations. The distribution of broadcasters by region has been reported elsewhere (2).

These data clearly indicate an abnormal density of emitters in Italy (with regard to both the population and the surface) relative to other Countries. In the United States, for example, broadcasters amount to about 10,000; more precisely, 8,763 radio- and 1,030 TV-emitters were reported to be in operation in 1993 (3).

In spite of remarkable social differences among different areas of the Country, there is no clear distinction in the distribution of broadcasters in northern and southern regions. The differences seem mainly related to geographic conditions, and to the orography in particular.

Orography also plays a major role in the concentration of emitters on a local scale. Panoramic areas are in fact preferential sites for antennas, which may pose significant problems for exposure, due to the overlapping of radiation from multiple sources. A preliminary analysis of data for Lombardy shows that about 80% of emitters are located in alpine and sub alpine areas, 13% within the Province of Milan, and only 7% in plain provinces which cover about 30% of the surface of the region.

In the pilot study on Lombardy, an analysis has also been performed of the density of emitters with respect to the resident population (Table 1), which seems to be a rather good indicator of possible critical areas.

Table 1. Distribution of radio- and TV- broadcasting emitters, with respect to the resident population, among the municipalities of the Region Lombardy

Number of emitters per 10,000 inhabitants	Number of municipalities
< 1	292
1 - 5	157
5 -10	54
10 - 20	44
20 - 30	14
30 - 40	6
40 - 50	8
50 - 100	8
> 100	2

The density of broadcasters per 10,000 inhabitants exceeds the value of 50, for example, only in 10 out of the 585 municipalities of the Region. On-site experimental measurements have actually indicated that the exposure limits recommended by the IRPA/INIRC guidelines (4) are frequently exceeded in areas open to the public near clusters of antennas located at these sites.

CONCLUSIONS

The national archive of Italian radio- and TV-broadcasters is of fundamental importance to monitor the future development of installations not only from the point of view of the service, but also of the health and environment impact. The experience gained on limited analyses at local scale also indicates its relevance as a tool to identify areas where critical exposure conditions may occur and remedial actions are needed.

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MAGNETIC RESONANCE SAFETY
BIOEFFECTS OF GRADIENT MAGNETIC FIELDS

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ABSTRACT

From the safety point of view, there are areas to consider in operating a Magnetic Resonance (MR) facility. Time-varying magnetic fields, caused by the switched gradient on imaging systems, will induce currents in conductive pathways in the body. There are different thresholds for eliciting excitation for different tissues. Information was obtained from a MR research facility which could be bear in mind when a small sample is investigated although there are limitations upon the use of this information for biological tissues. Because a number of effects in man occur, there are recommendations made by the National Radiological Protection Board (NRPB) in the UK and the US Food and Drug Administration (FDA) has issued safety guidelines as well, which are considered in this work.

INTRODUCTION

MR involve exposure to three types of magnetic and electromagnetic fields. These are, the main magnetic field B_0 , the alternating magnetic field produced by the gradients and the radiofrequency field.

The aim of this work is to point out the effects in the body that may result from the alternating magnetic fields produced by gradients and the important factors which are responsible for inducing a physiologic response to an induced voltage.

During Magnetic Resonance (MR) imaging numerous gradient magnetic fields are turned on and off at rapid rates, for varying amount of times, and to varied peak gradient amplitudes. Placing the body within the MR system and rapidly switching the current on and off within the gradient magnetic field coils, two effects are produced, a) the creation of a gradient magnetic field b) the potential for induced voltages within the patient (and / or other electrical conductors) in the bore of the MR system.

There are potential bioeffects related to gradient magnetic fields from induced currents or voltages in the human body.

CAUSES AND EFFECTS OF TIME-VARYING MAGNETIC FIELDS

The production of eddy currents is a well known physical mechanism whereby time-varying fields can induce changes in biological systems. Biological tissues, containing ions in water, are conductive and it may be assumed that at certain rates and magnitudes of field changes, sufficient density of eddy current could be produced to affect the more sensitive biological tissue.

There are potential bioeffects related to gradient magnetic fields from induced currents or voltages in the human body. It is necessary to determine the amplitude of the induced voltages to know whether or not tissues will be affected by such induced voltages and currents. To calculate induced current densities, it is important to know the electrical resistivity of the induced current loop. This is determined by the tissue through which the induced voltage will attempt to produce a current flow (1).

Several factors are responsible for physiologic responses to induced voltages, including the strength of the static magnetic field, the orientation of the gradient magnetic fields being switched relative to the patient's tissue, the size of the greatest diameter of the patient's body, the frequency of the stimulus, the duration of the induced voltage, the shape of the waveform, the width of the pulse and the sensitivity of tissues. The magnitude of the voltages and /or currents induced within the patient will be determined by several factors as well, these include the electrical resistance in the circuit produced in the tissue, the cross-sectional area of the induced current flow, and the rate of change (versus time) of the gradient magnetic field itself. All induced voltages or currents will occur during the times that the gradient magnetic field strengths are changing - the rise and fall times of the gradients-. The greater the magnitude of the rise and fall times, the greater the induced voltage and current. There is a greater changing magnetic field over time (or dB/dt) as one proceeds further away from the centre and toward the ends of the gradient coils, where the time rate of change of the gradient magnetic fields is the greatest.

In MR, the health consequences due to time-varying magnetic fields in the extremely low frequency range are concerned with nervès, blood vessels and muscles that act as conductors in the body. Rapidly changing magnetic fields induce electric currents in tissue that could be sufficiently large to interfere with the normal function of neurones and muscle. The threshold current density above which the effects may occur varies according to the tissue involved, the pulse width and the pulse repetition frequency of the induced current (2).

Ventricular fibrillation is the most serious adverse response that may be anticipated during exposure to rapidly changing magnetic fields. The heart is most sensitive to stimulation at frequencies between 10 Hz and 100 Hz. 3 A m^{-2} rms is considered therefore to represent a conservative value for fibrillation threshold at frequencies outside this range. Significant disturbance of the heart beat (pump failure) may begin to occur above about 3 A m^{-2} rms.

The visual sensations are referred to as magnetophosphenes (3). Retinal stimulation is considered to be the most sensitive index of induced currents and it occurs when the peak rate of change of magnetic flux density is about 2 T s^{-1}.

EVALUATIONS IN A MR RESEARCH FACILITY

In a volume distribution of electrical current the parameter of importance is the current density (J). The biological system may be regarded as a loop and the important factors to consider is the current density induced in this loop, which can be calculated as:

$$J = 0.5\, r\, \sigma\, (dB/dt) \qquad [1]$$

where, r: the loop radius, σ: conductivity of the sample and (dB/dt:): the rate of change of magnetic field.
There are limitations upon the use of this equation for biological samples since biological tissues are not homogeneous. For example, the skeletal muscle, which acts as the main eddy currents conductor, is anisotropic. The presence of boundaries between different types of tissue will cause distortions to the idealised circular current path and alter the magnitude of currents. Bearing in mind these limitations, it is possible to obtain an estimated current density in the air inside the small gap of an MR research facility and from this, it could be possible to do an approximation of the threshold value in a small sample inside the magnet.

In the experimental work, X and Y flat rectangular gradient coils made by etching technique were used with a permanent magnet (B_0 = 0.59 T, gap: 18 mm). Trapezoidal waveforms were generated with 5V amplitude, rise time 480 µs, pulse width 20 ms, period 26 ms, (dB/dt)air = 4 T/s. Gap for sample: 15 mm (r = 0.0075 m) (4).
From equation [1], the current density in air , Ja, inside the gap of the magnet was calculated as: Ja = 0.037 x 10^{-14} A/m^2 with σa = 2.5 x 10^{-14} S/m and (dB/dt)a = 4 T/s. The current density in any sample is : Js = 0.5 r σs (dB/dt)s
The threshold value (dB/dt)s can be calculated as:

$$(dB/dt)s = (Js/Ja)\, (\sigma a/\sigma s)\, (dB/dt)a \qquad [2]$$

If σs = 0.3 S/m (2) and Js = 2.5 A/m^2 , the threshold value in a small sample will be: (dB/dt)s = 2222 T/s. This value can be compared with other estimations made for different authors. Thresholds values for different tissues and different density currents can be evaluated in this way bearing in mind the limitations of these approximations.

CONCLUSIONS

These last years there has been increased public concern about the bioeffects of gradient magnetic fields.
The health consequences are not related to the strength of the gradient field, but rather to changes in the magnetic field that cause induced currents. Because of this, it is important to obtain information concerning the pulse generation, to examine the conditions for the production of a detectable response and to obtain better understanding of biological effects involved.

This issue must be reconsidered as the maximum strengths and the rise time capabilities (dB/dt) of the newer and faster MR systems continue to improve.

Thresholds values for different tissues and different density currents can be evaluated for a small sample. The previous calculations can be carried out bearing in mind the limitations of these approximations, as was mentioned before. The threshold value obtained in this evaluation was, (dB/dt)s = 2222 T/s.

In view of the rapid progress in clinical imaging as a diagnostic aid, The National Radiological Protection Board recommends continue modifications to the original advise given in 1981. Some of these new recommendations are, (NRPB 1981,1983) (5), a) the root mean square value of rate of change of dB/dt should not exceed 20 Ts^{-1} for pulse widths above 10 ms, b) for periods of magnetic field change shorter than 10 ms: $(dB/dt)^2_{rms}\ t < 4$ where t is the duration of the change of the magnetic field in seconds.
For sinusoidally or other continuously varying periodic magnetic fields, the duration of the change can be considered to be the half period of the magnetic flux waveform.

The US Food and Drug Administration (FDA) has issued safety guidelines as well and suggested operating limitations for MR imaging devices regarding the gradient magnetic fields associated with MR imaging procedures Concerned with the rate of change of magnetic field, the limit patient exposure to time-varying magnetic fields with strengths less than those required to produce peripheral nerve stimulation, was suggested. An alternative is to demonstrate that the maximum dB/dt of the system is 6 T/s or less. There are other alternatives related to axial and transverse gradients (6).

Working with alternating magnetic fields the NRPB or/and the FDA recommendations must be considered.

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ACKNOWLEDGEMENTS

I wish to express my sincere thanks to Dr. R A Lerski, Head of Medical Physics of the University of Dundee, Dundee, UK, for providing technical support.

THE EUROPEAN PRE-STANDARD ENV 50166 "HUMAN EXPOSURE TO ELECTROMAGNETIC FIELDS"

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SCOPE AND CONCEPT

The CENELEC Prestandard ENV 50166 (1) deals with the prevention of adverse effects of human exposure to electromagnetic fields within the frequency range 0 Hz-300 GHz. It is based on the well-established short term effects, the stimulation of nerves and muscle tissue at low and heating at high frequencies. Reports on long term effects such as the induction or promotion of cancer have been scrutinised and it was found that the current available evidence had not established a connection. Basic restrictions, which must not be exceeded, are given for biologically relevant quantities and complemented by reference levels which are more readily measurable thus facilitating the checking of compliance. The Prestandard does not apply to the deliberate exposure of persons during medical research, diagnosis or treatment. Furthermore, safety hazards associated with the ignition of flammable materials or the triggering of explosive devices are not covered, and the interference of permissible fields with implants cannot be excluded.

LOW FREQUENCIES (0-10 kHz)

At low frequencies there are basic restrictions in terms of induced current density, contact current, electric field strength and static magnetic flux density. The ENV sets 10 mA/m2 as the basic restriction for workers within the frequency range 4 Hz-1000 Hz. The current density restriction is set inversely proportional to the frequency below 4 Hz and directly proportional to the frequency above 1000 Hz. This accounts for the specific stimulation characteristics of excitable cells. For the general population a precautionary factor of 2.5 was chosen. Harmful indirect effects are prevented by limiting the contact current, to 3.5 mA over the entire low frequency range for workers and to 1.5 mA for the general population.

High electric fields can make themselves felt through surface effects. To prevent annoyance a basic restriction is set for electric fields at 30 kV/m above 0.1 Hz (42 kV/m at 0 Hz). Similarly, a basic restriction for static magnetic fields at 2 T prevents vertigo and nausea.

Reference levels as limits for the electric and magnetic field strengths are derived from the basic restrictions using worst case assumptions for the geometry and position of the body and the relevant organs, with respect to external fields. Thus compliance with reference levels automatically ensures compliance with the basic restrictions on induced current density. However, reference levels may be exceeded if the basic restrictions are met.

HIGH FREQUENCIES (10 kHz-300 GHz)

The ENV uses the following quantities as basic restrictions in the high frequency range: Induced current density and contact current at frequencies up to 10 MHz and 3 MHz, respectively, the specific absorption rate SAR with different values for whole body, limbs, local heating and the specific absorption SA for pulses of a duration of less than 30 μs at frequencies above 300 MHz.

The current density is set for workers at f/100 mA/m^2, whereas the precautionary factor of 2.5 is maintained for the general public. Between 10 kHz and 100 kHz the contact current limit increases proportionally to the frequency and, between 100 kHz and 3 MHz, remains constant at 35 mA and 20 mA respectively.

At frequencies of around a few MHz the specific absorption rate SAR takes over from the induced current density to become the significant dosimetric quantity for establishing exposure limits. The threshold for effects which are considered detrimental to health is observed at 4 W/kg, averaged over a 6 min. time interval and over the whole body. This corresponds to a systemic temperature increase of less than 1°C at normal conditions. To derive the basic restriction for workers a safety factor of 10 is applied which gives an SAR of 0.4 W/kg. For the general public 0.08 W/kg was chosen. In the high frequency range both rms and peak values are set for the reference levels. Above 10 MHz, reference levels are also given for the mean power density.

DISCUSSION

Some European countries have already developed national standards or guidelines for the protection of the population from electromagnetic fields (2,3,4). The CENELEC Prestandard is the result of their national electrotechnical committees' harmonization effort. Problems to overcome were the questions whether a one tier

or a two tier structure of the standard should be adopted, and how to define the two levels of protection. For example Great Britain's NRPB guidelines give one set of limits which are valid for everyone whereas Germany's DIN/VDE Standard and Austria's ÖVE Standard distinguish between controlled/uncontrolled areas and workers/general public, respectively.

Beside CENELEC IRPA/ICNIRP and CEU have developed exposure limits at international level. The IRPA/ICNIRP limits are confined to 0 Hz, 50/60 Hz and the frequency range 100 kHz to 300 GHz (5,6,7). The limits of the CEU's draft Directive are related to workers only (8). The draft Directive defines exposure limits as upper limits as well as threshold values below which no adverse health effects are expected. In the region between threshold values and exposure limits there is a set of three action levels. These are not limits but rather field levels, the exceedance of which requires certain countermeasures. At the lowest action level, workers should be informed about their exposure situation, provived with protective gear and training. At the next action level a program for the reduction of exposure has to be developed and areas have to be marked off. Activities at levels higher than the 3rd action level are regarded as hazardous, authorities have to be informed and a systematic health surveillance has to be provided for exposed workers.

This discussion attempts to compare CENELEC's ENV limits with those limits given by IRPA and CEU, being aware that, particularly in the case of field strengths, such a comparison can only be approximate because of differing definitions.

		CENELEC	IRPA	CEU
induced current density , head and trunk	[mA/m^2]	10	10	10
electric field strength	[kV/m]	10	10	19,6; 12,3; 6.1
magnetic flux density	[mT]	1,6	0,5	0,64; 0,4; 0,2
contact current	[mA]	3,5		1,5

Table 1:Limit values of CENELEC ENV 50166, IRPA guidelines and CEU draft Directive at 50/60 Hz for exposure of workers over a full working day.

		CENELEC	IRPA	CEU
frequency range	[Hz]	$1,0.10^4$-$3,0.10^{11}$		$1,0.10^5$-$3,0.10^{11}$
SAR 6 min.	[W/kg]			
whole body		0,4	0,4	0,4
extremities		20	20	20
head and trunk		10	10	10
SA	[mJ/kg]	10		10
contact current (0,1 - 3 (100) MHz)	[mA]	35	(50)	50

Table 2: High frequency limit values of CENELEC ENV 50166, IRPA guidelines and CEU draft Directive for exposure of workers over a full working day.

Generally, the regulations of CENELEC, IRPA/ICNIRP and CEU are based on the same criteria. For workers there is full agreement between the basic restrictions of CENELEC, the basic criteria of IRPA/ICNIRP and the exposure limits of CEU for the induced current density and SAR, which are the starting points for deriving the respective field strengths (e.g. Tab.1 and 2). Differences exist between the contact current limits of CENELEC and CEU, established to control indirect effects.

Figure 1 shows the reference levels of the ENV in comparison with the electric field limits of the IRPA/ICNIRP guidelines and the medium action level of the draft Directive. Figure 2 displays the corresponding magnetic flux density limits. There is a satisfactory agreement between the three regulations. The differences however reveal that there is a need for harmonisation. The same holds true for the exposure limits for the general population as set by CENELEC and IRPA/ICNIRP. For details the reader is referred to the documents in the reference list.

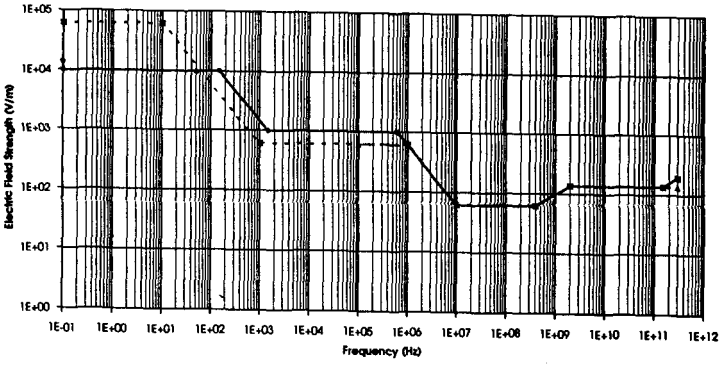

Figure 1: Electric field limits of CENELEC (full line, dots), IRPA/ICNIRP (dashed line, triangles) and CEU (dotted line, squares) for workers exposed over a full working day

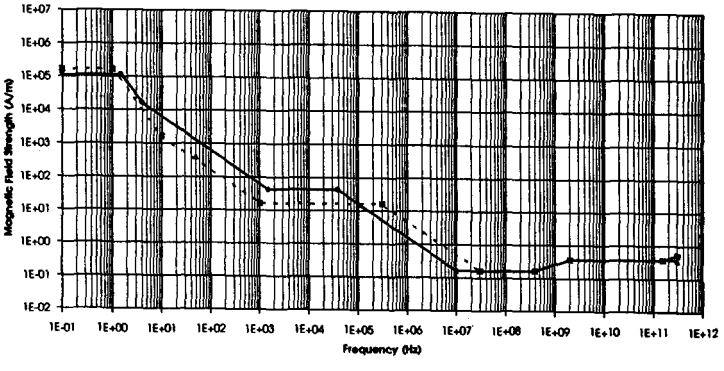

Figure 2: Magnetic field limits of CENELEC (full line, dots), IRPA/ICNIRP (dashed line, triangles) and CEU (dotted line, squares) for workers exposed over a full working day

ACKNOWLEDGEMENT

As chairman of CENELEC's Technical Committee TC 111 which developed ENV 50166 I have the privilege to thank the national committee members and in particular the chairs of the subcommittees Dr. Gabriel and Dr. Maddock as well as the secretaries Dr. Hutzler, D.I. Krause and Dr. Mariutti for their outstanding work.

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IRPA9
1996 International Congress on
Radiation Protection
April 14-19,1996
Vienna, Austria

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Acceptance _____
Mini-Presentation _____

PAPER TITLE: RF Induced Body Currents: Practical Measurements

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MAJOR SCIENTIFIC TOPIC NUMBER: 4.1 or 7.2

ABSTRACT:

With the growing interest in induced current measurements and the presence in several international standards, several instruments for measuring induced currents are now commercially available. Comparing induced current measurements using differing measurement principles and instruments from different manufacturers with rule-of-thumb estimates and free-field measurements provides insight into variations in observed data.

Data is also presented using clamp-on induced current sensors to estimate contact currents.

Typical measurement situations using induced current measurements are presented in comparison with ICNIRP Guidelines on Protection Against Non-Ionizing Radiation and the IEEE C95.1-1991 Standard for Safety Levels with Respect to Human Exposure to Radio Frequency Electromagnetic Fields, 3 kHz to 300 GHz.

ASSESSMENT OF COMPLEX MICROWAVES OCCUPATIONAL EXPOSURE IN RADAR MAINTENANCE ACTIVITY

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INTRODUCTION

The modern development of the society has determined the increase of thousand times greater than the natural fond of the humankind exposure to a complex combination of electromagnetic man-made fields and radiations of extremely various strength and frequencies (4).

A special contribution to this environmental change has had in the last decades the appearance and the explosive development of the microwaves generating appliances such as radars used in a great variety of military and civilian applications and which essentially contributes to the electromagnetic pollution (4,6,7).

In the above mentioned context which firstly interests the occupational environment, it is necessary to improve the exposure limits, as well as, the emission standards, in order to better protect the human health and well-being (7). From this point of view, the estimation of the microwaves occupational exposure risk constitutes, alongside the health status assessment, one of the priorities of the Occupational Health because the theoretical and practical problems related to the bioeffects of this kind of radiations are far to be clarified.

Our study has been carried out in a factory where one performs research, production and especially maintenance of microwaves generating devices.

OBJECTIVE

To thoroughly analyse the working conditions, both ergonomical and physical, and to colligate them in order to obtain a better exposure assessment for the improvement of the protection of the microwaves exposed workers.

METHODS

Ergonomical analyse has comprised the study of all the postures, movements, gestures, physical and psychological charges of the workload, as well as, the study of all the partial timing for obtaining reliable photographs of the day-time work. The technical features of the radars (constructive as well as functional) and the relative distances between them and the workers in every phase of the work have also been studied.

The assessment of the presence, frequency and the average power density of the microwaves has been performed by a Russian PO1 measuring instrument with a set of probes able to measure electromagnetic fields in the range 10 Mhz - 12 Ghz, the provided parameter being the average power density (mW/cm^2). Multiple measurements for each type of radar and for each position in the activity have been done according to the methods described in the NCRP Report No. 119 "A Practical Guide to the Determination of Human Exposure to Radiofrequency Fields" (1993), (3).

As a result of the correlation between the ergonomical studies and the physical measurements, average and local Specific Absorption Rates (SARs) have been assessed by using the methods and diagrams from the "Radiofrequency Radiation Dosimetry Handbook (Fourth Edition)", C.H. Durney,1986, (1).

RESULTS

The activity is performed in relative small work-spaces with glass walls in metal frames and there are multiple, unshielded and rather close microwaves generating devices.

The working position is generally sitting down at tables and benches, but also standing up and in bent position. The motor solicitation both static and dynamic is rather reduced with the preponderance of the gesture that requires precision. The psychological solicitation implies: the thinking, the constant attention, the memory, the visual and auditive perception.

The generators emit in the range 0.2 - 10 Ghz with mean emission powers from some watts to hundreds of kilowatts and the waves are usually modulated as impulses. During the tests the devices usually run at nominal parameters and the shielding carcasses are removed, so multi-directional and variable power leakages occur generating complex field distribution, with extremely various power densities and frequencies.

It is to be mentioned that the physical phenomena like reflection and dispersion of the microwaves favoured by the characteristics of the workplaces could determine sensible increases of the local fields up to four times owing to the stationary waves and/or to the multiple interferences. The work positions create exposures both in near and in far field which especially concern the hands, the head and the fore part of the trunk. The movements multiply the complexity of the exposure and make more difficult its appropriate assessment. Thus it is obvious that there are complex conditions of continuous (sometimes quick) variations in space and time of the frequencies and power densities with frequently potential additional effects.

The measurements in normal work conditions for 15 types of generators in each stage of activity are synthetically presented below in terms of the frequency.

For the frequencies 8-10 Ghz we have found average power densities between 0.01 and almost 10 mW/cm^2, with averages for the different points that define the workplace of 0.8-2 mW/cm^2. At frequencies around 2-6 Ghz, the average power densities were 0.04-0.5 mW/cm^2. At frequencies smaller than 2 Ghz the measured values were between 0.3-1 mW/cm^2. It is to be especially mentioned the found values of many mW/cm^2, that exists in accidental conditions at many devices but the weight of which during the worktime is difficult to be established.

Based on the photographs of the day-time work, and on the physical measurements, we have estimated a wide range of various whole body SARs: 0.01 - 2 W/kg. There is a great variation during the day-time work of the estimated SARs at the same subject, which shows on one side the complexity of the exposure conditions and on the other side the wide range of possibilities of interference of the microwaves with the human organisms. There are also many inter-individual differences depending of the device being repaired, of its situation in the workplace and of the specific phase of the activity.

Trying to calculate some time-weight averages of the whole body SARs, we have found values of 0.08-0.31 W/kg which are under the threshold considered by IRPA and WHO as a basis for the setting of the occupational exposure limits ().

The local peak SARs, even more difficult to estimate, were greater for the head (neck) and hands (wrists), their values being of few ten times greater than the average whole body SARs. However, there is a highly non-uniform spatial and temporal distribution of local SARs which reflects the same complexity of the exposure conditions.

CONCLUSIONS

The workload implies continuous and various movements with consecutive great variations of partial and total exposures to complex field distribution, with extremely various power densities and frequencies, generated in a close and small workspace by multiple unshielded generators.

Although we have found whole body SARs values of 0.08-0.31 W/kg which are under the IRPA threshold of 0.4 W/kg (2,5,6), we consider that the peak values of both whole body SARs and local SARs are quite important, as well as, the pulsed character of the emission and consequently we consider that supplementary limits for pulsed microwaves as recommended by the documents of IRPA and WHO, are to be further enhanced and implemented in national legislation.

Beyond its laborious characteristics, this method allows a rather reliable exposure assessment, which could be used in setting limits for special exposure conditions, in order to avoid hot spots and, consequently, to improve the protection measures.

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PULSED RADIOFREQUENCY MICROWAVE FIELDS AROUND A QUADRUPOLE PARTICLE ACCELERATOR: MEASUREMENT AND SAFETY EVALUATION

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INTRODUCTION

Pulsed radiofrequency microwave radiation (RFMR) fields occur during the use of high power microwaves in plasma heating in fusion research, plasma and solid state diagnostics, particle accelerators and collidors, pump sources in lasers, material processing as well as in high power radars. This paper describes the experimental work done at Trombay for measurement of pulsed RFMR fields in the working area of a radiofrequency quadrupole (RFQ) accelerator with the use of a meter calibrated in continuous field and interprets the observed fields in the light of existing protection criteria for pulsed RFMR fields(1).

EXPERIMENTAL DETAILS

Experimental assembly of the RFQ accelerator and the asociated equipment is spread over an area of about 30 m^2. Figure 1 shows the layout plan of the assembly. The location marked R.F.chokes contains two chokes, RFC-1 and RFC-2. The radiofrequency microwave equipment operating at 55 MHz is confined along the west side wall of the hall housing the accelerator. The measurement of the electric field strength has been made using General Microwave Corporation, model 40, radiation hazard monitor. The meter is graduated to read continuous fields for frequencies ranging between 200 kHz and 40 GHz.

SALIENT OBSERVATIONS

The response of the meter has been corrected for the duty cycle(2). The duty cycle of the pulsed power is 1.6 percent. Table 1 lists out corrected power densities for pulsed fields. Of all the locations at which power density measurements have been made, those in the vicinity of two radiofrequency chokes RFC-1 and RFC-2 show maximum leakage in the present development stage(3). Pulsed field power densities over the working area have been depicted in figure 2 in the form of a three dimensional representation of the electric field strength around the RFQ accelerator. The distances shown, are on the floor along and at right angles to the entrance to the hall.

CONCLUSIONS

Observed fields in the working area of the RFQ accelerator at Trombay are much lower than the peak power limits recommended in their standards by world bodies like INIRC/IRPA, ANSI and NRPB to prevent high specific absorption associated with exposure to increasingly shorter pulse width high amplitude fields(4,5). However in view of the increasing use of these fields in nuclear energy and allied applications and inadequate present day understanding of the mechanism of interaction with human body, there is continuing need for such studies(6).

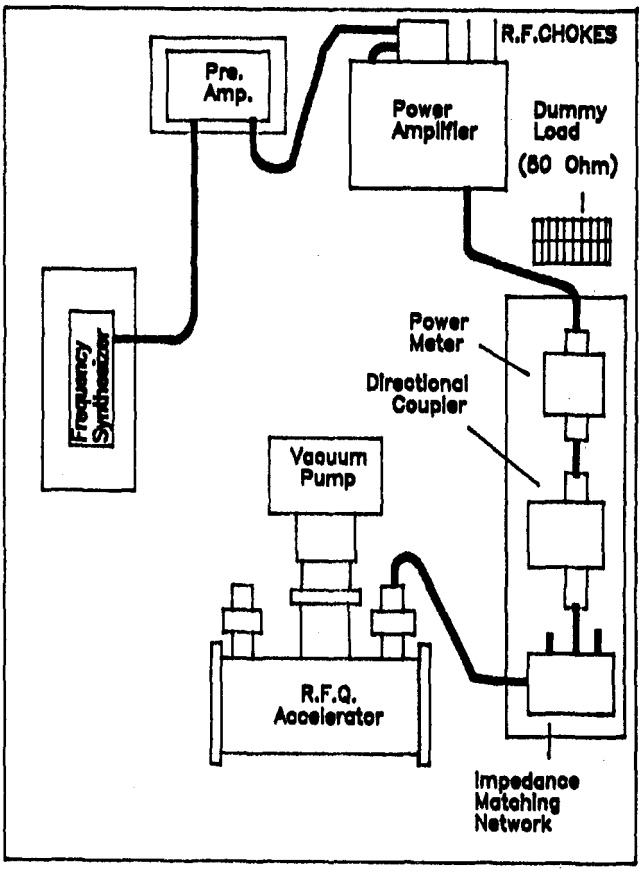

Figure 1: Layout plan of the RFQ accelerator located in the Van De Graaff building at BARC complex, Trombay, Bombay.

Table 1

Variation in leakage power density in pulsed mode of operation at RFC-1 and RFC-2 locations of power amplifier, with change in the operating power of the system

S.No.	System Power (watts)	Power density (mW/cm^2) RFC-1	RFC-2
1	200	3.75	3.75
2	250	4.37	5.00
3	300	5.62	5.62
4	350	6.25	6.87
5	400	6.87	7.50
6	450	7.50	8.12
7	500	8.12	10.00

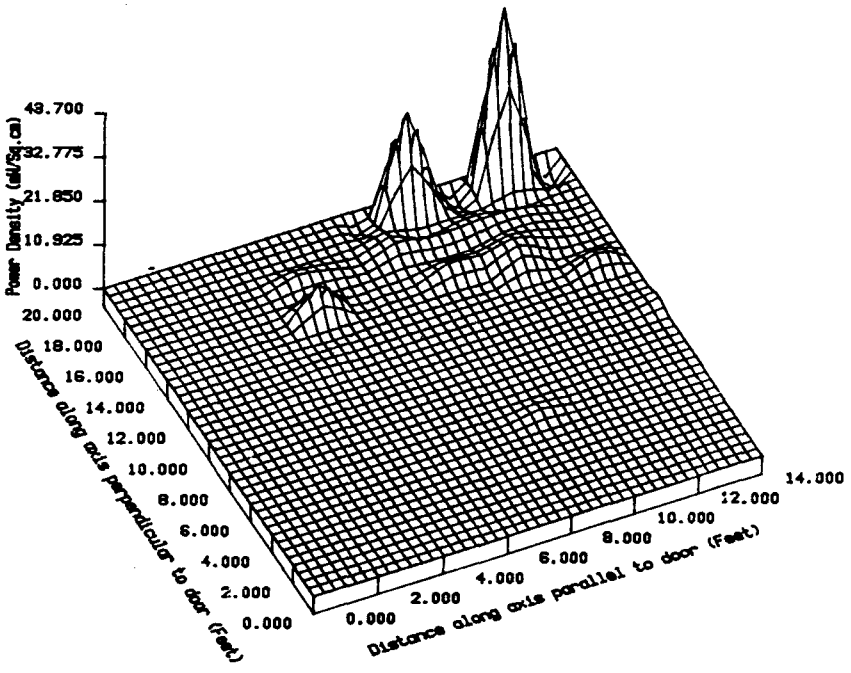

Figure:2. Three dimensional representation of the pulsed electric field strength around the RFQ Accelerator working area.

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DESIGN AND CONSTRUCTION OF A BROAD-BAND ELECTRIC FIELD PROBE

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ABSTRACT

The design of a broad-band electric field probe based on a resistive film diode antenna on RT/Duroid substrate to measure the electric RF/MW fields as constructed at the National Radiation Protection Department (NRPD) of the Atomic Energy Organization of Iran (AEOI) are described in this paper. A square law diode detector with a matching circuit and also low pass filter have been used to produce a dc current proportional to the square RF voltage across the resistive antenna gap. A double-strip coplaner waveguide has also been designed to transfer this dc current to an amplifier with an output signal showing the electric field intensity in one direction. By using three mutually orthogonal resistive antennas, an isotropic electric field probe was made. All parts of this probe have been completely modeled and solved by the MATLAB computer program to determine the optimum values of the elements of the probe. The frequency response of the probe has also been theoretically found to be flat in the range 0.8 to 3 GHz. It was found to be quite satisfactory compared with those of similar probes commercially available. The probe is being used routinely in practice.

INTRODUCTION

Due to widespread applications of radiofrequency and microwave (RF/MW) sources in different disciplines, measurements of exposures from such fields are necessary for protection of workers and public. This requirement in Iran has been enhanced by the enforcement of regulations of the Radiation Protection Act of Iran (1). During the past two decades, significant advances have been made in instrumentation and techniques for measuring complex RF/MW fields applying resistive thin film screens (2) and resistive dipole antennas (3). These techniques have been investigated in this study both theoretically and experimentally for design and construction of a wide-band isotropic electric field probe using a three mutually orthogonal resistive dipole-diode. The design characteristics and performance of the probe as constructed at the National Radiation Protection Department (NRPD) are presented and discussed in this paper.

DESIGN CHARACTERISTICS AND EXPERIMENTS

By using a thin film technology, a resistive dipole antenna on an RT/Duroid substrate has been constructed with a design to have a wide-band antenna with a zero current at the end point and a maximum current at the center. A hyperbolic sinusoidal current distribution on the antenna was used to calculate the effective electrical length of the antenna and the open circuit RF voltage in the antenna terminal. A chip diode detector (which has an exponential characteristics) on the gap of the antenna was used. Whenever it operates in it's square law region, the dc component of the output current will be proportional to the applied squared voltage to the diode (4). Both the diode and the antenna impedance have dependency on the frequency so that the variation in frequency can affect them by a loading effect. Therefore, a matching circuit was used to cancel the loading effect (3). By this method, a relatively constant dc output for a constant amplitude of RF/MW exposure in different frequencies was obtained. To eliminate the ac component in the output of the diode and also to isolate the antenna and matching circuit from the other parts, a RC low pass filter was used. To transmit the

detected signal to the preamplifier, a resistive thin film symmetrical double-strip coplaner waveguide was used. This transmission line operated as a low pass filter and it provided an additional ac compensation (5). The dc output signal of the transmission line was amplified for monitoring of a high impedance meter. Figure 1 shows the circuit of one dipole-diode combination as designed and constructed in this work.

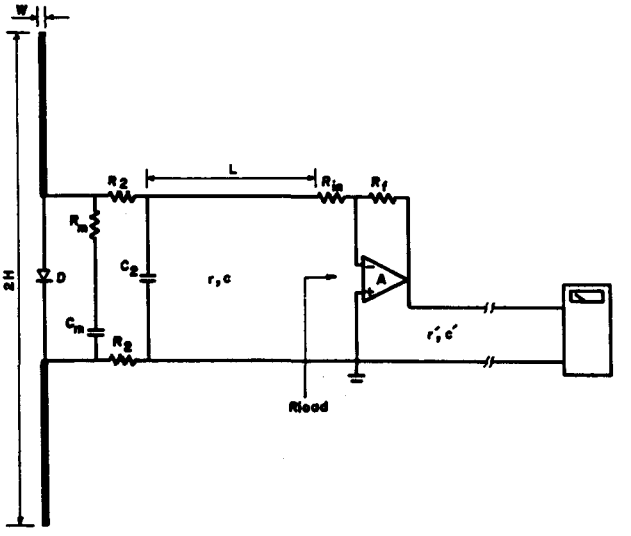
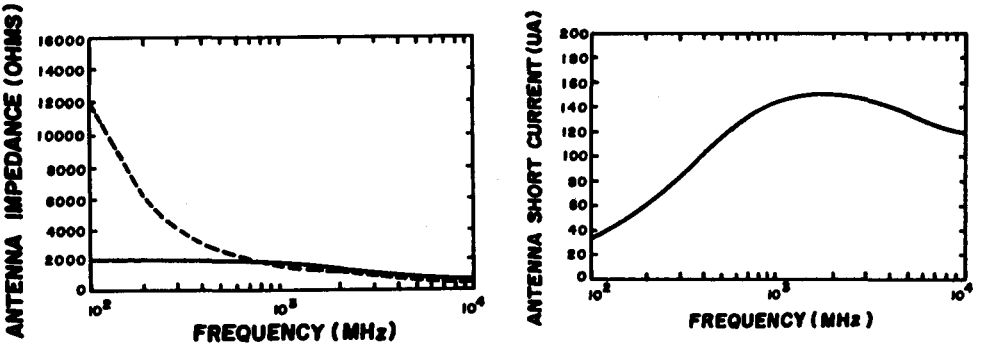


Figure 1. The circuit model of one directional dipole-diode combination RF/MW detector.

The electric field probe has been constructed based on a three orthogonal dipole-diode combination. Each dipole-diode is like the model shown in Fig. 1. A computer program was written using MATLAB to determine the optimum values of the elements of the probe such as antenna half length (H), dipole width (W), transmission line characteristics, etc. The output of the program for half length of the dipole was found to be 2 cm. The width and resistance of the dipole are parameters which limit the lower cut off frequency (2). A narrower width increases the lower cut off frequency but it makes the analysis easier. However, there is a trade off between the width of the dipole and the frequency of interest which was determined to be from 1 to 3 GHz. By running the program, the width of the dipole and antenna resistance were found to be respectively 0.1 cm and 3 kΩ per leg. Figure 2 shows the real and imaginary parts of the antenna impedance and it's short current versus the frequency.



(a) (b)

Fig. 2. a) The real and imaginary parts of the resistive dipole impedance; b) the antenna short current.

By modeling the diode and the other parts of the probe and by running the program at different frequencies of interest, the frequency response of the probe was determined, as shown in Fig. 3. It can be concluded that the probe has a response flat in the frequency range of 0.8 to 3 GHz which matches with our preset frequency range of interest. The probe was constructed based on the output results of the program as shown in Fig. 4. By comparative calibration of the probe, the RF/MW exposures in the environments of such fields and in the working areas were measured. The response of the probe was compared with those of other available field probes and it's response was acceptable.

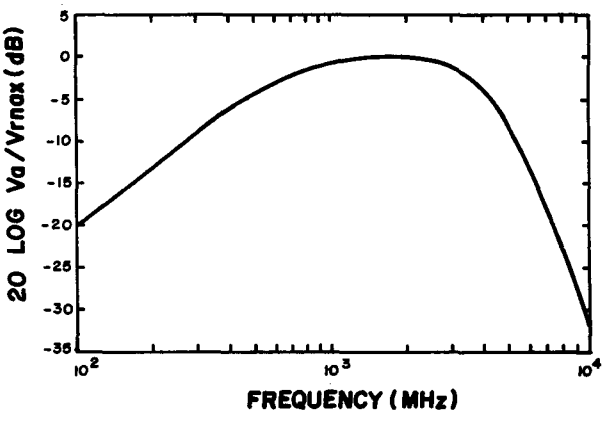


Figure.3 The frequency response of the probe.

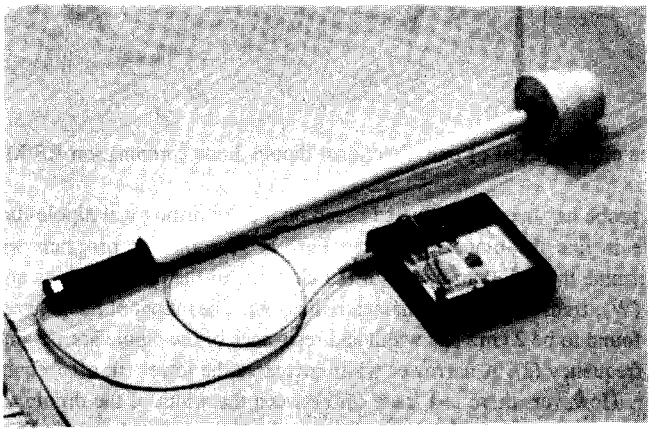


Figure 4. The electric field probe as constructed in this study.

CONCLUSION

In this study, a wide-band RF/MW measuring probe was designed and constructed which showed satisfactory performance characteristics and it has been applied in practice. Studies are underway to further upgrade the system.

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BROAD BAND SENSORS CALIBRATION BY A GTEM CELL

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ABSTRACT

Increasing interest of public for non ionizing radiation hazard and local authorities surveillance necessity demand for -easy to use, reliable measurement instrumentation [1]. Triaxial sensors, with their inherent large frequency response, are excellent candidates to reduce in field difficulties related to narrow band tools utilization, as antennas connected to spectrum analyzer. GTEM cell [2,3] is a relatively novel tool in EMC field, the use of which seems overcome some limitations of standard TEM cells [4] and antennas. The continuos improvement in theory of operation and manufacturing broadens its field of application from canonical electromagnetic compatibility radiated immunity and radiated emission tests [5,6]. The large operation frequency range (from DC to some GHz) and the compact design, well suited to laboratory use, offers the opportunity of conduct fast tests at a convenient location.

1. INTRODUCTION

A preliminary GTEM cell (EMCO 5302 model) characterization is presented in order to investigate its capability in broad band sensors calibration. VSWR, Electric Field Uniformity, etc., are measured and their values compared to specifications. A compact broad band electric field sensor is used to assess consistency and repeatability of measurements. Comparison between GTEM data and field strength values evaluated by this sensor, independently calibrated by manufacturer, is displayed and the optimum position of the sensor inside the test volume is investigated. In order to optimize test time and repeatability an *ad hoc* software is implemented to drive all used instrumentation. Finally, critical evaluation of GTEM performances is presented and suggestions are given to improve its reliability.

2. MATERIALS AND METHODS

In fig. 1 is presented a scheme of GTEM cell we investigated about. This model is sufficiently compact to occupy a modest room volume, and its weight is tolerable by common bench.

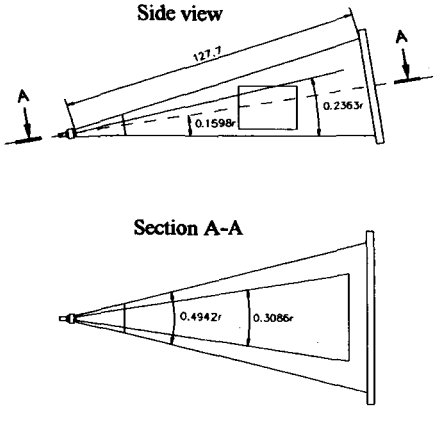

Fig. 1: Side and plant view of EMCO 5302 GTEM! model.

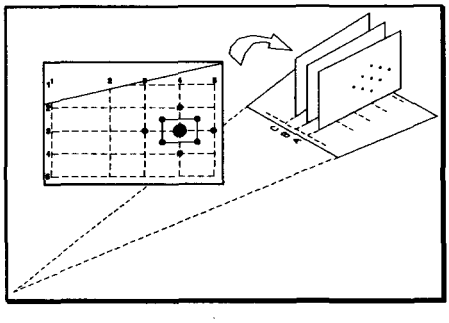

Fig. 2: Layout of measurement points. On each plane orizontal spacing is 4.5 cm and vertical spacing is 3 cm. Points in the inner surface were chosen midway closer neighbors. Planes separation ranges from 3 cm near the apex cell to 4.1 cm near the end cell.

After first evaluations was evident that results were not affected by door position, therefore it was removed to improve inside access. All measurements were so performed using lateral door aperture.

Field uniformity has been evaluated in three vertical planes bounding a volume of trapezoidal shape of approximately 8 (w) x 13.5 (l) x 14 (h) cm^3 (fig. 2). On each plane twenty-five measurement points were selected, even if only highlighted ones are presented. The central point is halfway septum height.

Experimental set up differs according to parameter to be measured. Time domain measurements were performed with a TDR HP 54121T and a RF Impedance Analyzer HP 4191A directly connected to cell input. Field measurements were performed using a HP 437B Power Meter, a Farnell PSG 1000 RF Source, a Kalmus 747LC 50W amplifier, a Werlatone 1795 coupler and an Alenia broad band Electric Field sensor 18RV1001-1. All instruments were driven by appropriate software LabVIEW based (National Instruments), developed by the authors, *via* a GPIB bus. In fig. 3 is shown a typical test set up.

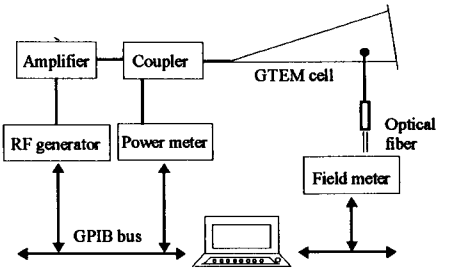

Fig. 3: Electric Field measurements test set up.

3. TDR MEASUREMENTS

Fig. 4 presents the TDR of cell in two different configurations. Traditional time axis is replaced with length axis to improve readability. Connector section shows most of the noise, as expected. Some variations are present in the transition zone where septum meets resistors. It is worth that, substantially, impedance does not exceed 54 Ω in the working region (approximately 78-105 cm far from apex). Various tests were made to assess influence of external factors on data repeatability and a negligible impedance variation is only present in correspondence with side aperture. As a matter of fact, TDR measurement with opened side door presents variation of about 1 Ω.

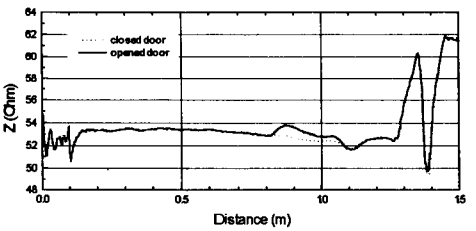

Fig. 4: TDR measurement of GTEM cell with closed and opened side door.

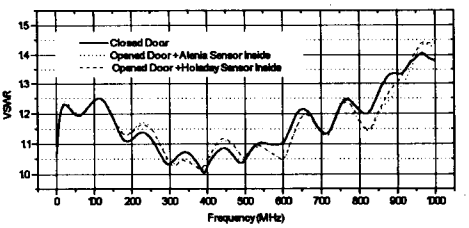

Fig. 5: VSWR measurement of GTEM cell: closed side door (continuum), opened side door with Alenia sensor inside (dotted) and opened side door with Holaday sensor inside.

Further analysis was made computing VSWR of the cell (fig. 5). Specifications declare a value (1.5:1) higher than our result, but manufacturer documentation includes a graph where VSWR is in substantial agreement with present data. Most range shows a value less than 1.2 while a maximum of 1.4 is found at 966 MHz.

Sensor insertion changes a little the shape, but not in a significant way. On the other hand, different sensors (e.g., Holaday STE sensor) were proved to identify eventual problems, but results show no significant differences.

4. FIELD MEASUREMENTS

Electric Field Strength measurements have been used to state field uniformity inside the cell. This parameter is relevant because it is related with the maximum DUT volume the cell can house. The generated fields were around 100 V/m with a 300 V/m full range.

In fig. 6 a comparison between the central point (position B[3,4]) and the rightmost one (position B[3,5]) is shown. The maximum difference is 2.7 dB at 940 MHz.

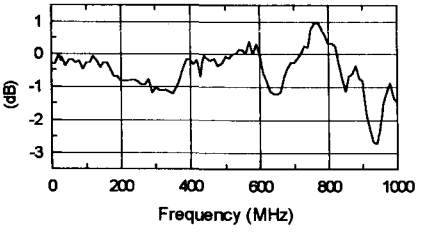

Fig. 6: Electric Field Uniformity on plane B between the central point (position B[3,4]) and the rightmost one (position B[3,5]).

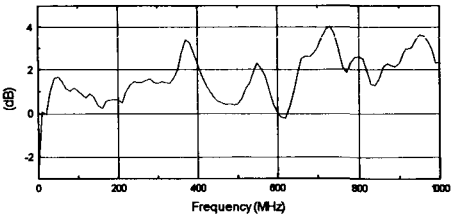

Fig. 7: Electric Field Strength difference between GTEM cell values and Alenia sensor at the central point position.

It is worth to point out that most of data differs for less than 1 dB. Analogue results hold for the other positions. These preliminary results allow a moderate optimism in the optic of a possible use of GTEM cell as sensor calibrator.

Fig. 7 shows the results of comparison between Electric Field Strength measured at position B[3,4] (central point) by the Alenia RV181001-1 sensor (calibrated by the manufacturer) and the Electric Field Strength as computed by geometrical GTEM parameters. The latter calculus was implemented by a dedicated software and correction for coupler insertion loss, coupler coupling factor and GTEM VSWR were taken into account. In the worst case the mismatch is comprised in 4 dB.

5. CONCLUSIONS

GTEM cell is becoming a valuable tool in performing some specific test up to now realizable only by means of different integrated approaches as TEM cells and anechoic chambers or large OATS sites. In particular, some interesting perspectives open in the field of sensors calibration. Drawbacks found at certain frequencies suggest that further investigations are required, but results in the most of frequency range are valuable ones.

Attention must be paid in frequency characterization of instrumentation in use. In our experience, common ten points often delivered with instrument by manufacturer are inadequate to probably describe the behavior in the full working range and interpolation is a rough expedient to solve the lack of data.

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IRPA9
1996 International Congress on Radiation Protection
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PAPER TITLE

EFFECTS OF RADAR OCCUPATIONAL EXPOSURE ON THE MALE FERTILITY

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MAJOR SCIENTIFIC TOPIC NUMBER 2.2. (see page 7)

ABSTRACT (See instructions overleaf)

Effects of radar occupational exposure on the male fertility

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Objective: The estimation of possible harmful effects upon the male fertility due to the occupational exposure to low and medium levels of pulsed microwaves.

Methods: 55 males occupationally exposed to microwaves have been studied compared to matched controls. Exposure assessment and consequently thorough investigation of the fertility have been made: complex fertility questionnaires concerning sexual dynamics troubles, reproductive history, pregnancy outcomes, time to pregnancy, spousal abortion rate, birth weight, gestational age, congenital malformations, etc. Serum testosterone analysis and seminal analysis (sperm count, volume, motility) have also been made.

Results: Exposure assessment has pointed out whole body SARs between 0.01 and 1W/Kg. Epidemiologic research has shown significant increase of sexual dynamic impairment, significant increase of time to pregnancy and decrease of the fertility. Low levels of serum testosterone, without statistical relevance, have been recorded. Slight but statistically significant decreases in sperm count and motility have also been found.

Conclusion: Since the exposed group dimensions are rather low, the statistical power of our study is somehow slight but, however, it must be considered at least an alarm signal and a stimulus towards further research in order to elucidate these found harmful fertility effects of the microwaves.

SPECIFIC FEATURES OF PSYCHOLOGICAL REACTIVITY IN PROFESSIONAL EXPOSURE TO PULSED MICROWAVES

Gh. Bălăceanu, R. Dănulescu

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Iaşi, Romania

OBJECTIVE AND CHARACTERISTICS OF EXPOSURE

The literature contains controversed data concerning nervous and behavioral effects of chronic human occupational exposure to pulsed microwaves. Our paper follows some psychological changes in 49 occupationally exposed subjects, technicians and engineers with 38,9±3,2 years of mean age and 15,5±5,1 years length of service in activities of maintenance of radars in the vicinity of some generators unprotected by carcass or with inadequate protection, in a relative limited space with much and generally unshielded sources. The frequencies are from 0.2 to 10 Ghz with the mean power of density from 0.1 to above 2 mW/cm^2, with peak power density exceeding thousands times the mean power density. The estimated SARs are between 0.01 and 1-2 W/Kg, greater especially for head and hands.

METHODS

Both in exposed (E) subjects and control (C) subjects (36 Ss from the same enterprise with similar profession, mean age and length of service, without exposure), were used the followings tests:

- the Pointage Test (to punctuate the middle of 100 squares with the side of 1 cm);
- the Tapping Test (the sum of points tapped with the left and right hand in six seconds);
- the Bourdon-Amfimov attention test (the barrage of O and C letters in time of 10 minutes);
- the Praga Distributive Attention Test;
- the I_2 Intelligence Test (with the subtests: analogies, number series, useless words and antonyms);
- the Cattell Anxiety Questionnaire (5 PF);
- the Subjective Symptoms Questionnaire (from the Swedish Toxicopsychological Battery);
- an original test (Gh.Bălăceanu) that involve distributive attention, quickness, manual accuracy, simple discrimination of shapes and colours (the task is to place 36 cylindrical pieces with a drawed face in triangle square or circle coloured in red, yellow, blue or green, on a table with circles containing the same figures arranged at random);
- the reaction time test (series of 10 red or blue visual stimuli for each hand).

The subjects were examined in one session with pauses between tests. The comparisons between the exposed and control group were tested by using the "z" or "chi-square" test.

RESULTS

Tests	Measured variables	Mean results				Test of significance
		Exposed Ss	SD	Control Ss	SD	
Reaction time	time in ms	341.8	73.8	310.0	54.1	z=2.23 p<0.05
Pointage	time in seconds	58.2	16.9	46.9	11.7	z=3.56 p<0.001
Tapping	sum of points tapped with both hands	70.5	10.5	74.4	7.4	z=1.96 p=0.05
Bălăceanu	time in seconds	82.6	17.2	73.3	11.9	z=2.87 p<0.01
	imprecisions number	8.2	6.8	3.0	7.2	z=2.86 p<0.01
Bourdon-Amfimov	sum of correct barraged letters	450.3	97.3	471.8	84.8	NS
Praga	solved points	67.2		75.4		z=2,59 p<0.01
I_2	solved items to:					
	- antonyms	9.3	1.2	9.7	0.5	NS
	- analogies	7.8	1.5	8.4	2.2	NS
	- number series	7.9	2.1	8.9	1.2	z=2.44 p<0.05
	- useless words	7.9	2.1	8.8	1.2	z=2.46 p<0.05
	- total score	32.1	6.5	36.0	4.7	z=3.15 p<0.01
	- time of solve (min)	13.5	2.3	9.2	3.1	z=6.9 p<0.001
Subjective symptoms Quest.	percentage of:					
	- lability	38.8		43.7		NS
	- tiredness	20.4		18.7		NS
	- extroversion	36.7		18.7		$chi^2 = 4.35$ p<0.05
	- neurosis	28.6		31.1		NS
Anxiety Ouest	percentage of:					
	- normal subjects	47.9		73.5		chi square =
	- anxious subjects	52.1		26.5		=5.3 p<0.05
	- subjects able for stress and crises	5.9		32.3		z= 3.04 p<0.01
	- subjects with increased anxiety	19.6		5.9		z=1.97 p<0.05
	- subjects with normal C (ego strength)	80.4		94.1		z=1.96 p=0.05
	- subjects with normal O (paranoiac defence)	70.6		88.2		z=2.05 p<0.05
	- Ss with anxious/ inhibited ego	9.8		0		z=2.28 p<0.05

DISCUSSIONS

As we see, the exposed subjects have a significant increase of RT, reduced motor reactivity, reduced efficiency in tasks which involve distributive attention, manual accuracy, ability to reason with number series and categories of words. The increased time of solve the items of Intelligence Test put in evidence a reduced efficiency in cognitive tasks. The questionnaire of subjective symptoms points out the increase of extroversion symptoms, which correlated with the preference to participate to passional discussions (46.7% in exposed group vs 11.8% in control group, $z=3.26$, $p<0.001$), can indicate the increase of excitatory state which can subsequently induce inhibition and tiredness. This conclusion is supported by the increased percentage of subjects which relate frequent periods of strong tiredness (66.9% exposed subjects vs 47.1% control subjects, chi-square = 6.09 $p<0.05$) and on the other hand, as we have had seen, the reduced psychomotor reactivity and functional capacity of distributive attention. They have also more frequent troubles of recent memory (they forget what they recently think, 66.7% from exposed subjects vs 41.2% control subjects, chi-square = 7.7 $p<0.01$). The reduced percentage from exposed subjects which relate that they not dream (46.7% exposed subjects vs 75.5% control subjects, $z=2.65$, $p<0.01$) can also support the idea of increased tiredness and/or the trouble of recovery sleep. It was observed also the possibility to install an inferiority complex (55.6% exposed subjects vs 32.4% control subjects, $z=2.04$, $p<0.05$, relate that they have some difficulties to discuss about them with others).

The questionnaire of anxiety put in evidence increased percentage of anxious subjects in exposed group, more frequent increased anxiety, weakness of Ego, paranoiac defence (as compensatory factor), inhibited Ego, which can reduce the ability to cope with stress and crises and can develop neurosis or angst hysteria.

CONCLUSIONS

These findings support the idea of unfavourable influence of chronic occupational exposure to pulsed microwaves on psychological reactivity, reducing the individual resistance to stressful events, increasing the level of tiredness, reaction time, troubles of recent memory, inducing an excitatory state that subsequently can induce inhibition and tiredness. These psychological changes can have some repercussions on individual organic resistance to physical and/or psychosocial potential harmful factors on the one hand and/or on worsening of other psychoorganic affections, on the other hand.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
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Abstract No. 90725
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Mini-Presentation

PAPER TITLE Clinical and Epidemiological Studies of Biological Effects of SHF-radiation

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MAJOR SCIENTIFIC TOPIC NUMBER 2.2 (see page 7)

ABSTRACT (See instructions overleaf)

Clinical and epidemiological studies of health of 80 people including the basic group of 30 people living for 5 years in the zone of daily SHF-exposure from 3 radars of 200, 1000, and 3000 MHz were carried out. The power density of SHF-radiation at the exposured sites ranged from 0,07 to 0,2 mW/cm2.

6 people (20%) of the basic group showed the complex of symptoms which are tipical for response on chronic SHF-irradiation: astheno-neurotic syndrome of various degree; disoders of the vegetative nervous system; neurogenic hyperventilation syndrom; thrombocytopenia and coagulopathy syndrome; myocardium function impairmehts: automatism (mainly, bradyarythmia), conductivity and excitability; changes in a number of biochemical blood characters and in T- and B-immune systems.

All the mentioned above allowed to conclude that the exposed people suffer from a chronic disease induced by radio-waves.

NATIONAL AND INTERNATIONAL STANDARDS ON THE MEASUREMENT OF ELECTROMAGNETIC FIELDS AND STATE-OF-THE-ART MEASUREMENT INSTRUMENTS

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INTRODUCTION

Exposure levels for non-ionizing radiation have been introduced world-wide on a technical basis. An ever growing number of countries tend to fix exposure levels on a political basis. These activities require measurement standards for both low-frequency and high-frequency measurement procedures and instruments. This standardization process has been started on different national bases but is not yet finished at an international level. I.e. in the low-frequency range, there are as yet no standardized measurement techniques in existence around the world. The result is that a large number of low-frequency devices are available commercially, but results from the different devices are not comparable. Moreover, there is no other branch of measurement technology where, in specifying technical data, so much information is either concealed or misrepresented. This is hardly acceptable in a field with major safety implications, as is the case with measurement of electromagnetic fields and waves.

GERMANY: VDE

When it stipulated its limits, the VDE also defined the necessary measurement techniques in DIN VDE 0848 (1). The VDE is thus an exception within Europe. These measurement techniques described in this standard are particularly valuable for the high-frequency range. However, precise specifications are lacking, along with requirements for test equipment. The standard leaves the choice of measurement technique to the person making the measurement, even though all errors and the influence of the frequency spectrum must be taken into account when determining the field strength values. The limit between high frequency and low frequency was set at 30kHz. For the low-frequency range, the norm is restricted to rudimentary guidelines for designing the test setup. Important is the restriction for the magnetic field probe, which is required to be a induction coil with an area of $0{,}01 m^2$.

EUROPE: EBU

At the European level, the EBU (European Broadcasting Union) has standardized measurement techniques, but they are tailored very specifically to the needs of the broadcasting industry (5). The effects of the various modulation techniques on the measurement results are documented in great detail.

INTERNATIONAL: IEC

In the eyes of the experts, the current situation is unacceptable: Limits have been stipulated, yet the measurement problem is left to the whims of the free market. The measurement techniques valid in Europe, particularly for verifying the limits of the two CENELEC norms (3, 4), were delegated to the IEC (International Electrotechnical Commission). IEC created two working groups with the aim of defining the test and measurement standards immediately under tight time constraints. The voting drafts of these papers are expected to circulate at the end of 1996. Both standards are focused to measurement procedures and measurement equipment with specific regard to human exposure. For the frequency range 100 kHz to 1 GHz, there is a draft dealing with radio communications (10). However, this standard has only limited use in industrial applications.

AUSTRALIA

In Australia, a newly revised version of the standard for measurement of high-frequency fields is now available in draft form (9). The original versions were highly practical documents, and the new standard is a continuation of this trend towards detailed description of test and measurement techniques. One new and innovative feature is the long-awaited description of a spatial averaging technique for avoiding excessive influence of RF hot spots and better recording of the whole-body exposure. Measurements are required at five points of a square (corners and center) having an edge-length of 300mm; the average of the five measured values is then displayed.

SWITZERLAND

The Swiss Federal Bureau for the Environment, Forests and Landscape (Bundesamt für Umwelt, Wald und Landschaft) has issued a recommendation with limits based on the IRPA values, including a detailed study of measurement of high-frequency fields (2). Here, the focus areas are error estimation in measurement and calibration, and investigation of measurement uncertainties. Radar signals are given special treatment. For low-frequency field measurements, the corresponding recommendation is not yet available.

USA

In the USA, the third revised version of the standard on measurement of high-frequency fields is now available (8). The fourth version is expected at the end of 1996. This standard will be used around the world as part of the standardization process at the national level, even if it is modified in each case. There is detailed coverage of three main areas: general test-related issues, instrumentation and actual measurements. In the fourth version, the section on instrumentation will most likely be revised since major technological advances have been made since the last version was published in 1991. A corresponding standard on the measurement of low-frequency electrical and magnetic fields is in preparation (7).

CONSEQUENCES FOR THE MEASURING DEVICES AND MEASUREMENT PROCEDURES.

It is a basic fact that the complexity of determining limits entails unusual problems for test equipment as well as for test personnel. One the one hand, the instruments must be simple to operate. However, the results must be precise, traceable and reproducible. Except for highly trained personnel in scientific and technical fields, the person making the measurements is generally uninformed about the theoretical and practical aspects of electromagnetic fields. In recent years, this problem has become more acute as the national and international limits recommendations have made their way into industry. This also explains the introduction of innovative measurement techniques for high-frequency as well as low-frequency applications (6). Due to space constraints, all of the requirements for field measuring devices cannot be considered here; this is a task for the IEC working groups. However, we will look at some of the most innovative advances in measurement technology over the past five years:

LOW FREQUENCY, ELF

3D-E-field measurement: When measuring electrical fields under high-voltage lines, the older one-dimensional technique was sufficient since it was possible to compute the field pattern in advance. However, the current standards of the VDE and Cenelec require measurement of the resultant field strength, regardless of application. Especially in industrial environments (switching fields, transformer stations, induction heating devices), the field pattern is a function of the installation and also varies over time, meaning that three-dimensional measurements are required. A setup of a state-of-the-art measurement device is shown in Figure 1 (12).

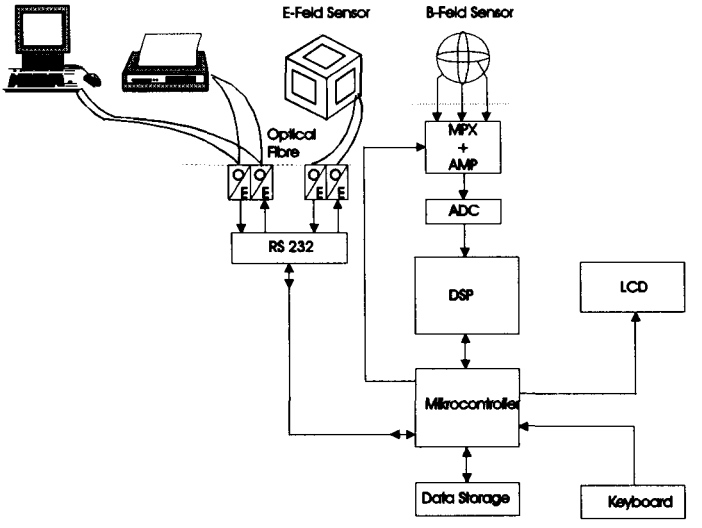

Figure 1 Setup of the EFA-3 measurement system for isotropic B- and E-field measurements

Tuneable filters: E-field and H-field measuring device must also allow selective recording of the field components since the international and national limits recommendations are all highly frequency dependent. Broadband equipment cannot handle this measurement when the frequency of the emitter is unknown and the existence of multiple signals cannot be excluded as a possibility. One example is an induction heating system, which simultaneously radiates 50/60 Hz fields with all their harmonics as well as heating frequencies in the 25 kHz range. Newer measuring devices are equipped with filter functions for handling such situations.

RF & MICROWAVE

Wide frequency ranges: Extremely broadband probes, primarily for applications in ISM areas and in telecommunications (AM & FM radio, GSM, TV) have been developed with frequency ranges from 100 kHz to 3 GHz and a negligible frequency response. Figure 2 shows the typical frequency response curve.

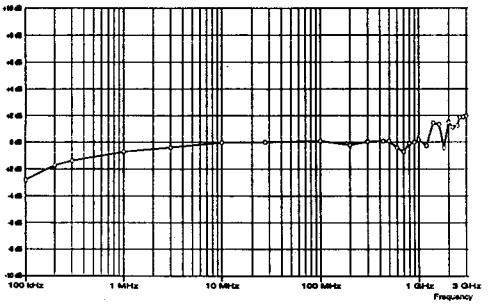

Figure 2 Typical Frequency Response of EMR-30 100 kHz to 3 GHz

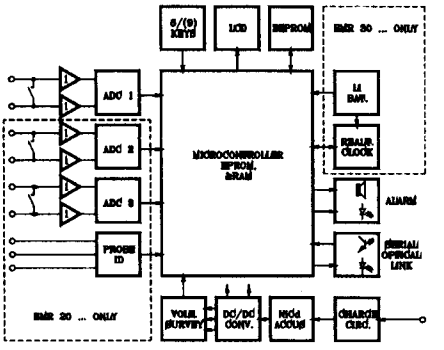

Figure 3 Triaxial digital axis processing for RF & Microwave measurements

Triaxial evaluation: Today, using trichannel digital evaluation of the individual axes of an isotropic sensor, it is easy to cover a dynamic range of 60 dB with a single probe (11). All of the limits stipulated in national and international standards can be covered without resorting to exchangeable probes with their associated calibration problems. Figure 3 illustrates the modular design of a triaxial digital system.

No auto-zero: Elimination of the need for an "auto-zero" function is another advance. Due to their extremely high-impedance input stages, field meters require continuing offset alignment. In the past, it was necessary to locate a field-free space for this procedure, thereby providing a typical trap for untrained personnel. Newer devices automatically perform this alignment even under field exposure.

CONCLUSION

This paper describes the international activities and policies for standardizing the measurement equipment for electromagnetic fields. Activities of the various countries, mainly Germany (VDE), USA (IEEE), Europe (IEC) and Australia (AS/NZS) as well as others are covered. Based on these reports requirements for measurement equipment are discussed and new designs for instruments especially developed for the measurement of electromagnetic fields with particular regard to human exposure are presented.

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LASER: EXPERIMENTAL DETERMINATION OF RETINAL DAMAGES THRESHOLDS INDUCED BY MULTIPLE PICOSECOND PULSES.

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INTRODUCTION

Most laser safety documents advise the laser user that caution must be used in the evaluation of exposure to repetitively pulsed radiation since they are only limited data on multiple pulse exposure criteria. The empirical multiple pulse formula is based on some data indicating that there is generally a cumulative effect in multiple-pulse exposures. This effect is a reduction in the threshold energy per pulse relative to the single pulse threshold. The best fit to experimental data predicts that the threshold energy per pulse decreases in proportion to the fourth root of the number of pulses *n*. This model has been reported to be in agreement for exposures ranging from picoseconds to seconds. However, the very limited data reported in the sub-nanosecond domain of time were obtained with a low pulse ratio frequency. No data exists in the litterature concerning the effect on the retina of ultrashort pulses delivered with a high repetition rate. Also, it has been hypothesized that one cannot extrapolate from longer pulses width because the ultrashort pulse (single or multiple) induce effects which may involve fundamentally different mechanisms of damage. Moreover, laser exposure limits for a single-pulse duration less than 1 ns cannot be specifically provided by ICNIRP Guidelines or other publications on laser radiation because of a lack of biological data (1). The limit values for these exposure times has been derived by maintening the irradiances applicable to nanosecond pulses. With the goal in mind to obtain adequate data base for single and multiple ultrashort pulses, an experimental study has been directed toward determining the retinal damage thresholds induced by picosecond pulses emited in the visible spectrum.

METHODS

The experimental system is a dye laser pumped by a mode locked pulse compressed, frequency doubled Spectra-Physic Nd:YAG laser. A cavity dumper produces single pulses with an adjustable pulse repetition rate between single pulse and 1 million pulses per second (1 MHz). The wavelength at the output of the dye laser is 590 nm. A sample of the laser beam at 590 nm was deflected onto an autocorrelator which measures the pulsewidth and onto a photodiode and a calorimeter. The output signal of the photodiode, calibrated by the calorimeter, was converted with 8 bits resolution by a 100 MHz analogic-digital converter and was transferred to a computer. The animal used in the experiments was the rabbit Fauve de Bourgogne. Rabbits were used primarily because the average ocular transmission and absorption characteristics are very similar to the human ocular media for the visible spectrum and secondly to compare with the other experiments (2). The theorically expected spot size formed on the rabbit retina with the dye laser optical system was about 30 μm. To aid alignment of the animal's eye, a 1 mW HeNe laser was mounted co-axially with the main laser

beam path. Rabbits were anesthetized with intramuscular injection of ketamine hydrochloride (10 mg kg^{-1}), placed on a stand permitting us to produce in most cases retinal lesions aligned on the visual streak. The pupils were dilated before to exposure, and the eyelids were held open with a wire speculum. The treatment and procedures used in this study conformed to the European Community Guidelines on Animal Experiments. Two investigative methods were used. A direct ophthalmoscopic examination was made 15 mn and 24 h after the exposure using a fundus camera. Any presence or suspicion of lesion was photographed. Immediately after the direct ophthalmoscopy, all animals were examined using fluorescein angiography. Photographic recordings were made for each observation of fluorescein leakage.

RESULTS

The results were obtained for repetitive pulse trains varying from 1 000 to 200 000 pulses. The pulse repetition frequency (PRF) used were 10 kHz, 100 kHz and 1 MHz. Because the low energy characterizing a single pulse in our experimental laser configuration, no retinal injury has been observed for 1 MHz pulse train including less than 100 pulses. Similarly, few damages have been detected with lower PRF than 1 MHz and the data were not enough to calculate a probit regression line. In our experimental conditions, the lowest fundoscopic threshold levels were obtained 24 h after exposure with the direct ophthalmoscopic method; whereas using fluorescein angiography, the best time to detect an injury was immediately after the exposure. The experimental results are in good agreement with earlier works (3-4). After employing both investigative methods, we founds that fluorescein angiography appeared to be the most sensitive technique. These observations show that a fluid leakage is involved in the damage process. For each experimental condition, data were scored and processed by a method of probit analysis (5). The median effective dose (ED_{50}^{MP}) which is the energy expected to produce a retinal damage in half of the exposures is given in Table 1.

Number of pulses *n*		**100**	**1 000**	**10 000**	**20 000**	**100 000**	**200 000**
Time period (interval) (s)		1.10^{-6}	1.10^{-6}	1.10^{-6}	1.10^{-6}	1.10^{-6}	1.10^{-6}
ED_{50}^{MP}	(μJ)	6 *	28	217	-	1 570	2 950
ED_{07}^{MP}	(μJ)	4 obs	21	133	-	677	1 483
Time period (interval) (s)		1.10^{-5}	1.10^{-5}	1.10^{-5}	1.10^{-5}	1.10^{-5}	1.10^{-5}
ED_{50}^{MP}	(μJ)				1100 *		
ED_{07}^{MP}	(μJ)	-	-	-	545 obs	-	-
Time period (interval) (s)		1.10^{-4}	1.10^{-4}	1.10^{-4}	1.10^{-4}	1.10^{-4}	1.10^{-4}
ED_{50}^{MP}	(μJ)		36 *				
ED_{04}^{MP}	(μJ)	-	25 obs	-	-	-	-

obs observed value
* extrapolated value

Table I. Median effective doses calculated from fluorescein angiographic data obtained on rabbit retinas with multiple pulses trains (pulsewidth 8.10^{-12} s, $\lambda = 590$ nm).

DISCUSSION-CONCLUSION

The results show a slight additivity for pulses separated by less than one microsecond. The injury threshold per pulse decreases when the number of pulse *n* included in the train increases. The best equation fitting the relation between the energy threshold of the pulsed train and the number of pulse *n* is given by:

$$ED_{50}^{MP} = 0.01n + ED_{50}^{SP}$$

where ED_{50}^{MP} is the threshold value for a multiple pulses train of n pulses corresponding to a probability of 50% and ED_{50}^{SP} the threshold value for a single pulse with the same probability. The ED_{50}^{SP} extrapolated from our results, in the range of 1 000 to 200 000 pulses, is 55 μJ and the correlation coefficient r^2 is 0.9995. This energy threshold is higher than other values reported in the litterature (6-7) but the threshold value of 6 μJ reported by Birngruber et al (6) for Chinchilla Grey rabbits give the same fitting equation with our results and the correlation coefficient is 0.9984.

The severity of the damage does not seem to increase with the number of pulses. The damage observed was very similar after an irradiation of 100 or 200 000 pulses. No hemorrhagic lesion has been induced by a train of 200 000 pulses. Such lesions have been histologically observed at different delay post exposure (8).

Number of pulses n	**100**	**1 000**	**10 000**	**100 000**	**200 000**
Limit value ($n^{-1/4}$ formula) (μJ)	$4.8\ 10^{-10}$	$2.7\ 10^{-10}$	$1.5\ 10^{-10}$	$8.5\ 10^{-11}$	$7.1\ 10^{-11}$
ED_{50} per pulse of the train (μJ)	$6\ 10^{-8}$	$2.8\ 10^{-8}$	$2.2\ 10^{-8}$	$1.6\ 10^{-8}$	$1.5\ 10^{-8}$

Table II. Comparison between the results (ED_{50}^{MP}/n) and the corresponding limit values.

The comparaison of the data with the limit values of the guidelines (Table II) show that the empirical multiple-pulse formula, predicting that the threshold energy per pulse decreases in proportion to the fourth root of the number of pulses n, seems acceptable for picosecond pulses delivered with a high repetition rate. The safety marge is reasonable but is also variable depending the number of pulse. The empirical $n^{-1/4}$ formula is acceptable but does not describe specifically the mechanim of the repetitive-pulse injury thresholds.

Acknowledgments-This study was supported by contract 94/001 from the Direction des Recherches Etudes et Techniques du Ministère de la Défense.

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PROTECTION FROM SOLAR ULTRAVIOLET RADIATION IN HONG KONG

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ABSTRACT

Increasing skin cancer rates in many countries have been attributed to increasing solar UV radiation due to global depletion of stratospheric ozone. The majority of the existing solar UV radiation monitoring stations are found in high latitude countries nearer to the 2 poles. We are building one such station in Hong Kong at latitude 22.5 °N. The station can provide information on solar UV irradiance, exposure limit, irradiance forecast and stratospheric ozone depletion in this region of the world.

Protection factors of fabrics and clothing for protecting against solar UV radiation have been determined and analyzed. The spectral transmittances of the fabrics in the wavelength range 250 to 400 nm were measured by means of a solar simulator and a double monochromator spectroradiometer. Effect of washings on the protection factors were simulated and determined.

INTRODUCTION

Depletion of stratospheric ozone by chlorofluorocarbons (CFC) was predicted more than 20 years ago and was first discovered in the late 1970s. Since then, ozone reductions were reported in many countries and the largest and deepest hole on record was observed in 1992 (1). It is believed that CFC will continue to increase in the stratosphere leading to more frequent and profound occurrence of the ozone holes (2). Many countries have set up networks of solar UV monitoring stations. But so far, the majority of them are located in high latitude countries nearer to the 2 poles. Little data are obtained near the tropic in South-east Asia. Hong Kong at 22.5 °N will provide data on solar UV exposure to one of the most densely populated cities in the world.

The solar UV emitted from the Sun is divided into 3 regions: the UVA (315 - 400 nm); UVB (280 - 315 nm) and UVC (100 - 280 nm). However, as a result of absorption by ozone in the atmosphere, no UVC and only about half of the UVB can reach the surface of the Earth. The irradiance is dependent on latitude and time of the day and year as well as on atmospheric conditions, such as cloud cover and atmospheric ozone content, that affect penetration of the radiations.

METHODS

1. Solar UV radiation

The biological response of the skin to ultraviolet radiation is very wavelength dependent. Various action spectra are proposed (3), (4) and they all show that UVB radiation is far more effective at producing erythema than UVA radiation. Between 300 nm and 315 nm, the response drops by three orders of magnitude. This drastic change in response requires a very precise spectral measurement of the solar spectrum or large error in the calculated biological response will result.

The station is located on the roof top of one of the buildings in The University of Hong Kong and is about 350 m above sea level. The site is away from the urban area and also far away from exhaust chimneys of other buildings in the University. The solar spectral irradiance is measured by means of a spectroradiometer (Model IL790, International Light Inc.) that uses a wide angle diffuser to provide a near cosine acceptance angle. The double monochromator contains two gratings blazed at 240 nm so that deep UV can be measured with a stray light rejection ratio of 10^6 to 1. The spectroradiometer is not used for continuous spectral irradiance measurement because of the complexity involved compared to other simple detectors. Instead, two UV-biometers from Solar Light Co. (Models 501 UVA & 501 UVB) are used to record the UV radiation 24 hours a day. One of the meters gives UVA irradiance in terms of mW cm^{-2} and the other gives UVB effective power in terms of minimal erythemal dose (MED) per unit time.

Atmospheric data including cloud cover, visibility and duration of bright sunshine are obtainable from the Royal Observatory of Hong Kong.

2. Protection factors for clothing

The protection factor (PF) is given by the ratio of effective UV radiation dose for unprotected skin to that of skin protected by the clothing (5), while both the UV spectral irradiance and the biological action spectrum are taken into account. Mathematically, PF is given by

$$PF = \frac{\sum_{250}^{400} E_\lambda S_\lambda \Delta_\lambda}{\sum_{250}^{400} E_\lambda S_\lambda T_\lambda \Delta_\lambda}$$

where E_λ = action spectrum given by IRPA (3)
S_λ = solar UV spectral irradiance in W m^{-2} nm^{-1}
T_λ = spectral transmittance of the clothing
Δ_λ = bandwidth in nm; and
λ = wavelength in nm.

A 1000 W solar simulator (Model 81191, Oriel Corp.) was used as the source and the spectral irradiance was measured by the spectroradiometer described earlier. The diffuser at the input of the monochromator collected both directly transmitted and scattered light from the clothing. The spectral transmittance was determined by noting the reduction in UV intensity when the clothing was placed in front of the monochromator.

RESULTS AND DISCUSSIONS

1. Solar UV radiation exposure

The two UV-biometers provide data every 15 minutes and the data are stored in a data-logger which can be transferred to a computer at any time. Daily variations of UVB MED and UVA irradiance are shown in Fig. 1. During a clear sky day, the change in UV intensity is fairly gradual, but in cloudy days, the intensity can change abruptly depending on cloud cover and atmospheric conditions. Basically, the UVB MED and UVA irradiance have similar variation throughout the day. At large zenith angles, it is noted that UVB radiation is absorbed more rapidly by the atmosphere than UVA radiation.

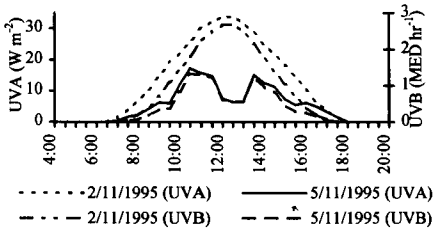

Fig. 1 Variation in measured UVA & UVB

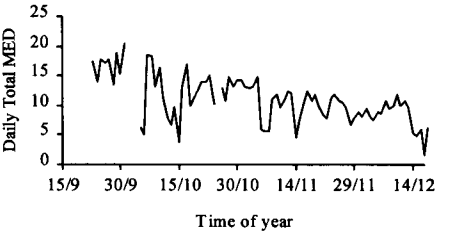

Fig. 2 Variation of the daily total MED

Since the UV-biometers were installed not long ago, measurement data before September 1995 were not available. Fig. 2 shows the total daily MED from September 1995 to December 1995. It ranges from a minimum of about 1 MED in December to about 20 MEDs in hotter months. It is expected that above 25 MEDs could be reached in June or July. The monthly average daily total MED is calculated for the few months in Table 1. The table also lists the duration of exposure to exceed 1 MED at solar noon under clear sky conditions. Since the limit could be exceeded easily around noon time, it is important that outdoor workers be protected with hats, sunscreens, sunglasses and clothing, particularly during Summer.

Table 1. Monthly average daily total MED and the duration of exposure for exceeding 1 MED at solar noon under clear sky conditions

Month	Average daily total MED	Time to exceed limit (min)
September	15.3	13.9
October	12.4	15.3
November	10.0	19.5
December	8.1	24.6

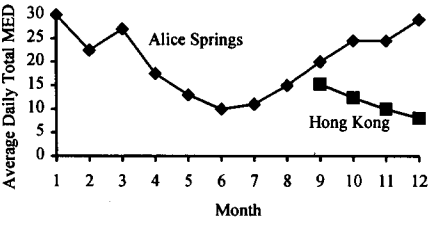

Fig. 3 Comparison of daily total MED with that in Alice Springs, Australia

Countries that have the same latitude should have comparable solar UV intensity in clear sky days. But countries that have equal latitude but in opposite hemisphere of the Earth will experience different solar UV intensity even in the same season because of the different relative position of the Earth and Sun. Fig. 3 shows the total daily MED measured in Alice Springs, Australia (23.5 °S) compared to Hong Kong (22.5 °N).

2. Protection factors for clothing

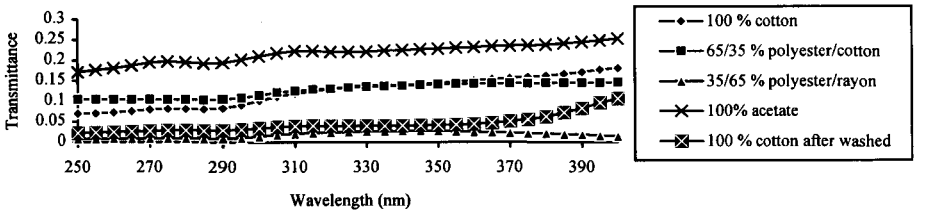

Fig. 4 Spectral transmittances of different types of fabrics

Samples of cloth were obtained from manufacturers and only thin and light samples were measured as the thicker ones would no doubt, have very high PFs. In general, the PF value depends on a number of factors such as weave, color, weight, stretch and water content (5). Typical spectral transmittances of some fabrics are shown in Fig. 4. The cloth samples were further tested for stability of the PF values by washing the cloths for a number of times. The cloths were stirred in a beaker of water with laundry powder for 15 minutes to simulate the action of a washing machine. They were then dried under an infra-red lamp to complete one washing cycle. 0, 1, 3, 7 and 13 washing cycles were repeated. Out of the 35 samples that have PFs less than 50, 71 % have PF increased, 6 % decreased and 23 % fluctuated or remained more or less the same with increased number of washes. The increase in PF is mainly due to shrinkage of the fabrics after the washes leading to a tighter weave of the fabrics. The largest change in PF was observed in a 100 % cotton fabric which increased by 3.2 times. The fabrics that have smaller PFs after the washings were not special ones, and the decrease was not more than 10 %, so it could be concluded that any decrease in PFs were mainly due to experimental errors.

Table 2. Protection factors of some combined fabrics

Sample no.	PF1	Composition 1	PF2	Composition 2	PF1 x PF2	Measured PF
1	8.5	100 % acetate	4.6	100 % acetate	39	30
2	4.6	100 % acetate	5.6	65/35 % polyester/cotton	26	26
3	8.9	100 % cotton	7.9	100 % cotton	70	36
4	4.6	100 % acetate	8.6	65/35 % polyester/cotton	40	35
5	4.6	100 % acetate	8.9	100 % cotton	41	30

Table 2 shows the combined protection factor of fabrics by putting two samples of cloth together and then measuring the combined spectral transmittance. Because PF is dependent on the spectral irradiance of UV radiation reaching the surface of the fabrics, and the second fabric underneath the surface one would receive a spectral irradiance quite different from the one incident on the surface fabric, therefore the combined PF of the two fabrics will not be equal to the product of the two individual PFs. A second more important factor that affects PF is that the combined transmittances for some types of fabrics differ quite significantly from the predicted transmittance as shown in Fig. 5. Though the measured PF value might be much lower than predicted, the actual difference in protection power is insignificant.

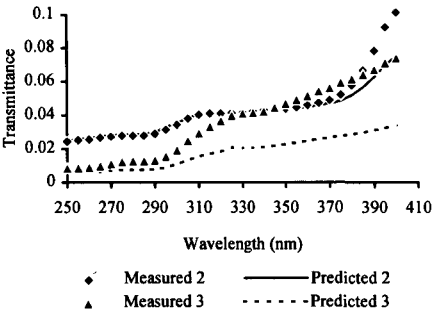

Fig. 5 Transmittance of combined fabrics (refer sample no. to Table 2)

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UV-RADIATION INDUCED UDS IN HUMAN LYMPHOCYTES EXPOSED *IN VITRO* TO LOW LEVELS OF IONISING RADIATION

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INTRODUCTION

It has been reported that lymphocytes of persons exposed to low doses of ionising radiation exhibit enhanced rate of UV radiation induced unscheduled DNA synthesis (1,2). Similar effects were observed in spleen cells of mice exposed to single or continuous exposures to low levels of radiation (3). This phenomenon has been attributed to radio-adaptive response (4). In order to eliminate the role of cell repopulating events and extra cellular factors it is necessary to check if the response exists when the priming dose is delivered *in vitro*. However, such an attempt was not envisaged due to the technical difficulties that will be encountered in delivering a chronic low dose irradiation to cells suspended *in vitro*. An attempt was therefore made to study the phenomenon by delivering a single low dose exposure to whole blood *in vitro*.

MATERIALS AND METHODS

a. Donors: Heparinised peripheral blood samples from 20 healthy non-smoking volunteers, aged between 25 and 45, who were employees of this centre and not engaged in radiation work, were collected in aseptic conditions at the Township Hospital and immediately transported to the laboratory. Blood obtained from each person was divided into three portions A, B and C (6ml each).

b. Irradiation: Portion C was exposed to a dose of 3 mGy with gamma radiation using a 185 kBq ^{60}Co source at a dose-rate of 12 mGy/hr at 37°C. During this period, the other two portions (A and B) were also maintained at the same temperature free from any radiation. Following this treatment, both irradiated and unirradiated portions were incubated at 37°C for three hours.

c. Lymphocyte separation: Lymphocytes were separated from all the portions using Histopaque (Sigma) and were washed three times in phosphate buffered saline and suspended at a concentration of about 3 x 10^6 cells per ml in PBS.

d. UV exposure: Lymphocytes obtained from portions B and C were placed in open petri dishes maintained at 4°C and exposed in dark to 30 Joules of 254 nm UV radiation emanating from a mercury lamp (Jarrel Ash) at a fluence rate of 3 J/m^{2}s.

e. Cell viability: The lymphocytes obtained were tested for viability by the trypan blue dye exclusion test immediately after separation and also after UV exposure.

f. Incubation: Immediately after UV exposure lymphocytes from all the three portions were suspended in RPMI-1640 (Sigma) culture medium supplemented with 20% autologous plasma, 5mM hydroxyurea (Aldrich), 1.85 MBq/ml tritiated thymidine (sp act 629 GBq/mmol BRIT,Bombay, India) and incubated for 90 minutes in complete darkness.

g. UDS assay : The incubation was stopped by the addition of cold thymidine and the lymphocytes were washed three times in PBS and finally suspended in 1 ml PBS. UDS was quantified by the method of Mitchell and Mirsalis (5). 0.8 ml was used for the estimation of DNA using diphenylamine reagent and the optical density at 601 nm was read spectrophotometrically (Hitachi, Japan). The minimum detection limit as per the study of Sandell (6) was found to be 4.6 μg DNA. The remaining 0.2ml aliquot was counted for its tritium activity in a liquid scintillation analyser (Packard, USA) having an efficiency of 68%. The Bq/μg DNA values were calculated and the results compared.

RESULTS AND DISCUSSIONS

The viability of lymphocytes measured immediately after separation and after UV exposure was >98 and >95% respectively. The DNA content in all test samples was greater than 8 μg. The Bq/μg DNA values obtained in these samples are presented in Table 1.

S.No	Sample	No. of Samples	Average Bq/μgDNA	P-Value (Student's t test)
1.	Control (A)	17	0.42 ± 0.11	
2.	UV exposed (B)	17	3.69 ± 2.03	< 0.001
3.	γ + UV exposed (C)	17	5.75 ± 2.23	< 0.01

Table 1. Incorporation of ^{3}H Thymidine in DNA of samples subjected to UDS

These results show that there is a significant increase ($p < 0.01$) in incorporated radioactivity in 17 out of the 20 samples that had received a prior 3 mGy exposure when compared to samples that had received only UV radiation exposure. The Bq/μg values for control samples ranged between 0.12 and 0.9 with an average of 0.42 ± 0.11. This activity is due to non-specific incorporation of tritiated thymidine. The values for UV exposed samples ranged between 1.4 and 7.76 with an average of 3.69 ± 2.03. This significant increase in activity is indicative of repair synthesis due to UDS. The samples exposed to gamma and UV radiation showed values ranging between 1.95 and 9.35 with an average of 5.75 ± 2.23. The significant increase in incorporated radioactivity due to the prior exposure of 3 mGy can be attributed to an adaptive response to ionising radiation. Similar results have been observed in cells exposed *in vivo* (5).

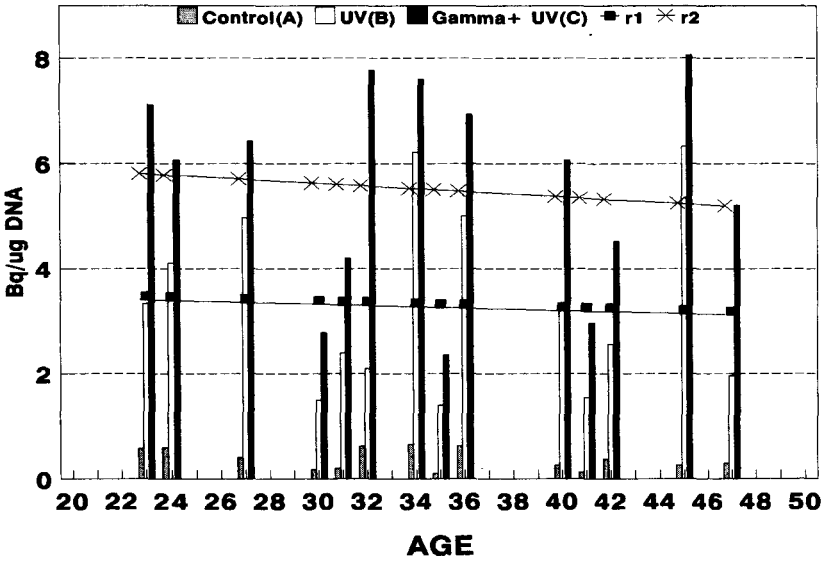

Fig. 1 UDS as a function of age

ıe UDS data as a function of age is shown as a bar chart in Fig 1. 14 out of the 17 were chosen since a few of the X values overlapped. (For example, the values l 9.35 do not appear in Fig.1). The response to UV radiation and that of γ+UV ns in these samples with respect to age did not show any correlation as shown by a ıon analysis. $r_1 = -0.0813$ for samples exposed to UV radiation and $r_2 = -0.099$ for s exposed to both gamma and UV radiation. The negative correlation suggests that UDS endency to decrease with age.

In the present study a time gap of about three hours between the priming and the nging dose was chosen since previous reports (7,8) suggest that such a time gap is ısary for the expression of adaptive response. Enhanced levels of poly (ADP) ribose nerase (9,10) and other inducible proteins produced during this time gap may have a ıounced influence in the expression of this response.

The variations observed in the Bq/μg DNA values among the 17 samples and the blunt ponse in 3 samples in this experiment could be attributed to the heterogeneity factors known to associated with adaptive response and to donor variations in their capacity to perform DNA pair (11,12).

ONCLUSIONS

The results obtained in this study indicate that

1. Enhanced rate of UV-radiation induced UDS can be evoked when a priming dose is delivered in-vitro.
2. A single low priming dose of 3 mGy delivered at a dose rate of 12mGy/hr is sufficient in inducing an enhancement in UDS.
3. A three hour interval between the priming and the challenge UV exposure seems sufficient to induce the phenomenon *in vitro*.
4. The significant enhancement in the rate of UDS in this study eliminates the ambiguity that the response may be dependent on extra cellular factors and cell repopulating events.

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EXPOSURE TO UVR: RISK QUANTIFICATION FOR NON-MELANOMA SKIN CANCER

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ABSTRACT

Solar UVR has an acknowledged role in both non-melanoma skin cancer (NMSC) and malignant melanoma (MM). Episodic exposure to UVR, sufficient to cause erythema, has been shown to be a major risk factor for MM. For NMSC and for skin damage the lifetime cumulative solar UVR exposure is of more importance. A software package has been developed to assist in estimating a person's lifetime exposure.

INTRODUCTION

Long-term UVR exposure of the skin is generally accepted as the most important cause of NMSC. Much of our knowledge on the influence of solar UVR is derived from epidemiological studies of skin cancer. Such studies are greatly strengthened when the actual exposure of the subjects can be quantified. This is difficult and most studies use surrogates such as latitude, sunlight hours, or ambient UVR measurements for personal exposure. Within a given population the actual exposure will depend on many factors including place of residence, occupation, recreational activities, annual vacation, outdoor behaviour and most importantly the personal protective measures practised (1). This information together with a global solar UVR database are used in a windows-based software program which has been developed to generate monthly UVR exposure data for different anatomical sites over a person's lifetime.

For a given genetic susceptibility, the two most important factors in determining the relative risk of NMSC from solar UVR exposure are age and environmental UVR exposure. In the program, *UVRISK*, thegenerated UVR exposure data, is used in a simple power law relationship to calculate the cumulative NMSC risk. The program is used as an aid in the determination of pensions for chronic skin damage sustained as a result of armed forces overseas service. The lifetime exposure and risk is given for several different scenarios involving change in place of residence, employment and outdoor behaviour. The benefits of behaviour modification are clearly demonstrated.

EXPERIMENTAL METHOD AND RESULTS

Personal solar UVR exposure

The mid-month ambient UVR for a given location has been calculated by use of a modified Bjorn (2) model. The computed spectral power distribution has been weighted by the CIE (3) reference action spectra for erythema to give the effective irradiance. The data are then expressed in terms of minimal erythemal doses (MEDs) where one MED is 200 J m^{-2} of effective irradiance (4). Data have been calculated in 5° bands from 75°S to 75°N. The effect of long-term average cloud cover has been included and the results verified for Australia. Anatomical distribution and personal protection data are from published studies. The exposure calculation is performed for anatomical sites including face(cheek), hand(dorsum), back (shoulder), arm(upper) and leg. The effective dose to the skin for the given month and body site according to the following:

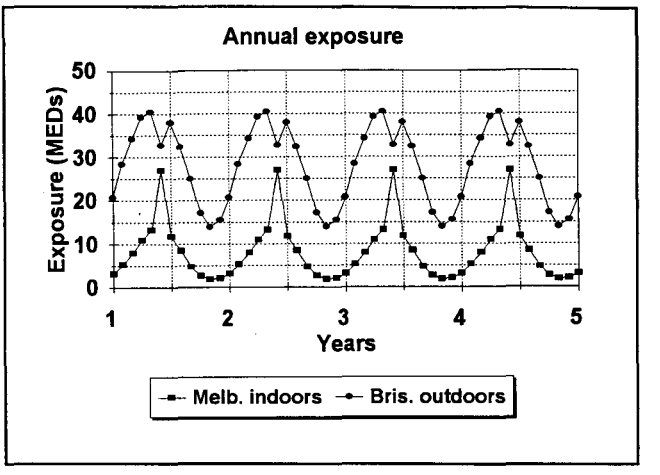
$$PAE(a,m) = \sum_{n=1}^{n=8} MAE\,(m,L_n) \cdot AF \cdot SF_n \cdot OF_n \cdot CF_n \cdot ESF_n \cdot W_n$$

where PAE(a,m), (personal ambient

exposure (in MEDs) for a specified anatomical site, a and month number, m); n (specified activity); m (month of the year); MAE(m,L_n),(daily ambient exposure for the middle of the month m in location L and activity n); AF, (body anatomical factor); SF_n, (site factor for activity n), OF_n (outdoor factor for activity n); CF_n(clothing factor for activity n); ESF_n(environment shade factor for activity n) and W_n(number of days in the month that the activity was performed). The cumulative personal ambient exposure for a specified anatomical site up to the evaluation age of T (months) i.e. $MED_{cum}(a,T)$ is determined by summing over T months.

Figure 1 shows the calculated annual exposure for two scenarios.

Figure 1. *Calculated annual exposure for Melbourne (37.8°S) indoor worker with 50% outdoor recreational exposure and for Brisbane (27.5°S) outdoor worker with similar recreational activities. At 40 years of age the Brisbane worker has had 3.4 times the effective UVR exposure of the Melbourne worker.*



The risk of NMSC from solar UVR exposure

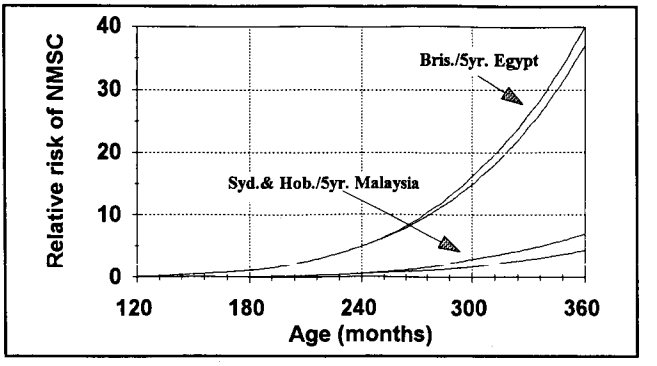
It has been shown that, for a group of subjects with a given genetic susceptibility, the two most important factors in determining the relative risk are age and environmental UVR exposure. A simple power law relationship { $Risk \propto (MED_{cum})^{\beta-1} (age)^{\alpha}$ } expresses the cumulative risk in terms of these factors (5) where α is a numerical constant associated with the age dependence of the cumulative incidence and β is a biological amplification factor. This relationship applies to the situation where the annual exposure received by an individual remains constant throughout life. Generally exposure does not remain constant - children often receive greater exposure than adults, adults can change their city of residence and their occupation. In order to take this into account this risk equation is modified (6) to estimate the risk of NMSC at age, T, for a given body site as:

$$Risk \propto [MED_{cum}(a)]^{\beta-1} \sum_{t=0}^{T} [MED_{ann}(a) \text{ at age } (T-t)]\ t^{\alpha-\beta}$$

where $MED_{cum}(a)$ is the cumulative effective UVR dose to the skin and $MED_{ann}(a)$ is the annual effective UVR dose to the skin, both being for a given body site. We assume (7) values of α=5 and β =2 as the result of combining the data for SCC and BCC and taking into account that BCCs are about four times more prevalent than SCCs.

Figure 2. shows the increased risk arising from 5 years of overseas service. For Egypt the increased risk of 8% arises from poor protection in a desert environment. For Malaysia (~4°S) the low latitude and poor protection accounts for an increased risk of 60%.

Figure 2. *Effect of overseas service on a veteran. Results show an increase in risk as a result of serving for 5 years in Malaysia or Egypt. Increase is due partly to the higher UVR environment but more importantly to the poor protection practices while serving.*



Protection and behaviour modification

Figure 3 shows that even for an outdoor worker large changes in both exposure dose and risk can be attained with the adoption of even modest behaviour and protection changes.

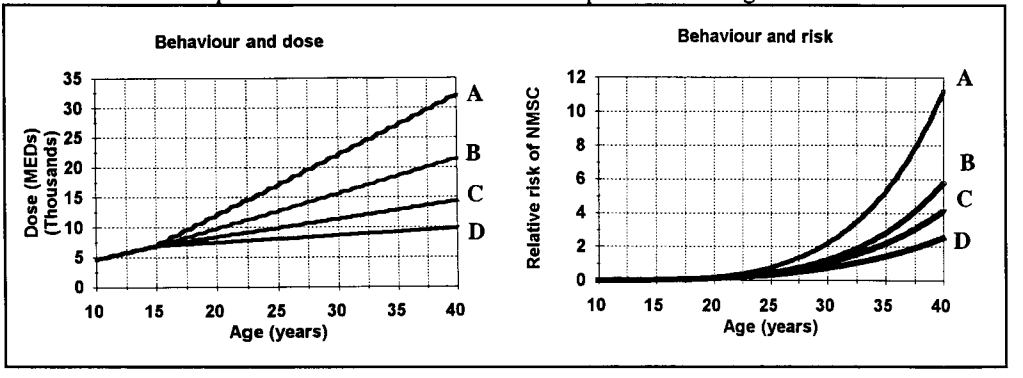


Figure 3. Effect of behaviour change on personal dose and risk of NMSC. A - 100% outdoor worker, unprotected during work and recreation; B - 50% outdoor worker, unprotected; C - as for B but SPF 5 protection during work; D - as for B but SPF 5 protection during work and recreation.

CONCLUSION

The program described has proven to be both user friendly and efficient in evaluating exposure doses and risk of NMSC. Changes in protection and behaviour are readily assessed. Further verification is planned.

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THE INFLUENCE OF DOSE DISTRIBUTIONS ON THE RESULT OF UV-BIODOSIMETRY

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INTRODUCTION

Disinfection of drinking water with ultraviolet radiation has become a common method in Austria and in many other countries. The water usually is disinfected in flow through systems with low pressure mercury lamps as UV source, which emit predominantly UV radiation with wavelength 253.7 nm. Because of varying residence times of microorganisms and the spacial distribution of fluence rate in the irradiation volume, caused by different distances from the radiation source, by absorption of radiation in the water and by reflexion at the walls of the reactor, the microorganisms passing through in a turbulent flow, receive different fluences. In Austria UV-disinfection plants for drinking water must deliver a minimal dose of 400 Jm^{-2} for radiation of wavelength 253.7 nm (1,2). The fulfillment of this demand is proved during type testing. As dosimeter suspensions of bacterial spores are used whose UV-susceptibility has to be measured in a laboratory irradiation device. The dose, determined in this way, is called Reduction Equivalent Dose (3).

THEORY

If UV-inactivation of microorganisms used as biodosimeter follow first order kinetics their survival rate depends on dose in the following way:

$$\frac{N}{N_0} = e^{-a \cdot D} = 10^{-h \cdot D} \quad (a > 0, h > 0)$$

(N: Number of microorganisms after irradiation, N_0: Number of microorganisms before irradiation, a: UV-susceptibility of microorganisms (base e), h: UV-susceptibility of microorganisms (base 10), D: dose). The Reduction Equivalent Dose (RED) then can be calculated as:

$$RED(f(D),a) = -\frac{1}{a} \cdot \ln\left(\frac{\int_0^\infty e^{-a \cdot D} \cdot f(D) \cdot dD}{\int_0^\infty f(D) \cdot dD}\right),$$

where f(D) is the probability density of the dose distribution. If f(D) is distributed symmetrically, then the expectation value of dose E(D) is equal to the mean value μ of the distribution: $E(D) = \mu$. The expectation value of survival $E(\frac{N}{N_0})$ is:

$$E(\frac{N}{N_0}) = \int_{\mu-k}^{\mu+k} e^{-a \cdot D} \cdot f(D) \cdot dD = e^{-a \cdot \mu} \cdot \int_{-k}^{k} e^{-a \cdot u} \cdot f(u+\mu) \cdot du$$

or, for ξ between 0 and a:

$$E(\frac{N}{N_0}) = e^{-a \cdot \mu} \cdot 2 \cdot \int_0^k \cosh(a \cdot u) \cdot f(u+\mu) \cdot du = e^{-a \cdot \mu} \cdot \cosh(a \cdot \xi)$$

Because of $\cosh \geq 1$ follows:

$$E(\tfrac{N}{N_0}) \geq e^{-a\cdot\mu} \quad \text{and} \quad \mu \geq \text{RED}.$$

If the probability distribution of UV-dose is symmetric, the RED measured with microorganisms with exponential survival rate in general is smaller than the arithmetic mean of the dose distribution.

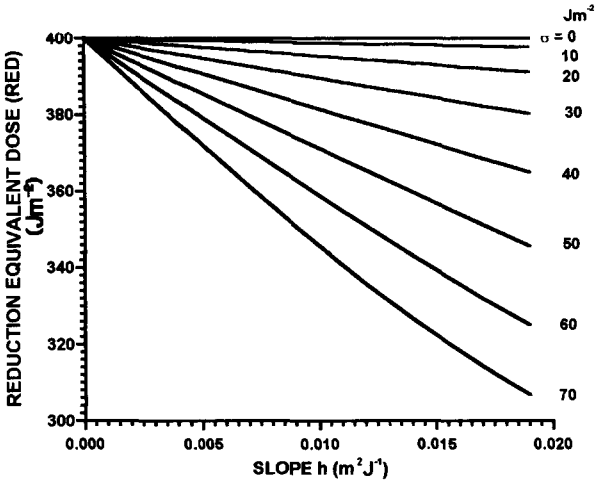

Figure 1. Dependence of RED from UV-susceptibility h of the biodosimeter (shoulder broadness = 200 Jm^{-2}) and from the standard deviation σ of the dose distribution (normal distribution, μ = 400 Jm^{-2}).

MODEL CALCULATIONS

The true shape of the dose distribution among the microorganisms passing a flow through UV-disinfection plant is unknown. Therefore we assumed the presence of normally distributed doses, from which follows:

$$\text{RED}_{\text{norm}}(h,\mu,\sigma) = -\frac{1}{h}\cdot\lg\left(\frac{\int_0^\infty e^{\frac{-h\cdot D}{\lg e}-\frac{(D-\mu)^2}{2\sigma^2}}\cdot dD}{\int_0^\infty e^{-\frac{(D-\mu)^2}{2\sigma^2}}\cdot dD}\right)$$

Integration was performed by numerical methods and in using survival curves with shoulder. We found a decrease of RED for incresing σ, that is for broader dose distributions and for more susceptible biodosimeters (increasing h). If the dose distribution is very narrow (σ very small) no difference exists between the Reduction Equivalent Dose and the arithmetic mean of of the dose distribution. These results are given in Fig. 1.

MODEL EXPERIMENT

In order to prove these theoretical considerations with an experiment we produced suspensions of spores of *Bacillus subtilis* ATCC 6633 with known dose distributions. The spores were cultivated with two different methods (4) resulting in different UV-susceptibilities

h of the microorganisms. The spores were irradiated in a laboratory irradiation device with doses from 200 to 600 Jm^{-2} and with an increment of 50 Jm^{-2}. Distinct volumina of these suspensions were mixed in order to get suspensions with known and symmetric dose distributions. The resulting discrete dose distribution is shown in Fig. 2.

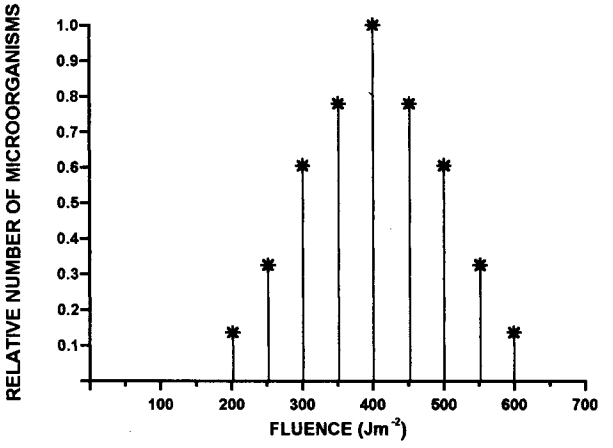

Figure 2. Experimentally produced discrete dose distribution with arithmetic mean $\mu = 400\ Jm^{-2}$.

The REDs following from the survival rate in the experiment were determined and compared with theoretical values. The results are given in Tab. 1. Within the accuracy of the method experimental and theoretical values showed good agreement.

slope h (m^2J^{-1})	μ (Jm^{-2})	RED (calculated) (Jm^{-2})	RED (experiment) (Jm^{-2})
0.0134	401	296	308
0.00687	402	339	337

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Comparison of hazard evaluation meters and spectroradiometers for the measurement of short wavelength light sources,

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ABSTRACT (See instructions overleaf)

Comparison of hazard evaluation meters and spectroradiometers for the measurement of short wavelength light sources

H. Brusl, N. Winker, K. Schulmeister, K. Duftschmid

In order to protect the health of workers in trade and industry, measurements of short wavelength light are required regularly. As the biological effect on the radiation is strongly wavelength dependent, spectroscopic measurements and subsequent multiplication with the respective action spectrum are performed to calculate effective irradiance. However the measurements are difficult to perform and measurement equipment is bulky and expensive. Instruments are available to measure effective irradiance directly, however in terms of dynamic range, influence of straylight and accuracy, the performance of these radiometers is poor compared to spectroscopic measurements. In order to characterise the usefulnes of hazard evaluation meters for workplace measurements, serveral typical UV and blue liegt sources found in industry have been measured with a double monochromator and the resulting effective irradiane values have been compared with valueus obtained with hazard evaluation meters. The comparison shows that great care has to be taken when performing short wavelength exposure measurements with hazard evaluation meters, as the measured effective irradiance values can differ substantially from the values as measured spectroscopically.

THE STUDY OF PHOTOTRANSFER THERMOLUMINESCENCE IN $CaSO_4$:Dy USING LASER AND ULTRAVIOLET NON-IONIZING RADIATION

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INTRODUCTION

Due to the growth of the use of ultraviolet (UV) and laser radiation in industry, medicine and university, the preoccupation with damages from these types of radiation has increased so much. In many establishments the control of health hazards of both laser and radioactive sources is made by the same group, and the possibility of using a TLD system to monitor laser radiation has been investigated (1). If thermoluminescent (TL) materials are preirradiated with known doses of ionizing radiations, the dosemeters are sensitive to laser radiation by phototransferred thermoluminescence (PTTL) and the laser dose can be measured (2). Like in many TL materials, $CaSO_4$:Dy irradiated with high dose of gamma radiation, annealed and exposed to UV and laser radiation, shows phototransferred thermoluminescence. The PTTL is a technique that was developed through the study of light effects in TL materials. The objective of this work is to study the PTTL in $CaSO_4$:Dy using UV and laser non-ionizing radiation The PTTL can be utilised for various dosimetric applications e. g., high dose dosimetry, dose reevaluation, UV and laser radiation dosimetry. The objective of this work is study different parameters of PTTL in $CaSO_4$:Dy using laser and UV radiation.

MATERIALS AND METHODS

Teflon pellets of $CaSO_4$:Dy produced at IPEN weighting 20 mg each,. with a 6.0 mm diameter and a thickness of 0.8 mm were used (3). The pellets were annealed at 300 C during 15 minutes before the gamma and laser or UV irradiations and thermally treated at 100 °C during 15 minutes after laser or UV irradiation.

A source of ^{60}Co with 15.0 TBq was used for sample irradiation. The samples were always irradiated under electronic equilibrium conditions, that is, the samples were placed between 3 mm thick Lucite plates.

The laser irradiation system contains a N_2 pulsed laser of 10 kW with pulse duration of 10 ns and wavelength of 337 nm. The UV irradiation system was composed by a Hg lamp Bausch & Lomb SP-200 and a Kratos GM-200 monochromator.

For the thermoluminescent measurement a Harshaw model 2000 AB TL reader was used. The linear heating rate was set at 10 C.s^{-1}, and the reading cycle was performed within 36 s, with a constant flux of N_2 of 4 l.min^{-1}. Light emission was integrated in the temperature interval between 200 and 360 C.

The pellets were first calibrated with ^{60}Co and selected at a range of 5 % of reproducibility.

Each reported value corresponds to the average of five measurements.

RESULTS

With respect to the laser radiation were studied two parameters:

- The PTTL response with a laser exposure time of 5 minutes as a function of gamma dose of ^{60}Co. It was measured between 10 Gy (10^3 rad) and 65 Gy (6.5×10^3 rad). The results are showed in Figure 1 and it is seen that the response is linear in this interval of measurements.

- The dependence of the PTTL response, using a gamma dose of 30 Gy (3×10^3 rad), with laser exposure time was studied from 5 to 20 minutes. It is seen in Figure 2 that the time of 15 minutes has the greater response and was found to reduce for the time of 20 minutes.

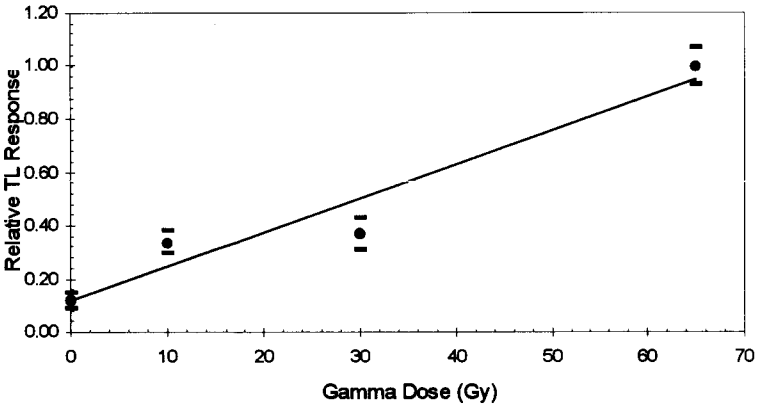

Figure 1. Dependence of TL Response as a Function of the Gamma Dose for $CaSO_4$:Dy

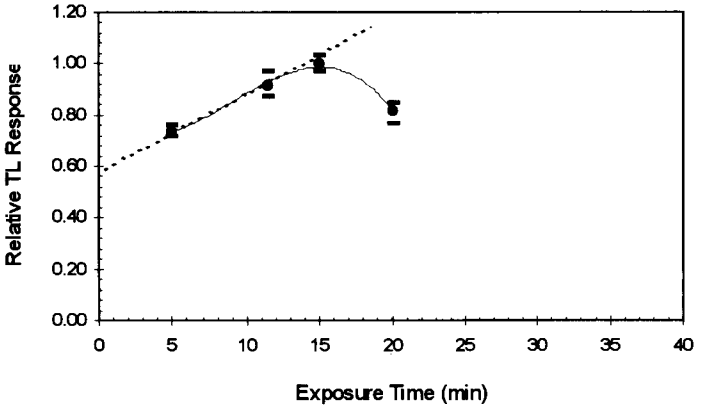

Figure 2. Dependence of TL Response as a Function of the Laser Exposure Time for $CaSO_4$:Dy

For the UV radiation was studied the dependence of the PTTL response as a function of UV exposure time. It was used a gamma dose of 10 Gy (10^3 rad) and a light wavelength of 310 nm and the time range studied was from 5 to 30 minutes. It is seen in Figure 3 that a better response was obtained for the exposure time of 20 minutes and was found to reduce for 30 minutes.

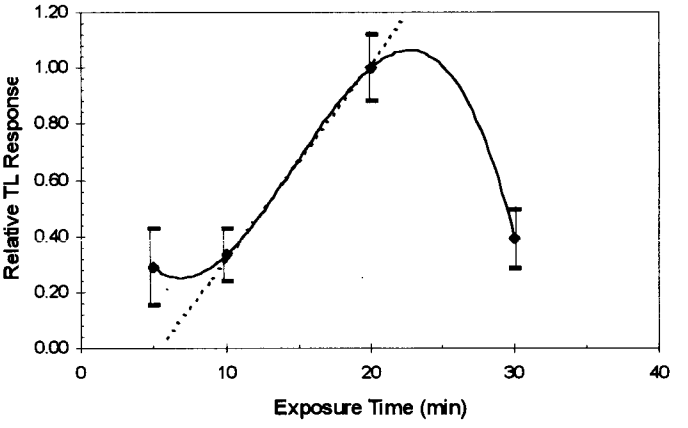

Figure 3. Dependence of TL Response as a Function of the UV Exposure Time for $CaSO_4$:Dy

The curves' behaviour of the figures 2 and 3 can indicate that there is an equilibrium between the population and depopulation of the traps placed in these temperatures. Comparing these two figures it can be seen that both curves are similar, presenting a linear and sublinear range, expected result considering their close wavelength (310 and 337 nm). It is expected that the lower limit for the laser radiation be similar to UV.

The objective of this project is to propose a theoretical model to explain this phenomenon and determine the minimum and maximum detection limits.

CONCLUSIONS

This work can be very useful to those institutions which work with non-ionizing radiation, as laser and UV and these preliminary results show that $CaSO_4$:Dy Teflon pellets present a good performance to PTTL to be used in laser and UV dosimetry and permits to continue this research.

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COMBINED TL-ESR MgO DETECTORS FOR UV RADIATION

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INTRODUCTION

The ever-increasing demand for the control of radiation field parameters, including ultraviolet (UV) radiation, produces a set of problems among which is enhancement of reliability and validity of measurements over a wide dose range including emergency dose control. Real efficiency of the most popular materials used, for example, for thermoluminescence dosimetry, is restricted by the upper level of the measured doses, 10^2 to 10^4 Gy. However, the desired upper level of the dose interval can reach 10^8 Gy at ambient temperatures of up to 500 K (1,2). To solve this kind of problem by thermoluminescence dosimetry only is apparently not possible.

One of the most promising trends in this field is development of new working solid detector substances allowing one to obtain dosimetric information when measuring the parameters of several physical phenomena accompanying the interaction of the radiation with the detector's material. An example of such an aproach in the search for new materials has been given elsewhere (3). The basic physical phenomena used were thermoluminescence (TL) and electron spin resonance (ESR).

The purpose of this presentation is to study the dosimetric properties and operational characteristics of combined TL-ESR detectors for UV radiation on the basis of MgO:Mn, Fe crystals that have been subjected to a special thermochemical treatment (STT).

EXPERIMENTAL

MgO:Mn, Fe single crystals grown by the arc-fusion method contained 0.02 wt.% of Fe impurity. The samples were subjected to thermoreducing treatment in a vacuum oven with a graphite heating unit at 1000 to 2000 K for 1 to 10 h.

The detectors were irradiated using an deuterium lamp ultraviolet light using an interference filter with λ=250 nm or with X-rays from a Mo target tube operating at 55 kV and 15 mA.

After UV or X-ray irradiation the line with g=2.0037 appears in the sample ESR spectra (Fig.1). Lines g_1 to g_6 belong to Mn^{2+} impurity and line A corresponds to Fe^{3+} impurity ions. With thermally treated samples, line A is not observed before irradiation. After irradiation this line appears again and its amplitude increases with the increase of absorbed dose.

In MgO: Mn, Fe the ESR signal with g=2.0037 is known to correspond to the Fe^{3+} impurity ions (4). The TL maximum in the temperature range of 300 to 500 K is brought about by the recombination of thermo-ionizing electrons with Fe^{3+} impurity ions (5). At the same time, thermoreducing treatment of MgO crystals results in the emergence of Fe^{2+} ions and high concentration of electron trapping centers in the samples. During irradiation, electrons liberated from Fe^{2+} ions are captured by centers created as a result of the treatment, which leads to higher concentration of Fe^{3+} ions and, accordingly, higher ESR signals. When annealing thermo-ionizing electrons recombine with Fe^{3+} ions, this results in the ESR signal drop in the temperature range 350 to 450 K and appearance of the TL maximum. When the

anneal temperature increases to 950 K, the ESR signal drops rapidly in the temperature range 350 to 450 K and more slowly in the temperature range 450 to 950 K, until the signal disappears completely at 950 K.

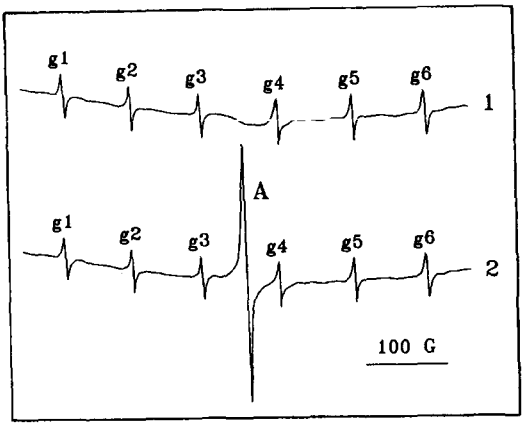

Figure 1. ESR spectra from virgin MgO:Mn, Fe single crystals (1) and crystals subjected to special thermochemical treatment (1) after (2) and befor UV or X-ray irradiation.

The ESR signal of Mn^{2+} ions does not change when irradiating and annealing. The fact that the ESR signal of thermally treated MgO:Mn, Fe crystals cannot be annealed even at high temperatures allows one to use them for measuring radiation dose at high temperatures. The dose dependence of the ESR signal of crystals irradiated at 570 K is linear in the range of 3 to 3×10^2 Gy. Under UV irradiation, the TL intensity dose dependence is linear in the irradiance range 10^{-1} to 10^3 $\mu J\cdot cm^{-2}$ (Fig.2, curve 1). The ESR signal is registered starting from irradiances of 2×10^2 $\mu J\cdot cm^{-2}$ and grows in a linear fashion with irradiance from 10^3 to 10^5 $\mu J\cdot cm^{-2}$ (Fig.2,curve 2). During the available time, the ESR signal did not reach saturation.

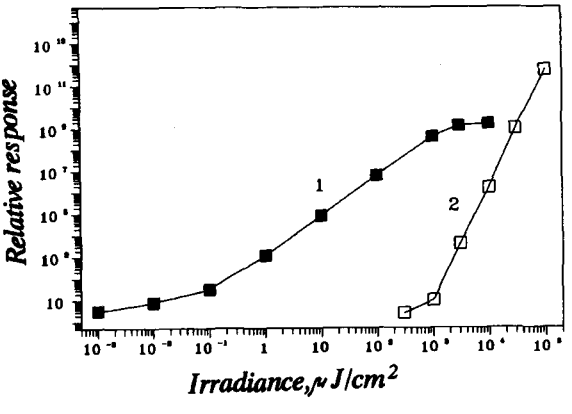

Figure 2. Irradiance dependence of the TL intensity (1) and ESR signal (2) of MgO:Mn, Fe single crystals due to UV irradiation.

Thus, MgO:Mn, Fe single crystals or ceramics can be used as combined TL-ESR detecting media after reducing thermochemical treatment.The main features of this material are the following:

1.High reliability and validity of dosimetric measurements due to readout of information from one and the same sample using two independ methods (TL and ESR).

2.Possibility of repeated reproduction of dosimetric information by the ESR signal after TL readout and influence of high temperatures ($T < 800$ K) on the sample.

3.High sensitivity and wide range of registered doses for dosimetry of UV radiation. The total range of registered irradiances using ESR and TL exceeds 8 orders of magnitude (the low level is 10^{-3} $\mu J \cdot cm^{-2}$; the upper level is greater than 10^{5} $\mu J \cdot cm^{2}$).

4.Possibly of enhancing the accuracy of dosimetric measurements for the ESR signal registration when reading out information by the ratio of the intensity of the Fe^{3+} ion line to the intensity of Mn^{2+} ion lines, whose parameters remain stable when irradiation and annealing.

5.Possibility of pre-adjustment of the TL reading devices, according to the results of ESR measurements.

6.Possibility of evaluating the current dose by the TL output and dose accumulated over a longer period of time (month, year) by the ESR output.

CONCLUSION

It can be said that the combined TL-ESR MgO:Mn, Fe detector can cover irradiance ranges of particular interest, such as erythermal dose for skin (16 $mJ \cdot cm^{-2}$) or also lower values which can be used in biological research.

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LASER: STUDY BY CONFOCAL AND ELECTRONIC MICROSCOPY OF RETINAL DAMAGES INDUCED BY MULTIPLE PICOSECOND PULSES.

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INTRODUCTION

The current laser safety standards do not define the exposure limit values for pulsewidth less than one nanosecond. It has been hypothesized that one cannot extrapolate from longer pulse widths because the ultrashort pulse contains high peak power and may induce non linear effects. These effects may involve fundamentally different mechanisms of damage particularly for repetitively ultrashort pulses (1-2). Most laser safety documents advise the laser user that caution must be used in the evaluation of exposure to repetitively pulsed radiation since they are only limited data on multiple pulse exposure criteria. The empirical multiple pulse formula is based on some data indicating that there is generally a cumulative effect in multiple-pulse exposures. No data exists in the litterature concerning the histological effects on the retina of ultrashort pulses delivered with a high repetition rate (3). The aim of this work was to study the evolution of single retinal lesions after exposure to trains of picosecond laser pulses. The experimental approach we have developped corresponded to :
1) observation of 100 µm thick and to characterize lesions by fluorescence microscopy after different specific stainings,
2) epoxy resin embedding of lesion areas for further studies by light and standard electron microscopy.

METHODS

The experimental system is a dye laser pumped by a mode locked pulse compressed, frequency doubled Spectra-Physic Nd:YAG laser. The pulsewidth was 8.10^{-12} s. The pulse repetition rate used in these experiments was 1 million pulses per second (1 MHz) and the number of pulses included in the train was 200,000 pulses. The wavelength at the output of the dye laser is 590 nm. A sample of the laser beam at 590 nm was deflected onto an autocorrelator which measures the pulsewidth and onto a photodiode and a calorimeter which measure the energy. The animal used in the experiments was the rabbit Fauve de Bourgogne because the average ocular transmission and absorption characteristics are very similar to the human ocular media for the visible spectrum (4). The theorically expected spot size formed on the rabbit retina with the dye laser optical system was about 30 µm. To aid alignment of the animal's eye, a 1 mW HeNe laser was mounted co-axially with the main laser beam path. Rabbits were anesthetized with intramuscular injection of ketamine hydrochloride (10 $mg.kg^{-1}$), placed on a stand permitting us to produce in most cases retinal lesions aligned on the visual streak. The pupils were dilated before to exposure, and the eyelids were held open with

a wire speculum. 3 to 5 lesions were produced in each eye which were distributed over about 1 cm^2 area to overcome interactions. Animals were killed 4.5, 24, 48 and 96 hours after exposure. Retina were fixed overnight with 2% glutaraldehyde 2% paraformaldehyde in buffered solution (0.1 M cacodylate pH 7.3) after gradual dissection of the eye immersed in the fixative. The exposed area was cut, treated during at least 2 hours in the buffer containing 10% saccharose and 10% DMSO before freezing. Nuclei were stained with propidium iodide (PI) which allows to identify easily condensed DNA in apoptotic cells and a O-Gal residues were stained by bandeiraæ simplicifolia lectin labelled with FITC which visualizes blood vessels and macrophages (5). The treatment and procedures used in this study conformed to the European Community Guidelines on Animal Experiments.

RESULTS

Light microscopy

Injuries induced by higher energies (Total intraocular energy > 3 mJ) could produce damaged areas larger than the retinal spot size of the laser beam. The swelling and rising of the retinal layers permite to identify easily the damaged area (6). The common observations are:

- the Brüch's membrane appears undamaged,
- the retinal pigment epithelium is disturbed: the repartition of the melanin granules is inhomogeneous,
- the outer plexiform layer is dammaged with an area of vacuolization.

At the opposite, for smaller energies, the damaged area corresponding to the threshold values cannot be easily determined because pathologic changes are very subtles.

Confocal microscopy

The retinal image diameter of the beam was 15-30 µm whereas the diameter of the early œdema detected by fluorescein angiography during the first hour post treatment was 150-200 µm.

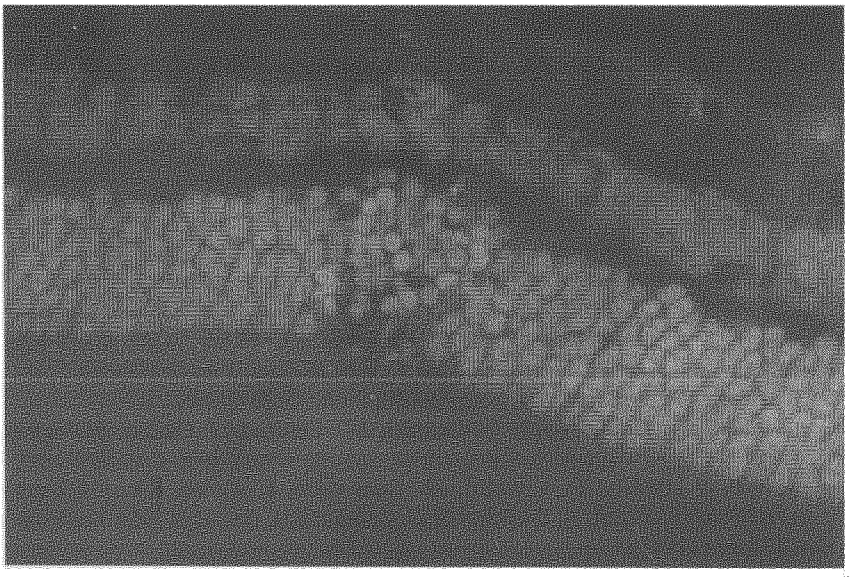

Figure1. Retinal damage induce at the outer nuclear level detected by confocal microscopy with propidium iodine staining (590 nm 200,000 pulses, 1 MHz, TIE ≈ 3 mJ).

Lower injuries can be detected within 50 to 100 µm thick. The technical choice of specifical stainings permited to visualize the damage induced to the inner grannular layer, the pigment epithelium and the choroid. The mixture PI-Lectin permited to observe picnotic or apoptotic nuclei. Preliminary results are given in the Table I. The homogeneously distributed apoptotic cells were confined to focal lesions in the outer nuclear layer after 4.5, 24 and 48 hours respectively. Macrophages were the only inflammatory cells in these lesions and were mainly involved in tissue repair by phagocytosis of apoptotic bodies and altered segments after 24 and 48 h. Only a few macrophages were encountered per lesion (<10).

Delay post irradiation	4h 30	24h	48 h
Diameter of the lesion	78 to 87 µm	84 to 100 µm	45 to 54 µm
Number of macrophage	0 to 1	0 to 1	2 to 4
% Apoptosis	20%	80%	10%

Table I. Evolution of the lesional parameters at different delays post laser irradiation.(590 nm 200,000 pulses, 1 MHz, TIE ≈ 3 mJ)

Electron microscopy

Each lesion were studied by confocal microscopy and some of them thereafter by electron microscopy. Inflammatory process could be induced by ultrashort pulses laser irradiation. Inflammatory cells such monocytes, polynuclear cells were observed at the junction of inner and outer segments. Cellular alterations were detected in the outer nuclear layer very similar to those observed during the cellular death induced by apoptosis (5). Electron microscopy allowed to characterize at 4.5 h pigmented epithelial cell alterations and to observe intralesional unphagocyted material up to 48 h. No lesion were detected after 96 hours suggesting a nearly complete repair.

Acknowledgments-This study was supported by contract 94/001 from the Direction des Recherches Etudes et Techniques du Ministère de la Défense.

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1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Secondary Optical Radiation of High Power Laser Beam Welding

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MAJOR SCIENTIFIC TOPIC NUMBER 7.3. (see page 7)

ABSTRACT (See instructions overleaf)

With laser materials processing, direct exposure to the laser beam is usually not the main hazard. The high power laser beam is enclosed up to the surface of the workpiece and the workstation is usually shielded. Secondary hazards such as UV-radiation, ozone and fume produced by the laser-workpiece interaction are more serious due to the every-day, long term exposure of the worker.

In this paper, results of spectroscopic measurements of the secondary optical emissions associated with CO_2 laser welding of steel and aluminum will be presented. Spectral measurements of the plasma emission at a distance of 50 cm showed that the allowed dose for UV-radiation and short wavelength visible light ("blue light hazard") per work day can be exceeded in as short as a few seconds and tens of seconds, respectively. Particularly high exposure levels were found when the welding process became instable and plasma shielding occured. Model calculations showed that the plasma temperature is at the order of 13,000 K, hence the maximum of the spectral irradiance is in the UV region. The visible part of the radiation is relatively low, which could cause a feeling of false security, because it possible to look into the plasma without closing the eyes.

BIOLOGICAL RESPONSES OF NIH 3T3 TO 193 NM EXCIMER LASER IRRADIATION. COMPARISON WITH 254 NM IRRADIATION.

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1. INTRODUCTION

During conventional UV radiations, activation of genes is tightly linked to the presence of DNA damages. At 254 nm, the major cellular chromophore is the nuclear DNA, with cyclobutane pyrimidine dimers being the major photoproduct (1,2). At 193 nm, DNA is more strongly absorbing than at 254 nm. However, the quantum yields of the photoproducts induced at these two wavelengths are different (3) and damage to cellular DNA in the form of pyrimidine dimers or single-strand breaks was reported to be only marginal or undetected (4,5). Peptide bonds, many amino acid side chains, insaturated lipids, esters, and other cellular molecules absorb at 193 nm (6,7). Consequently, the most of the energy is thought to be absorbed by proteins, leaving the DNA shielded. About 60 % of the radiation can apparently be blocked by 1 µm of cytoplasm (8). However, exposures of cultured human fibroblasts to subablative doses of 193 nm laser radiation resulted in changes of genes expression such as collagenase, metallothionein and *c-fos* (9). The mechanisms by which the 193 nm radiation affects gene expression are not known. The site of primary interaction of the radiation could be different from the site of the genetic response. Thus, the signal transfer could pass through the cytoplasm via the nucleus. One hypothesis is that cytokines may regulate the transduction pathway event. By example, the TNF-alpha which is induced by UV-radiations (10). More, it can activate transcription factors such as AP-1 or *c-fos* and stimulate the growth of normal fibroblasts (11). Moreover, TNF-alpha plays a major role in the inflammatory processes by enhancing the remodeling of extracellular matrix (12,13) in which mainly matrix metalloproteinases and collagenase participate. Besides, matrix metalloproteinases are responsive to cytokines and particularly the 92 kDa gelatinase (gelatinase B or metalloproteinase 9 or MMP 9) is induced and regulated by TNF-alpha (14).
To understand the cellular response to high energy laser radiation, we investigated cell proliferation and cell activation. The results showed a decrease of the cell proliferation and an increase of the cell activation. These variations are more marked after the irradiation at 193 nm. In addition, we have quantified the variations of TNF-alpha and gelatinase B in supernatants of fibroblasts irradiated with 193 nm laser radiation or 254 nm conventional UV radiation. The results showed that both wavelengths caused increased rate of TNF-alpha and of the gelatinase B (expressed under two forms, latent and active) but the laser was significatively more efficient.

2. MATERIAL AND METHODS

2.1. *Cell cultures*

The mouse fibroblasts NIH 3T3 cells obtained from American Type Culture Collection were grown in RPMI 1640 medium containing 10% of decomplemented fetal calf serum (FCS - Boehringer Mannheim), 20 mM L-glutamine and 100 µg of antibiotic mixture (penicillin, streptomycin, neomycin - Gibco-BRL) at +37°C in 5% CO_2. Two days before irradiation, $1.5 10^6$ cells were plated into 8.5 cm diameter plastic tissue culture dishes (Falcon). Prior to irradiation, the medium was removed and the cells rinsed with phosphate-buffered saline (PBS-Boehringer Mannheim). Irradiations as well as sham exposures of controls (zero dose) were performed in the absence of any solution, with plates held either vertically in the case of laser irradiations or horizontally in conventionnal UV irradiations. After irradiation, cells were plated for 8 hours into 96-well plates at 10^5 cells /well, with 200 µl medium per well.

2.2. Irradiations

Laser irradiations were performed with a SEL 520 SOPRA excimer laser at 193 nm (ArF), with nominal pulse duration of 13 ns. Laser beam was defocussed to illuminate an 8.5 cm diameter cell culture plate (56.7 cm^2) held vertically. The energy per pulse, measured with a joulemeter Molectron JD2000 detector, was 57 mJ/pulse, corresponding to a peak irradiance of $7.7\ 10^8$ W/m^2. Cells were exposed to fluences of 10, 50 and 120 J/m^2 by varying the number of pulses delivered with 1 Hz repetition rate. Irradiations at 254 nm was performed with a Bio-Sun system (Vilbert-Lourmat) where UV energy, delivered by 3 UV-VL20 lamps with a constant irradiance of 4 W/m^2, is controlled by microprocessor.

2.3. *Cell proliferation*

Cell proliferation was assessed by (methyl-^{3}H) thymidine incorporation.

2.4 *Cell activation*

A tetrazolium colorimetric assay, using the dye MTT [3-(4,5-dimethylthiazol-2yl)-2,5-diphenyltetrazoliumbromide], measures the reducing ability of mitochondrial enzymes.

2.5 *Quantification of total proteins*

Total proteins was quantified according to the sensitive method described by Bradford (15).

2.6 *Quantification of TNF biological activity in culture medium*
TNF-alpha secretion was measured by a biological activity test using actinomycin-D-treated L-929 cells as described by Fish and Gifford(16).
2.7. *Gelatin zymography*
Gelatinolytic activity was performed in cell supernatants as described by Heussen and Dowdle (17).
2.8. *Statistical analysis*
It was performed according to Student'st-test for unpaired data (StatView II software, Abacus concept Inc., Meylan, France). $p < 0,05$ was considered significant.

3. RESULTS
3.1. *Number of cells*
The number of cells was measured immediately and 8 hours after the irradiation. At 193 nm, the number of cells decreased sharply in a dose-dependent manner. The most pronounced decrease was observed 8 hours after the exposure. At 254 nm, a similar dose-dependent decrease was measured 8 hours after irradiation. A high mortality of cell (36%) was detected at 120 J/m^2 at 193 nm just after the irradiation, a smaller one was seen (15%) 8 hours later at the same dose. In opposite, at 254 nm no mortality was observed at any dose just after the irradiation and 8 hours post-irradiation, the cell mortality detected was at 8%.
3.2. *Cell proliferation and activation*
The proliferation of NIH 3T3 cells is markedly reduced, reaching about -75% at 120 J/m^2 after each kind of irradiation. In return, both wavelengths induced a dose-dependent cell activation in irradiated cells but the augmentation was about 2.4-fold greater after 193 nm laser exposure than after 254 nm irradiation at 120 J/m^2.
3.3. *Quantity of total proteins*
The wavelength of 193 nm induced a dose-dependent increase of the amount of total proteins in irradiated cells. Even though at 254 nm the increase was not significant between 50 and 120 J/m^2. The increase was 1.6 times greater after 193 nm laser irradiation than after 254 nm at 120 J/m^2. Irradiations induced cycloheximide-dependent increase amount of protein.
3.4. *TNF-alpha and Gelatinase B activity* (Figure 1)
A large dose-dependent increase was observed after 193 nm irradiation. In contrast after 254 nm irradiation, no increase was detectable except for the fluence of 120 J/m^2 but the TNF activity was 2 times lower. Both irradiations at 193 and 254 nm cause a significant and dose-dependent increase of both latent and active forms of excreted gelatinase B. The proportion of the two forms was identical but the augmentation was about 2.1-fold greater after 193 nm laser exposure than after 254 nm irradiation at 120 J/m^2.

4. DISCUSSION
To our experimental conditions, an important cell mortality is induced only by 193 nm laser radiation. This suggests that radiation damage induced by laser 193 nm exposure differs from that of 254 nm radiation. At 193 nm, the maximal contribution of radiation injury to cell death was observed immediately after irradiation. Cells either are too injuried and died within minutes, or are not damaged enough and survive after repair. At 254 nm, a wavelentgh which is known to produce DNA damages, no immediate cell death was observed and mortality was still low 8 hours later. The nuclear damages occuring at 254 nm irradiation do not induce an immediate cell death. Then, the cell mortality observed within minutes after irradiation may likely originate from an alteration of the cell membrane. The high absorption by cytoplasm and membrane proteins may account for the damages induced by 193 nm photons and the high cell mortality observed immediately after laser exposure. The decrease of the cell proliferation is important after 193 nm radiation. Cell activation which was really important after 193 nm irradiation, as shown by the cycloheximide dependent increment of total proteins. This suggests that cells have tried to repair their damages. This effect on cell activation is different from the one published by Kochevar and al, (18) who found a dose-dependent inhibition of the MTT reduction. Cell activation is increased and could be due to the increment of TNF-α which is involved in gene expression regulation and proteins synthesis. Indeed TNF-α ellicit a rapid induction of *fos* mRNA (19) and, it is well known that activation of *fos* is part of a typical response of mammalian cells to UV-C light (20). Our results confirm this *fos* induction which occur in fibroblasts following irradiation at 193 nm. The level of TNF-α could be considered as a parameter permiting to evaluate the damaging effect of 193 nm photons on the plasma membrane because it may be related to the change of arachidonic acid metabolism occurring in the altered plasma membrane (21,22). Kochevar and al, (18) have demonstrated an increase in cell membrane permability associated with a release of arachidonic acid within 15 minutes after exposure to 193 nm radiation whereas exposure to 254 nm radiation did not cause this release. They suggest that plasma membrane damage induced by 193 nm radiation was obvious from the release of arachidonic acid. This release of arachidonic acid observed by Kochevar and al, and the secretion of TNF-α in cell supernatants measured in our experiments seem to occur at approximately the same fluences (23) shown that arachidonic acid which is also released in response to TNF, induced *c-fos* mRNA. Furthermore Gelatinase B, involved in the degradation process of basement membrane, is more expressed after laser exposure, supposed to be changing the cell adhesion process. The results indicated that the 88 kDa active form which is involved in the TNF-α activation was specifically increased after 193 nm laser irradiation. However, the quantification of TNF-α biological activity in cell supernatant could be originated from the membrane compartment release from

damaged cells by 193 nm photons. The Gelatinase B increase could be also induced by extra cellular TNF-α but cell activation observed with the lowest fluence suggest that the origin of the quantified TNF-α did not only issue from membrane compartiment of dead cells. Moreover, the 88 kDa active form of Gelatinase B contribute to increase the quantity of TNF-α produced in surviving cells.
Thus, the results suggest that the plasma membrane is the site of the primary damage in cells exposed to 193 nm laser radiation. The mechanism of interaction between 193 nm photons and the plasma membrane is unknown but the primary effects of the 193 nm radiation are a cell activation, a more important expression of Gelatinase B and of TNF-α. This latter cytokine could be the signal of induction from the plasma membrane to the nuclear transcription factors of genes whose expression was specifically changed by 193 nm laser radiation.

Acknowledgments - This work was supported by Direction des Recherches Etudes et Techniques (DRET) from Ministère de la Défense.

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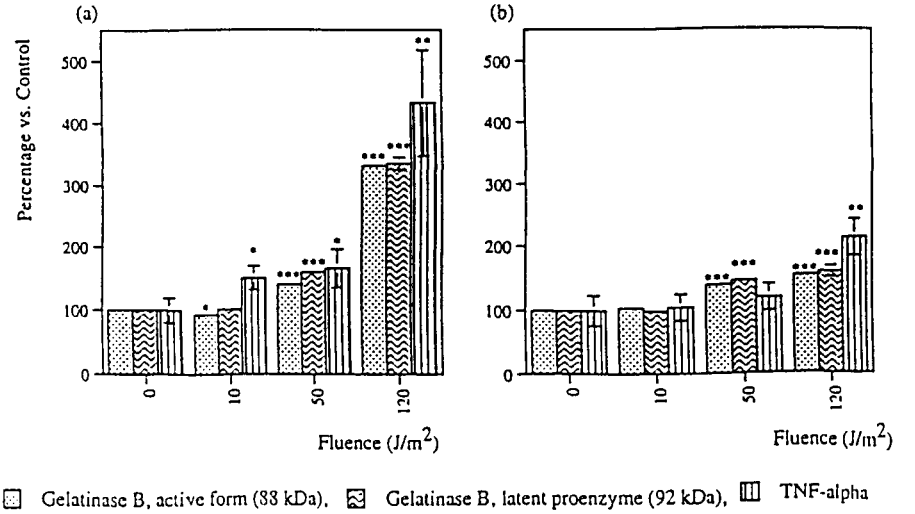
Figure 1 : Gelatinase B activities and TNF-alpha biological activity in cell supernatants, 8 hours after excimer laser irradiation at 193 nm (a) or UV conventional irradiation at 254 nm (b).
(*** p < 0,0007 ; ** p < 0,006 ; * p < 0,04)



Quantitative risk assessment - an alternative approach to laser safety?

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Introduction

The current international laser safety standard (IEC 825) gives a system of laser classification based solely upon the laser parameters without reference to the intended use. The procedure makes pessimistic assumptions to arrive at a "safe" classification related to the maximum hazard.

Hazard [1] *A hazard is something with the potential to do harm*

The manufacturer's requirements and the user guidance go on to manage the hazard by mitigating the potential for harm of higher classifications by introducing interlocks etc. Risk management is not considered in current laser safety standards.

Risk [1] *Risk expresses the likelihood that the harm from a particular hazard is realised*

It can be argued that the classification should also be based upon the intended use to which the laser will be put. The basis for this type of hazard analysis would have to be a risk assessment. Such a system of classification has an obvious advantage in that it would allow a wider variety of products to fall into the lower classes by virtue of their low risk.

This idea is attractive but the authors have reservations about its practicability. There are many different possible methods for making a risk assessment which vary in complexity and in the criteria used. It is difficult to see how a laser standard could adequately specify risk assessment techniques which are suitable for use in any and all situations. Furthermore, labelling would have to ensure that the intended use was unambiguously identified.

Such a system can only be necessary if classification is used as an overriding indicator of "safety", which currently it is not. It is risk that indicates the level of safety, being related to both the harmful consequences and the likelihood of occurrence.

Legislation and Risk Assessment

In 1974 the Health and Safety at Work etc Act (HSWA) came into force in the UK. This Act places duties on employers in respect of their employees and other persons (eg visitors and the general public etc) affected by work activities. The UK Ministry of Defence (MOD) is not exempt from this Act or Regulations which are made under it.

These impose on UK employers a "duty of care" to their employees (in the case of MOD this includes its sailors, soldiers and airmen) and the requirement to provide "a safe place of work" under HSWA. In addition they have to abide by the specific Regulations. MOD can seek exemptions from these on grounds of national security. However, it is policy to use this provision sparingly and only where truly essential and not for administrative convenience.

A European Union (EU) Directive caused the Management of Health and Safety at Work Regulations 1992 to be made in the UK. These require employers to assess all risks in the workplace even where specific Regulations do not exist. Similar regulations should also exist in all EU member states. In making such assessments MOD conforms to national or international standards and guidance wherever practicable. The assessment must be recorded if the risks are "significant". Assessments are open to inspection by the enforcing authority for the legislation, in the UK this is the Health and Safety Executive (HSE). After inspection HSE may instruct the MOD to make improvements or prohibit the practice.

In most cases quantitative risk assessment (QRA) will not be required to demonstrate safety. The deterministic methods used in current laser safety standards will be quite adequate. However, in some cases, deterministic methods lead to unrealistically large, and therefore unduly restrictive, hazard areas [2]. Where these exist QRA may be justified depending upon the complexity and the resources needed to carry it out.

Quantitative risk assessment - an example

Current laser safety standards are based upon preventing an exposure above the level of a Maximum Permissible Exposure (MPE). There is an implicit assumption made therefore that circumstances can be arranged such that this can be guaranteed.

When one considers a class 3B laser rangefinder, being operated from an aircraft, the opportunities for failure make such a guarantee is difficult to give. Figure 1 illustrates the off axis angular limits of a forward firing laser rangefinder being used with the Nominal Ocular Hazard Distance (NOHD) to sweep out a deterministic laser hazard area trace (LHAT).

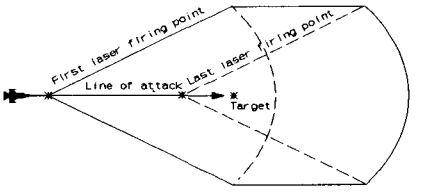

Figure 1 - Deterministic Laser Hazard Area Trace

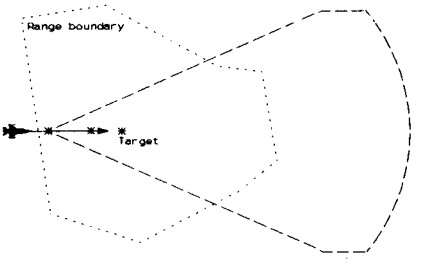

Figure 2 - Deterministic LHAT and range boundary

Often the deterministic LHAT may extend beyond the range boundary, Figure 2, and thus appear unacceptable. This certainly occurs in the UK and Europe, but perhaps not in the US or where ranges cover larger areas.

Under these circumstances a question needs to be answered - "what is the risk?" MOD uses QRA[3,4] to investigate the probability that the laser would in fact point at anything other than the target and the consequences if this should happen.

The elements used in the QRA are:

- equipment performance and fault condition parameters
- an eye damage model for the wavelength concerned
- atmospheric models for scintillation and absorption
- population models for the areas around the range

Since the MPE is an exposure at which no harm is caused it is more appropriate to use another criterion to assess the risk of harm. The eye model therefore uses the "minimum ophthalmoscopically visible lesion" (MOVL).[5]

Using a risk of 1 in 10^8 MOVL per attack the LHAT obtained is typically of the form shown in Figure 3. It has 2 components made up of a fault-free LHAT and a fault condition LHAT.

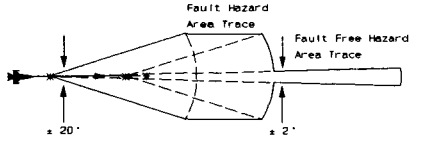

Figure 3 - Components of a QRA LHAT

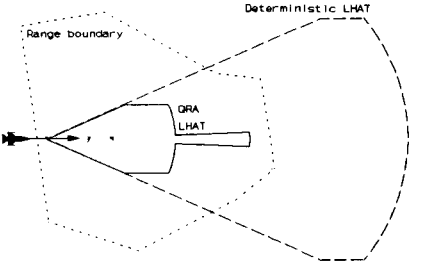

Figure 4 - Deterministic and QRA LHATs compared

It is usual that the QRA LHAT obtained by combining both the fault and fault-free LHATs is contained within the range boundary. Figure 4 illustrates this by overlaying the range boundary with both the deterministic and QRA LHATs.

Consideration of the Extended NOHD and the equivalent probabilistic modelling are beyond the scope of this paper. Never the less, the possibility of persons using binoculars is modeled and contained within the overall

risk criterion endorsed by MOD of 1 in 10^8 of achieving a MOVL per attack. Particular attention is paid to areas where the probability of binocular use is high, e.g. bird sanctuaries etc.

Tolerability of risk

The risk of 1 in 10^8 MOVL compares favourably with some other figures quoted by HSE[6], particularly as it is non-fatal:

Levels of fatal risk per annum (average figures, approximated)

1 in 100	risk of death from five hours of solo rock climbing every weekend
1 in 1000	risk of death due to work in high risk groups within relatively risky industries such as mining
1 in 10 000	general risk of death in a traffic accident
1 in 10 000	risk of death in an accident at work in the very safest parts of industry
1 in 1 million	general risk of death in a fire or explosion from gas at home
1 in 10 million	risk of death by lightning

or indeed the results from practical examples of QRA:[7]

- 1 in 10^6 per 8 h shift for failure of 3 safety interlocks on an injection moulding machine (usually non-fatal)
- fatal risk of 1·5 in 10^9 per hour of winding of a coalmine elevator (carried out after an accident caused 18 fatalities and which caused improvements to be made)

It might be suggested that the figure of 1 in 10^8 MOVL is impossibly small to be used as a practical estimate of risk. The authors tend to agree, but argue that while it may not be useful as an absolute measure, it is useful as an quantitative indicator of relative risk. So although it may not be combined with other estimates of risk, eg aircraft crash probability, to give an overall mission risk, it does indicate that laser safety is a minimal factor. In a similar way the authors would argue that such an assessment cannot and indeed should not seek to include factors for deliberate or negligent misuse of the laser. Consideration should certainly be given to the prevention of such occurrences and measures may need to be taken to prevent them, ie by adequate pilot/navigator training and supervision, backed by good design. However, unless they are shown to be reasonably foreseeable, they should not form part of the laser safety risk assessment.

The future

Current legislation and the regime of enforcement in the UK are adequate to enforce laser safety. QRA as a technique to assess the tolerability of risk is in use by both MOD and UK industry in various areas of safety.

The enforcing authority, HSE, recognise this and give guidance on its use:[8]

> "Assessing risks is necessary in order to identify their relative importance and to obtain information about their extent and nature. .....
>
> Risk assessments should be carried out by competent people, and professional health and safety advice may be necessary in some cases, especially in the choice of appropriate QRA techniques and the interpretation of results."

Specific laser safety legislation based solely upon MPEs would therefore, in our opinion, be too restrictive. If proposed EU legislation[9] on physical agents continues to call solely upon the European Standard or anything similar, MOD must seek to ensure that its current techniques and safe practices are not unduly affected. The authors suggest that industry and other users might also wish to preserve the flexibility of approach offered by risk analysis which does not either degrade or prevent the improvement of laser safety.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE MODIFICATION OF RADIATION-INDUCED MEMBRANE AND CELL DAMAGES BY HALOGEN-CONTAINING COMPOUNDS (HCC)

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MODIFICATION OF RADIATION-INDUCED MEMBRANE AND CELL DAMAGES BY HALOGEN-CONTAINING COMPOUNDS (HCC)

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It was found a strong increase in UV-radiation (270-300 nm) induced destructive damages of proteins, lipids, and isolated membranes of blood cells and lysis of the cells in the presense of some HCC polluting the environment.The investigations carried out on model systems (tryptophan and protein in solution) indicated that HCC promote the photo-induced breakdown of indol ring. According to chromatographic and spectral characteristics the products formed under such conditions are distinct from earlier known those of tryptophan destruction. The studies on artificial lipid and cell membranes showed that under absorption of UV-radiation of ecological range (λ > 300 nm) they are capable to play a role of"inside" photodynamic sensitizers.In the presence of HCC non-connected with the products indicated an increase in photo-induced oxidation of lipids also takes place. This effect was observed on native membranes, but not on liposomes. The form and maximum position of the action spectrum of this process evidence the involvement of protein tryptophan residues and a certain contribution of photodestruction of α-tocopherol. Similar increase in cell membrane damage in the presence of HCC was registered under ionizing radiation action.The problems of potential danger of discovered modification of radiation-induced damages of cell membranes in functioning cells and organisms under conditions of technogenic environment pollution, augmentation of near-UV component of solar radiation reaching the earth surface and increased radiation background are discussed.

NON-IONISING RADIATION: EXPERIENCE OF COMMUNICATING THE HAZARDS

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ABSTRACT

Scientific advice relating to non-ionising radiation protection, partly through necessity, contains a degree of scientific terminology. The advice has to be directed at a variety of audiences including users and members of the public, many of whom may not be familiar with scientific terminology. This paper presents NRPB's practical experience of effectively conveying this advice to the various audiences.

INTRODUCTION

The study of human interactions with non-ionising radiation and the determination of any deleterious health effects which may arise as a result of this interaction is complex, involving scientists from a wide variety of disciplines. In such a multi-disciplinary scientific environment the language used to communicate ideas and express opinions must, through necessity, be technically precise and rigorous. It is perhaps to be expected that when such a community concludes its work, reaches a considered opinion and proposes its views as best current scientific advice, the advice is phrased in the same language. Here lies a basic problem. If communication, whether written or verbal is to be effective it must be understood by its audience; and specialist scientists form but a small part of the overall audience.

The National Radiological Protection Board is responsible for the provision of advice to Government Departments of the United Kingdom, to corporate bodies and to members of the British general public in relation to protection against the hazards of both ionising and non-ionising radiation and it is with this large audience that the Board must communicate. Resources are finite and must be used in the most effective manner. This requires the various audiences and their needs to be clearly identified. The Government is perhaps the focus for the Board's formal advice on standards, but this paper focuses on three audience groupings; the radiological protection community; the general public and the users (employers and employees), principally the latter group. A number of approaches will be explored as will the interaction between these audiences.

RADIOLOGICAL PROTECTION PROFESSIONALS

Radiological protection professionals are in the best position to receive advice from the Board in its formal advice format. The Documents of the NRPB is the current route by which the Board promulgates its formal advice on non-ionising radiations (1-3). These publications are supported by the NRPB Report and Memorandum series where the research results are more extensive, and by peer-reviewed scientific papers in the open literature. These and various specific meetings provide the mainstream, of communication for those at the forefront of non-ionising radiation work. However, account also has to be taken of the radiological protection professionals whose background and training has focused on ionising radiations and who have had their area of expertise extended either by design or evolution to include non-ionising radiations.

Training can be a very effective form of communication, whether by directly training the individual or by training the trainers. The Board offers two professional-level training courses: the Advanced Radiological Protection (ARP) course aimed at the seasoned radiation safety professional, and the Post Graduate Radiological Protection (PGRP) course aimed at meeting the early training requirements of those embarking upon a career in radiation protection. Board staff, and invited speakers, communicate directly with the radiological protection community, influencing current thinking, promoting consistency and the development of a common approach in addressing radiological protection matters. Whilst still dominated by ionising radiation matters, the non-ionising radiation content is increasingly important.

The Board also provides a series of scientific seminars and produces a monthly Radiological Protection Bulletin, in which non-ionising radiation topics figure prominently.

GENERAL PUBLIC

Before looking at what may appear to be the next logical group, the users, it is necessary to first consider the general public. This is because the users, ie the employers and employees, are of course also members of the general public and will often reflect the generally held perceptions.

The public receive most of their information about non-ionising radiation and its potential health effects from the media. The information will be acquired by the media from press releases, high profile activities arising from litigious matters and from published scientific reports and papers. Again we come back to the basic problem that the source material is in a highly technical form and needs to be translated into plain language that allows the public to relate the potential hazards to their own experience. This process may be helped through press releases and briefings, and by providing radio, television and press interviews. This can be resource intensive, but it is necessary in order to provide a careful blend of scientifically accurate information and summaries in plain language. How this is used is another matter; the media has its own priorities that sometimes lead to selective presentation of information, sometimes producing a skewed or alarmist view of things. Nevertheless there are many balanced and thoughtful articles and programmes. Wishing away the bad bits is unproductive: we simply have to recognise realities and continue to influence the media in as honest a way as possible.

With non-ionising radiation issues we have the advantage of being able to learn from the pitfalls encountered previously in the ionising radiation field. However there is also the very large disadvantage of the word "radiation" and the pre-conception in the public's mind that this is automatically associated with cancers. Often the message to be communicated is that the possible health effects are different and that the hazards are significantly lower than the public's perception. However one can have equal difficulties getting over the message that common practices are more hazardous than generally accepted. A recent example has been the Board's advice on ultraviolet radiation (3) and the hazards associated with direct exposure to the sun and to sunbeds. This is perhaps not a message that many in the general public want to hear, as it directly affects their lifestyles, but it is an important matter and one that will be a challenge to our communication skills over a number of years.

The Board has put a lot of effort into another communication medium, namely a series of "At-a-Glance"leaflets. These are aimed at the interested layperson and are produced in both graphical and pictorial form utilising colour to enhance the visual impact. Relevant information, statistics and facts are presented in bullet point style. Each leaflet covers a particular subject area, for example Radon, Nuclear Emergencies, Medical Radiation, Transport of Radioactive Materials, Natural Radiation, Non-ionising Radiation, Ultraviolet Radiation and Electric and Magnetic Fields from the Use of Electricity. The At-a-Glance series has proved to be very successful with over one million copies distributed to date.

The Board receives a steady stream of telephone calls and letters from members of the public and following media coverage of non-ionising radiation matters this increases. Many enquiries can be dealt with by a package of information including reprints from the Board's journal, the Radiological Protection Bulletin and the At-a-Glance series. However, many questions require further input from scientific staff. Whilst there is clearly a need for a national body to be in a position to answer such questions in this way, it is resource intensive and that is one of the reasons for focusing effort on the more cost effective routes of media communication and leaflets.

THE USERS

Whilst there are some documents available that give specific guidance to users of non-ionising radiation, these are not as extensive as those for ionising radiation, partly due to the developing nature of the subject and partly due to the lack of hazard specific legislation. It is perhaps an area for improving communication in the future.

One area that has developed significantly over the past couple of years is the range of training courses available to communicate at this level. These have to be focused, both on the audience, eg safety professional, management, staff etc; and on the specific use of non-ionising radiation. The latter vary considerably: for example, from the localised nature of manufacturing uses of RF, such as heat sealing of PVC, to the communications industry where RF radiation is radiated from antennas; and for lasers from simple low power laser pointers through medical laser scalpels to fully interlocked high powered lasers. The direct needs of safety professionals are often easily addressed in straightforward lectures, however at the other end of the scale many trainees will not have been in a classroom environment since leaving school. For the latter group the use of good visual representations and practical demonstrations and measurements are essential. They can also be very useful to the safety professional in suggesting ways they may pass on understanding to a wider audience.

To address these needs the Board runs a range of training courses and has also collaborated with Loughborough University in developing laser safety training that addresses not just the radiation hazard but others such as electrical and fume hazards.

Many practical situations require measurements of non-ionising radiations as part of an overall assessment, whilst others are standard situations that have been previously well documented. Even in the latter case, measurements of non-ionising radiations as part of consultancy work may be necessary to allay fears of the workforce. For example in recent years concerns have been expressed in some quarters over the exposure to electromagnetic fields from visual display units (VDUs). Although the scientific evidence suggests that employees will not be harmed by these fields (4), enquiries are still received by the Board. Copies of publications and telephone reassurance may be sufficient to reassure some employees. However, in some cases this may not be sufficient and an employer may be in the situation where a number of employees are refusing to work at VDUs, or wishes to confirm that there is nothing wrong with a particular VDU. Under these circumstances it may be necessary to visit the premises to carry out measurements for reassurance purposes. These should confirm that the emissions from the VDUs are below the Board's investigation levels. However, if the results are presented in a written report, the employee and possibly the employer, may be overawed by the formal nature of the document. A more successful approach is to carry out a series of measurements with the concerned employees and trade union representatives present. They can be asked to read the meters on the measurement instrumentation (with care to ensure they do not influence the measurement results) and compare these measurements with the investigation levels. Any questioning can then be in the small group or on a one-to-one basis. Most employees are trusting of professionals who are prepared to go through this very open and direct process.

CONCLUSIONS

As in other areas of safety the development of scientifically sound advice is just one component of developing radiation protection for non-ionising radiation: resources to effectively communicate that advice are essential. The means of communication and the level of delivery must take account of the various audiences. This is not a new message, but one we should routinely remind ourselves of.

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IRPA9

1996 International Congress on Radiation Protection

Proceedings / Volume 4

Vienna, Austria, April 14-19, 1996

Organised by the Austrian Association for Radiation Protection

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PAPER TITLE CHARACTERIZATION OF HUMAN NORMAL AND TUMOR CELLS IRRADIATED WITH REPEATED LOW DOSES OF GAMMA RAYS

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ABSTRACT (See instructions overleaf)

Lately, a new type of the consequence caused by radiation have received much attention. It was found that tumor cells exposed to fractionated ionizing irradiation, with a high total dose, may become resistant to chemotherapeutic agents. The aim of the present study was to determine whether the same phenomenon will occur in human normal and tumor cells after exposure to a low total dose of irradiation divided in a small number of fractions. Since c-*myc*, c-Ha-*ras* and *p53* oncogenes are involved in both drug-resistance and neoplastic transformation, we also examined their expression in normal preirradiated cells.

Primary human lung fibroblasts or human cervical carcinoma HeLa cells were irradiated five days with daily dose of 0.17 Gy of gamma rays. The sensitivity of preirradiared cells (after 5 or 10 γ-ray-fractions) to different cytostatics was determined by colorimetric MTT dye assay and compared to the sensitivity of control cells. The expression of c-*myc*, c-Ki-*ras* and *p53* oncogenes in control and preirradiated cells was detected immunocytochemically. The results show that preirradiated cells became resistant to vincristine and vinblastine, but did not change their senisitivity to doxorubicin, 5-fluorouracil, mitomycin C or cisplatin. The expression of examined oncogenes was the same in normal preirradiated and control cells.

Our results suggest that irradiation of human cells with multiple small doses of gamma rays may change their susceptibility to chemical agents. These data may have practical application to radiation protection determination.

RELATION BETWEEN RADIO-ADAPTIVE RESPONSE AND CELL TO CELL COMMUNICATION

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INTRODUCTION

Ionizing radiation has been considered to cause severe damages to DNA and do harm to cells in proportion to the dose, however low it might be. In 1984, Wolff et al. showed that human peripheral lymphocytes adapted to the low-dose radiation from ^{3}H-TdR added in culture medium and became resistant to the subsequent irradiation with high-doses of X-rays (1). This response, which is called radio-adaptive response, is also induced by X-rays and gamma-rays in human lymphocytes (2) and Chinese hamster V79 cells (3). However, the mechanisms of and conditions for adaptive responses to radiation have not been clarified. With an objective of clarifying the conditions for adaptive responses of cells to radiation, we examined how the cell to cell communication is involved in the adaptive responses. We irradiated normal human embryo-derived (HE) cells and cancer cells (HeLa) in culture at high density with low-dose X-ray and examined their radio-adaptive responses by measuring the changes in sensitivity to subsequent high-dose X-ray irradiation using the Trypan Blue dye-exclusion test method. We also conducted experiments to examine the effects of Ca^{2+} ions and Phorbol 12-Myristate 13-Acetate (TPA) which are supposed to be involved in cell to cell communication.

MATERIALS AND METHODS

(1) Dose-dependency of Preliminary Irradiation Effect

Four hours after the start of culture of HE cells, the cells were irradiated with low doses of X-ray ranging from 0 to 20 cGy. Similarly, HeLa cells were irradiated with X-ray ranging from 0 to 40 cGy. The cells were cultured for another four hours, then subjected to high-dose irradiation of 200 cGy. After high-dose irradiation, the cells were cultured for five days before the number of viable cells was counted. The relative growth ratios of the respective groups were determined. The control groups received mock treatments for both the low-dose irradiation and high-dose irradiation.

(2) Effects of Calcium Ions and TPA

Four hours after the start of the culture of the HE cells, the regular culture medium was replaced by Ca^{2+}-containing Hank's Balanced Salt Solution (HBSS) or Ca^{2+}-free HBSS. Then each group received irradiation of 13 cGy or mock treatment. Ten minutes after the irradiation

or the mock treatment, the HBSS were replaced by the regular medium. Then the cells were cultured for four hours before receiving a high-dose irradiation of 200 cGy. TPA was added to the culture medium at a concentration of 100 ng/ml at 3 hours and 40 minutes after the start of culture of HE cells, then the cells were irradiated with 13 cGy 20 minutes after the addition of TPA. The culture medium was removed 20 minutes after irradiation at 13 cGy, the cell sheet was rinsed 3 times with PBS, and the regular culture medium was poured to the culture flask. The cells were cultured for 4 additional hours, and then irradiated with 200 cGy. After that, the cells were cultured for five days, and the numbers of viable cells were counted to determine the relative growth ratios of the cultures. The control groups were treated in the same manner as the irradiated groups except that both the low-dose irradiation and the high-dose irradiation were mock treatments.

RESULTS AND DISCUSSION

(1) Dose-dependency of Preliminary Irradiation Effect

The results are shown in Fig. 1. When a high-dose irradiation of 200 cGy was given to HE cells, the growth ratio of the living cells decreased to 37 % of that of the control. When a preliminary irradiation of 10 to 20 cGy was given to HE cells four hours before the irradiation of 200 cGy, the relative growth ratio significantly increased to 45-53 %; and a peak was reached at a preliminary irradiation dose of 13 cGy. In HeLa cells, however, preliminary irradiation by the entire dose range from 0 to 40 cGy did not affect the relative growth ratio after 200 cGy irradiation.

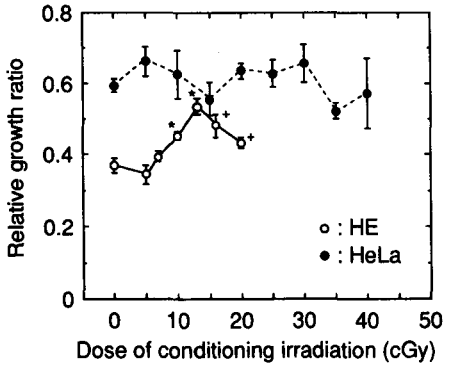

Fig. 1 Changes in relative growth ratio of pre-irradiated human embryonic cells and HeLa cells after receiving the challenging irradiation of 200 cGy, as a function of pre-irradiated conditioning dose. Relative growth ratio = growth ratio of irradiated cells / growth ratio of sham-irradiated control. Each point is the mean ± SD of six independent determinations. $^{*}p<0.01$, $^{+}p<0.05$ by paired t-test.

(2) Effects of Calcium Ions and TPA

The results are shown in Fig. 2. When the HE cells were given an irradiation of 200 cGy without any preliminary irradiation, the growth ratios of the living cells decreased to 31-37 % of those of the control groups irrespective of the presence or absence of Ca^{2+} in the HBSS

which replaced the regular medium. When an irradiation of 13 cGy was given to the HE cells in the Ca^{2+}-containing HBSS four hours before the 200 cGy irradiation, the relative growth ratio of the living cells increased significantly to 53 %. However, when the HE cells were given an irradiation of 13 cGy in the Ca^{2+}-free HBSS, the relative growth ratio of the living cells was about 32 %.

When TPA was added to the culture medium at 100 ng/ml during preliminary irradiation, the relative growth ratio of HE cells significantly decreased to 42%.

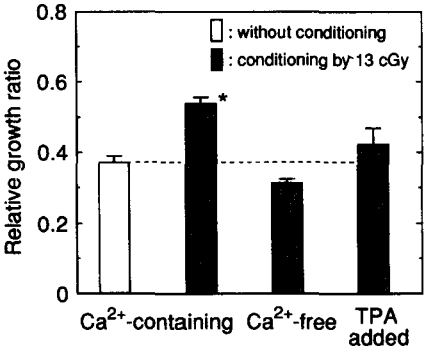

Fig. 2 Changes in relative growth ratio of pre-irradiated human embryonic cells after receiving the challenging irradiation of 200 cGy by the presence or absence of calcium ions and TPA in the medium during the conditioning irradiation. Each point is the mean ± SD of six independent determinations. *$p<0.05$ by paired t-test.

In the present study, to examine how cell to cell communication is involved in the adaptive responses of cells to radiation, we irradiated normal cells (HE) and cancer cells (HeLa) with low-dose X-ray and examined changes in their sensitivity to subsequent high-dose X-ray irradiation. As a result, the preliminary irradiation of the HE cells by low-dose X-ray moderated the decrease in the growth ratio of the cells due to the subsequent 200 cGy irradiation, and increased the radioresistance. This adaptive response was not observed in HeLa cells. When the HE cells were suspended in a Ca^{2+} ion-free medium or TPA added medium while receiving the conditioning irradiation, the adaptive response was not observed. The above-mentioned findings suggest a good possibility that low-dose irradiation enhances the radio-resistance of normal cells. The results also show that there is an optimum dose range for inducing the radio-adaptive response. This radio-adaptive response seems to involve cell to cell communication maintained in the normal cells (4).

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PAPER TITLE Effects of γ-quanta in the 0.1-50 cGy dose range on the conformation of chromatin in mammalian cells.

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ABSTRACT (See instructions overleaf)

The effect of ^{137}Cs γ-quanta on the chromatin conformation was studied by the method of anomalous viscosity time dependence. A few different cell types were exposed: VH-10 human fibroblasts, BALB\C mouse splenocytes, Sprague-Dowley rat thymocytes. The cells were irradiated within the 0.1-50 cGy dose range and then lysed for viscosity measurements. It was established for all types of cells that exposure to low doses of 0.5 cGy resulted in a statistically significant reduction of viscosity peaks. This reduction reached a maximum value approximately 40-60 min after irradiation. The reduction of viscosity was revealed at doses up to 4 cGy for human fibroblasts with the maximum effect of about 2 cGy. The opposite response, increase of viscosity, was observed after cell exposure to 10-50 cGy doses. From the linear approximation of this dose dependence, the increase of viscosity started at a dose of about 3±1 cGy. The effect of increased viscosity disappeared with time after irradiation, with kinetics similar to that of DNA repair. Repair of the chromatin conformation depended strongly on temperature in the 0-37°C range. To the contrary, kinetics of the 0.5 cGy effect did not depend on temperature. Thus, two different types of the chromatin response were observed in mammalian cells after low (0.5 cGy) and high (50 cgy) doses of irradiation. This study provides new evidence that doses as low as 0.5 cGy cause significant changes of the chromatin conformation in different types of mammalian cells.

BIOPHYSICAL ANALYSIS OF RADIATION-INDUCED CHROMATID AND CHROMOSOME ABERRATIONS AT LOW DOSES AND DOSE RATES

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INTRODUCTION

In the present study, two sets of experimental data have been analyzed: (i) *in vitro* and *in vivo* exposures of human blood lymphocytes to alpha particles from short-lived radon decay products (1,2), and (ii) *in vitro* irradiation of rat hepatocytes with gamma radiation (with and without pre-irradiation) (3). The biophysical multi-stage model for radiation-induced chromosomal damage used in these analyses is derived from earlier models for radiation-induced transformation (4,5) and liver carcinogenesis (6).

BIOPHYSICAL MODEL

A biophysical model for the production of radiation-induced chromatid and chromosome aberrations as a function of dose or dose rate has been developed, which is based on experimentally observable cellular mechanisms, such as the formation of single and double strand breaks or cellular inactivation. In this multi-stage model, a sequence of five states is assumed in the development of a cell from the initial unirradiated level (state 0) to a cell exhibiting chromosomal damage (state 4) (Figure 1). The intermediate steps in this sequence of events are: production of an unspecific DNA single strand break (rate k_{01}); production of a second specific DNA single strand break (k_{12}); interaction of both DNA single strand breaks to form a double strand break (rate k_{23}); division-related fixation of the DNA double strand break thereby producing chromosome aberrations (rate k_{34}); and, division-related fixation of two single strand breaks leading to chromatid aberrations (rate k_{24}=m). Single and double strand break formation may be reduced by adaptive mechanisms, while actual breaks may be repaired (rates k_{10}, k_{21} and k_{30}). Cells in the various states can be removed by radiation-induced cell death (rate k_d) and apoptosis (rate k_a), and increased by mitosis (rate m). In case of alpha irradiation, double strand breaks can also be produced by single alpha particle intersections (rate k_{03}). The values for the transition rates between the various states in this model were derived from *in vitro* cellular experiments.

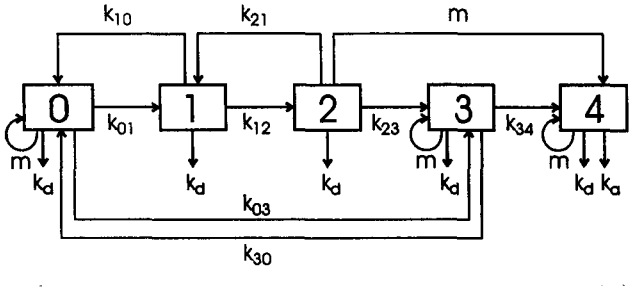

Figure 1. Biophysical multi-stage model for the analysis for radiation-induced chromatid and chromosome aberrations produced by alpha and gamma irradiation. Transition rates are described in related text.

ANALYSES OF EXPERIMENTAL DATA

In the first analysis, chromosome aberrations in human blood lymphocytes were exposed to short-lived radon decay products under *in vitro* irradiation conditions (1). Blood samples were irradiated by ^{214}Po alpha particles at doses between 0.02 and 26.9 mGy. The aberration data were collected from 11 experiments conducted during the period 1984 to 1992; a total of 64816 metaphases was scored in eight collaborating laboratories. Figure 2 illustrates the total chromosomal aberrations (dicentrics, rings, interstitial and terminal deletions) as a function of dose for blood samples taken between 1984 and 1986, i.e., prior to the Chernobyl accident. The data plotted here are based on 10687 scored metaphases. After a steep linear increase at very low doses, the aberration frequency gradually drops with increasing dose. The comparison between with these experimental data and our model simulations indicates that the biophysical model developed here is capable of reproducing the functional form of the experimentally observed dose-response curve.

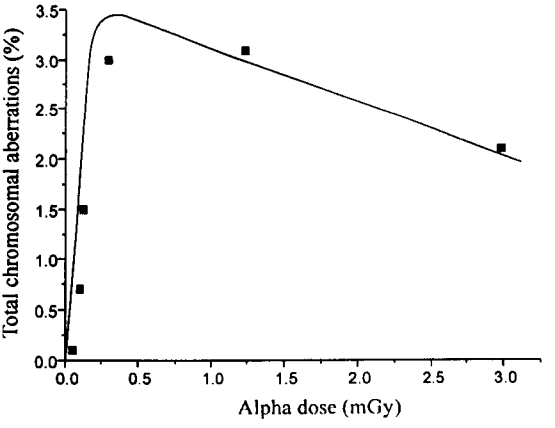

Figure 2. Total chromosomal aberrations (dicentrics, rings, interstitial and terminal deletions) in human blood lymphocytes for low dose *in vitro* alpha particle irradiation. The full squares refer to the experimental data, while the full line represents the model simulations.

A similar dose-reponse relationship has been reported for low dose and dose rate *in vivo* radon progeny irradiation of blood lymphocytes in miners and medical staff working in the Badgastein thermal gallery: total chromosome aberrations first rise linearly with dose up to about 7 mGy/year and then decrease with increasing alpha dose (2).

In the second analysis, primary cultures of adult rat hepatocytes were irradiated with ^{60}Co gamma radiation at much higher doses of 0.01, 0.1, 1 and 5 Gy at a dose rate of 33 Gy/h (3). In order to study potential effects of adaptive response mechanisms, the same set of doses was also applied to hepatocyte cultures pre-irradiated with 2.5, 10, 100 and 200 mGy 24 hours prior to exposure. As shown in Figure 3, induction of chromosomal aberrations for both normal and pre-irradiated cells rises sharply with dose in the low dose region and then remains level at higher doses. Most importantly, however, all pre-irradiation doses caused a significant reduction of chromosomal aberrations, which was highest for a pre-irradiation dose of 2.5 mGy. Our theoretical predictions of the shape of the dose-response relationship as well as of the effect of the induction of adaptive response mechnisms by pre-irradation are consistent with the experimental evidence.

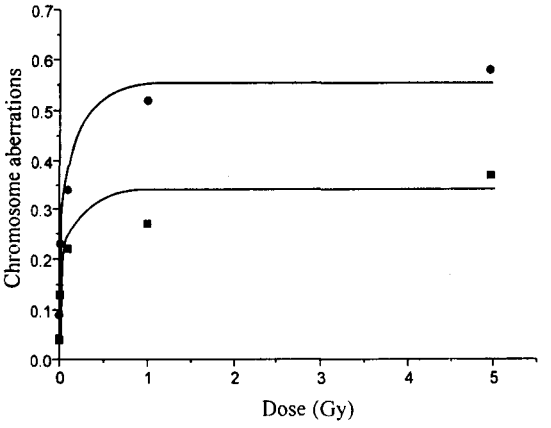

Figure 3. Total chromosomal aberrations in primary rat hepatocytes by *in vitro* gamma irradiation. The full circles (without pre-irradiation) and full squares (2.5 mGy adaptive dose) denote the experimental data; the full lines refer to the corresponding model simulations.

CONCLUSIONS

The three sets of experimental data analyzed in this study cover different dose regions (very low, low and high), cell types (lymphocytes, hepatocytes), radiations (alpha, gamma) and irradiation conditions (*in vitro*, *in vivo*, pre-irradiation). Despite these differences, a consistent pattern emerges. If chromosome aberrations are plotted as a function of dose, four distinct regions can be observed: First, at the lowest doses, no chromosomal damage can be statistically detected; above a certain dose level, the frequency of aberrations increases almost linearly with dose; following the induction of adaptive mechanisms, the dose-response may either exhibit a plateau or even decrease, depending on pre-exposure and dose rate; and, finally, chromosomal damage rises again at sufficiently high cellular doses.

Comparing our theoretical results with these three sets of experimental data, we may conclude that our model correctly predicts both the shape of the dose-response curve as well as the action of adaptive response mechanisms induced by pre-irradiation.

ACKNOWLEDGEMENTS

This research was supported in part by the Forschungsinstitut Gastein-Tauernregion, Project FPK 76.

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TRITIUM RADIOBIOLOGY

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INTRODUCTION

Actuality of the tritium biological effect researches is caused by its wide spreading due to accumulation of technogenic tritium in biosphere as well as the possibility of increasing of persons working with this nuclide because of real perspective of the thermonuclear energy production. Information about tritium effect in conditions, that is closely like the real conditions, i.e., under chronic isotope intake, is especially important.

The materials of many-years work of authors' group have been generalized in this paper on the kinetics of forming of early and late effects following the prolonged influence of tritium in broad range of doses including low level.

MATERIALS AND METHODS

Studying the chronic influence of tritium oxide (HTO) has been conducted for two species of laboratory animals on the different levels of integration (from molecular to population) by using the radiobiological, biochemical, hematological, cytogenetic, immunological, and pathomorphological methods. Animals were received the HTO during 3-6 months (dose rates of β-exposure ranging from 0.1 to 27 cGy/day, and absorbed doses ranging from 0.1 to 25 Gy).

RESULTS AND DISCUSSION

The study of tritium behaviour kinetics in chromatin of hematopoietic tissues under the prolonged HTO injections by 370 kBq/g per day has shown that 70-80% of nuclide were incorporated into DNA, and 20-30% of those were incorporated into chromatin proteins. The half-period of accumulation of tritium in DNA of hemopoietic tissues was 15-25 days. The excretion of label was being during two time period: 60-70% of label for 5-8 days; 30-40% of label for 12-18 days. Effectiveness of including tritium into carbon relations of DNA from water was 0.5-0.8, and if taking into account the easily exchanging tritium and tritium of 8th carbon of purine, it is approximately 1. Activation of the hematopoietic tissue proliferation (hemolytic anemia) did not influence on both the level of tritium incorporation into DNA and the effectiveness at the nuclide equilibrium period.

The common radiation burden on hemopoietic tissues as well as tritium inclusion into chromatin led to changes of DNA structure and reparation depending from dose, and also to nucleic acid metabolism changes. For prolonged injection of tritium oxide (37-1850 kBq/g per day, the absorbed doses of 1.2-28 Gy), the strong damages of DNA primary structure (by content of double-stranded DNA, and by single-stranded DNA molecular mass) were determined at and of nuclide injections in the animals, only which received moat amount of nuclide. The findings relate to DNA reparation at moment of tritium oxide intake in animals. The residual damages of DNA were remained after the intake end, and were found by additional test-exposure.

Content and rate of DNA biosynthesis depended from the dose rate and the absorbed dose. At first two months of β-exposure to incorporated tritium in dose range of 1-12 cGy/day, the DNA concentration in hemopoietic tissues was on subnormal

level, and that for high dose was on decreased level, despite the increase of absorbed dose. This was because of dose-depending activation rate of DNA synthesis from 14th day and practically during all period of nuclide injecting. However, under high values of rate of beta-exposure (the absorbed doses of 12 Gy), the decreasing of DNA content was indicated for studied tissues (thymus - 65% of control) at end of influence. After short-time recovery of this index, when radionuclide intake has finished, the secondary decreasing of DNA content was observed in animals of this group. The significant decrease of this index in hemopoietic organs corresponded the increasing of substances (polydeoxyribonucleases), that indicated to the DNA destruction, and reflected possibly the cell elimination with non-reversible radiation damages.

The DNA structure damage and their reparation developed to chromosome aberration. Cytogenetic studies of *Wistar* rats found the high sensitivity of myelocariocyte chromosomes to tritium damaging effect. During HTO injection (3.7-370 kBq/g per day, dose rates of 0.12-12 cGy/day, and absorbed doses of 0.24-25 Gy), the chromosome damages were mainly the aberrations of chromotide type in animals of all groups. The aberrations of chromosome type were being appeared with increasing the absorbed dose, which were mainly symmetrical exchanges. The time points of their appearance and frequency depended on the dose rate, and for maximum rate these damages began to be observed in 60 days after influence. At relatively later period the chromosome type aberration appearances and the absence of significant increase of chromosome damage in animal groups, when absorbed doses were being increased, evidence that there are adaptive processes under prolonged intake of tritium. The stable aberrations of chromosome type were being kept up to end of observation after cessation of HTO injections in animals, and the cell clones with atypical chromosome were determined for most of rats that were received the higher doses.

The nucleic acid metabolism changes developing under chronical effect of HTO in hemopoietic tissues made a basis for shifts of cell population kinetics in these tissues. It was found the dependence of hemopoiesis damages from nuclide dose burden under prolonged β-exposure to incorporated tritium with different dose rates (for rats -- 0.12-12 cGy/day, and for mice -- 0.04-4.4 cGy/day). The difference of tissue reactions of animals, which received the different HTO amounts, were expressing as different rates and depopulation. It was shown high sensitivity of multipotent and committed cells, that are precursors of myelo- and lymphopoieses, to tritium in comparison with morphologically recognized hemopoietic cells. The erythropoiesis damages under chronic effect of HTO was more clear, than granulocytopoiesis.

The clear dependence of character and fullness of recovery from radiation effect level was shown for bone marrow and lymphoid tissues after finishing of HTO injection. The recovery of cell content in these tissues of animals exposed to beta-radiation in dose of 12.5 Gy (dose rate of 12.0 cGy/day) was in 1-3 months later than after lower doses of exposure. The value of non-reversible decreasing of multipotent precursor compartment was being higher with increasing the absorbed dose. On this background, the recovery of lymphopoiesis, especially T-cell production, was defective. The population of committed precursors of T-lymphocytes stabilized on 40-60% level in comparison with B-lymphocyte precursors whose population reached the values of age norm in 3-6 months after the end of HTO intake. Also, the recovery of cellular structure in thymus and lymphatic nodes, which cells were mainly T-lymphocytes, was not full, and at the same time the cell content in spleen (B-cells) was being increased up to control level in 6-9 months.

HTO-induced changes of lymphopoiesis at different stages were being developed as the dose-depending damages of humoral and cellular immunity. During excretion of tritium oxide by amount of 370 kBq/g per day (mice, rats), the antibody production

was being lower up to 50% of control. Suppression of humoral response on the antigene at earlier time periods was caused mainly by insufficiency of T-helpers, and later of B-lymphocytes. Under prolonged intake of HTO by less amount, the antibody production changed during short-time period (for 185 kBq/g per day), or kept on level of age norm (for 37 kBq/g per day). The reliable dependence was obtained between the absorbed dose and the character and fullness of recovery of humoral and cellular immunity after the end of β-emitter intake. Antibody production, activity of normal killers, and T-inactivating lymphocytes recovered in 1-3 months after exposure to radiation in doses of 3-4 Gy, and in 9-12 months after the accumulated dose of 8.7 Gy (*CBA* mice).

The relative biological effectiveness (RBE) of tritium oxides in respect with γ-radiation (^{137}Cs) was determined basing on dependence of dose-effect after one-time injection of nuclide under the absorbed doses of 0.5-14.5 Gy. Taking into account the indexes of lymphoid organ's mass, the cell content in hemopoietic tissues, the damages of DNA structure and of nucleic acid metabolism, it may be indicated that RBE of tritium increased under decreasing of absorbed dose, and was 4-6 under minimum doses. The $DL_{50/30}$ value of RBE of tritium corresponded 2.32.

The late effects of tritium oxide intake are most important for assessment of its danger, particularly its cancerogenous influence. The publications on blastomogenous effects of tritium do not present the precise dose-depending outcome of malignant tumours at different sites (1,2). The late effects of prolonged beta-exposure to tritium incorporated were studied by us for following doses: 0.24; 2.02; 12.5; and 25.3 Gy. The frequency of malignant tumours in exposed *Wistar* rats exceeded the control level at 1.5-2.5 times. Revealed neoplasm at different sites may be divided on three categories by character of dose-effect dependence. Tumours of first category had the positive correlation with dose (leukaemia, lung cancer, skin cancer, osteosarcoma, and mammary gland cancer), at the same time tumours of second category had (lymphosarcomas of lung) were characterized by negative dose-dependence. The malignant tumours of endocrine organs, testicles, kidneys and urethra, gastro-intestinal tract did not have the reliable dependence on dose (3d category).

The difference of dose-effect dependence for frequency of different neoplasms induced by tritium makes the difficulty to use this criterion, as character of radiation danger. The non-stochastic index of dangerous effect for organism, such as lifetime shortening, is more integral and universal. The prolonged injection of tritium oxide by minimum amount (dose of 0.24 Gy) increased the mean lifetime (MLT) of animals by 12%, and in cases of injection of large amount of nuclide (dose of 2.02 Gy) the lifetime of animals did not differ from value of control. The further increasings of radiation dose up to 12.5 and 25.3 Gy led to decreasing of MLT of animals by 18% and 23%, respectively. However, decreasing of the MLT of animals was the same either in rats with malignant tumours, or in other animals. The absence of priority of malignant tumours for decreasing of the MLT makes possibility to think that there is single mechanism of thanatogenesis either for nature senescence, or for its acceleration, in particular, by radiation exceeding the specified level.

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PAPER TITLE Plausible molecular mechanisms of morphological abnormalities in coniferous species after Chornobyl accident

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1. (see page 7)

ABSTRACT (See instructions overleaf)

A lot of territories was contaminated with radionuclides (mostly ^{90}Sr and ^{137}Cs) after Chornobyl accident. The level of external γ-radiation in the Chornobyl region is from 1 to 40 mR/h at different lots. Some coniferous species (pine and spruce) form many different morphological abnormalities under these conditions of chronic irradiation. The changeability of needles' length during some vegetation periods is one of them. So, pine needles become essentially shorter (2-3 fold) in some years. Spruce trees have both lengthened and shortened needles as well as needles with normal length on the same tree. It was shown previously that the size of cells in abnormal needles is also different from the normal one. Since the cell size is closely connected to cytoskeleton functions so it was supposed that ammount of cytoskeleton proteins in modified cells must be different from the normal one. Such difference can be caused by changes in synthesis of cytoskeleton proteins as well as by changes in cell genome. Significant changes in protein spectrum and in individual proteins (actin, α- and β-tubulin, hsp70 etc.) were registered with immunochamistry and SDS-PAGE. With the aim to study genome of modified cells, DNA from them was examined with restriction fragment length polymorphism method, dot- and Southern-hybridization with β-tubulin gene. Some changes in DNA from abnormal needles were observed. It was supposed, that chronic irradiation could affect the genome stability of coniferous plants.

THE RADIOPROTECTIVE ROLE OF TYASTIME ON CHLOROPLASTS OF PHASEOLUS VULGARIS L.

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Introduction.

There are numerous studies about the effects of low doses of gamma ionising radiation. Unfortunatelly the probabilistic nature of interaction and the existence of too many parameters characterizing the interacting parts make difficult the appearance of reproducible results. In this work we present a part of our results regarding the action of gamma ionising radiation on both chloroplasts ultrastructure and pygments dynamics from Phaseolus vulgaris L. (1) Also, the radioprotective action of Tyastime was discussed.

Materials and method.

The Phaseolus vulgaris L. seeds were treated with Tyastime solution of 0,5‰, 1‰ and 1,5‰ concentration.

Using a Co^{60} source, the each sample of treated seeds was supposed to theree gamma irradiation doses of 10 Gy, 30 Gy and 50 Gy in two rates. (D = 10 Gy/min., D = 3,33 Gy/min.)

The ultrastructure modifications werw studied by electron microscopy and the dynamics of assimilating pigments, in two succesive generations (M_1, M_2) was observed using the H. H. Strain and A. A. Svec technique (2)

Results and discussion.

1. The Tyastime radioprotective effect on chloroplast ultrastructure

At irradiation rate of 3,33 Gy/min., no important changes of Chloroplasts ultrastructure were observed.

The tylacoidal system was destroyed and a large number of amilophere inclusions appeared at the irradiation rate of 10 Gy/min., especially at 30 Gy absorbed dose. (fig. 2).

Tyastime 1‰has not an efficient radioprotection at 30 Gy.

The granes have a different thylacoides per grana, and thylacoidal system from stroma were broken by the appearence of both amiliferes inclusions and plastoglobules. (fig. 3)

The most efficient radioprotective role was observed at the samples treated with 1,5‰ Tyastime and irradiated at 30 Gy, when the chloroplasts (fig. 4) have a nearly identical structure with the control sample. (fig. 1)

2. The Tyastime radioprotective effect on assimilating pygments dynamics.

The Assimilating pygments were studied after 1‰ Tyastime treatment followed by irradiation at D = 10 Gy/min.

From the Table I, it can see the strongly effect of irradiation on the first generation of Phaseolus vulgaris L., plants (M_1) and the radioprotective effect which appeared only at the second generation
(M_2).

The processe of pygments accumulation and the ratio chloropyl "a" / chlorophyl "b" have similar changes in the two generations.

According to the Lohman model (3), Tyastime contribute to the membrane stabilization owing to - SH groupes.

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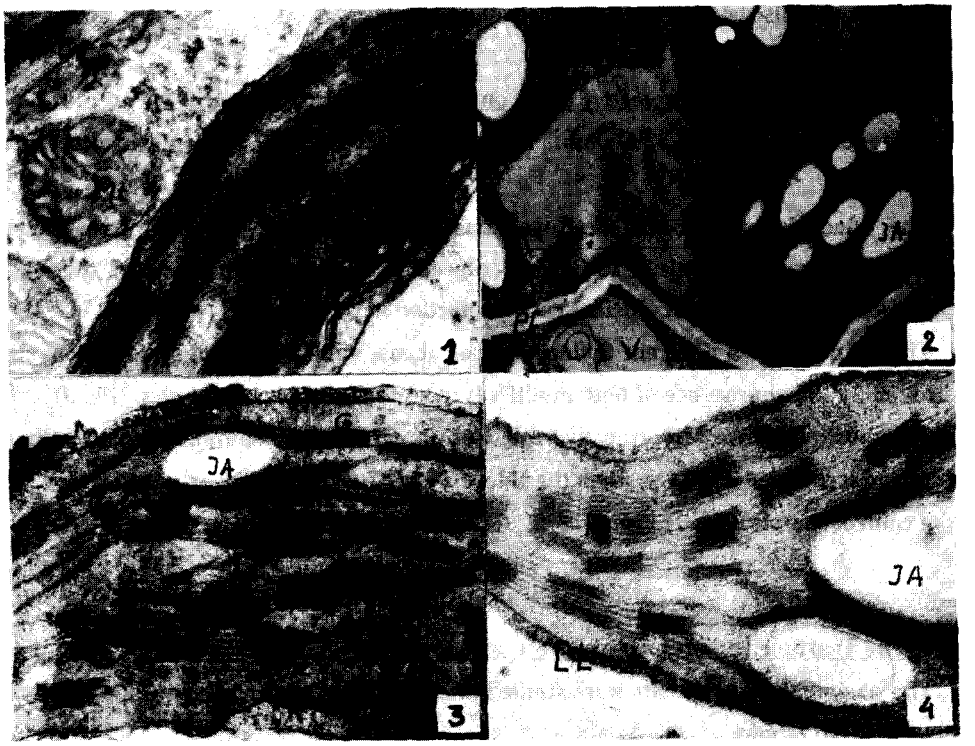

Fig.1=Control,Fig.2= 30 Gy,Fig.3= 30 Gy+ Tyastime 1% ,
Fig.3= 30 Gy+ Tyastime 1,5 % .

TABLE 1

VARIANTS	CHLOROPH.a		CHLOROPH.b		TOT.CHLOROPH		CHL.a/CHL.b		CAROTENOIDS	
	M_1	M_2	M_1	M_2	M_1	M_2	M_1	M_2	M_1	M_2
CONTROL	1,32	1,24	0,41	0,37	1,74	1,62	3,22	3,35	0,42	0,71
10 Gy, D=10 Gy/min	0,84	1,35	0,26	0,47	1,11	2,04	3,23	3,29	0,24	0,80
30 Gy, D=10 Gy/min	0,45	1,26	0,44	0,32	1,91	1,67	3,29	3,92	0,40	0,77
50 Gy, D=10 Gy/min	0,76	1,81	0,16	0,57	0,92	2,41	4,77	3,17	0,99	1,04
10 Gy,D=3,33Gy/min	0,61	1,37	0,15	0,32	0,77	1,91	4,06	2,63	0,21	0,89
30 Gy,D=3,33Gy/min	0,58	1,18	0,17	0,42	0,75	1,61	3,41	2,81	0,19	0,75
50 Gy,D=3,33Gy/min	0,82	1,70	0,20	0,52	1,03	2,24	4,10	3,26	0,24	1,00
T 1%	0,79	1,64	0,21	0,99	1,02	2,18	3,76	1,70	0,22	0,96
T 1%+10 Gy	0,53	1,62	0,14	0,48	0,68	2,12	3,78	3,37	0,16	1,01
T 1%+30 Gy	0,74	1,80	0,17	0,57	0,92	2,39	4,35	3,15	0,21	1,25
T 1%+50 Gy	0,79	1,68	0,24	0,52	1,04	2,22	3,29	3,23	0,45	1,00

D = RATIO
T = TYASTIME

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PAPER TITLE Synthesis stress proteins under action different stress

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1 (see page 7)

ABSTRACT (See instructions overleaf)

Stress proteins (metalothioneins 1 and 2, and heat shock protein 70) induction for 14-days *Pisum sativum* seedlings has been investigated under different stress factors. Acute and chronic irradiation, growth of seedlings in Cu^{2+}, Cd^{2+} and ^{137}Cs salt solutions, and high temperature were used as stress factors. Dependence of stress proteins synthesis of plants from different physical and chemical factors were investigated by the methods of Laemmli-electrophoresis and western-blotting with antibody against MT-1, MT-2 ad HSP-70. The stress proteins participation in the protection of plants from different enviromental conditions are discussed.

FAST NEUTRON IRRADIATION EFFECTS ON LIVER CHROMATIN STRUCTURE

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EXPERIMENTAL

The growing interest in neutron therapy requires complex studies on the mechanisms of neutron action on biological systems, especially on chromatin (the complex of deoxyribonucleic acid-DNA with proteins in eukaryotic cells).

The chromatin was extracted from a normal tissue-livers of Wistar rats-and from a tumoral tissue-Walker tumour maintained on Wistar rats. For chromatin separation, the method indicated in (1) was used, with a supplementary nuclei's purification through a passage on 1M sucrose, and the obtained purity verified by an absorption test (2). Suspension of cells-5×10^6 cells/ml concentration-from Wistar rat livers and from Walker tumours were also isolated. Irradiation doses from 5Gy to 100 Gy by fast neutron intense beams produced via d(13.5MeV)+Be (thick target) reaction at Bucharest U-120 Classical Cyclotron were used (3).

To study the post-irradiation effects, various methods were employed. So, the variation in the 260nm absorbency in chromatin thermal transition was pursuit (E_0-before and E-after transition). The chromatin-ethidium bromide complexes fluorescence with λ_{ex}=480nm and λ_{em}=600nm was analysed. To determine chromatin DNA strand breaks a fluorimetric method (4), with cells' suspensions as starting material was used. This method requires a partial treatment with alkali, producing three components: T-estimating the total fluorescence of DNA double helix, P-assigning the untwisting rate and B-the blank, where DNA is completely unfolded. The percentage of DNA double strand,-D-, remaining after this treatment, is:

$$\%D=100\times(P-B)/(T-B)$$

The intrinsic chromatin fluorescence was determined for tyrosine (λ_{ex}=280nm, λ_{em}=305nm), specific for basic chromatin proteins, and for tryptophane (λ_{ex}=290nm, λ_{em}=345nm), specific for acid chromatin proteins. Polyacrylamide gel electrophoresis was performed as indicated in (5).

The double fluorescent labelling of chromatin was realised with acridine orange for DNA and with dansyl chloride for chromatin proteins (6). Fluorescence intensity determinations were done with λ_{ex}=505nm, λ_{em}=530nm for acridine orange and with λ_{ex}=323nm, λ_{em}=505nm for dansyl chloride.

A Pye Unicam SP 1800 spectrophotometer and a Aminco SPF 500 spectrofluorimeter were employed.

RESULTS AND DISCUSSION

For normal chromatin, a diminution of the negative fluorescence intensity for chromatin-ethidium bromide complexes with the increasing of neutron dose (from 0.98 at 5Gy to 0.85 at 100Gy) was observed. This fact reflects chromatin DNA injuries, with the decrease of double helix DNA proportion. As is known, single and double strand breaks are produced under fast neutron action. The quantity of the remaining DNA double strand, determined by fluorimetric analysis of the unwinding DNA, also decrease with neutron dose.

A reduction of intrinsic fluorescence intensities with neutron dose was also observed (approximately 1.5 times from 5 to 100Gy), denoting a destruction of chromatin proteins' structure. The energy transfer efficiency also decrease under neutron action (from 0.224 at 5Gy to 0.135 at 100Gy), indicating a more unstable tertiary structure of chromatin.

For tumour chromatin, some results are presented in figures 1, 2, 3. So, in figure 1 are represented the relative absorbencies (E/E_0) obtained for the thermal transitions of chromatin samples versus fast neutron dose, lower values indicating a damaged chromatin DNA (with a smaller proportion of double helix).

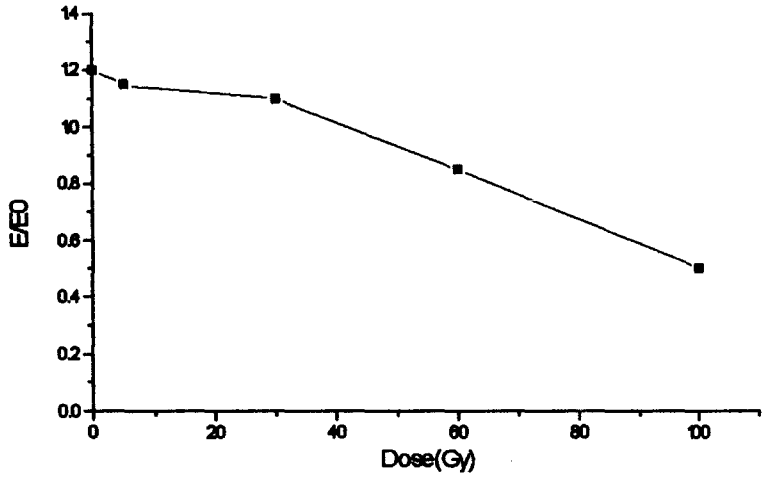

Figure 1Thermal transition of chromatin samples exposed to fast neutrons.

Figure 2 presents the fluorescence intensities of chromatin-ethidium bromide complexes versus fast neutron dose. Lower values also denote modifications in chromatin DNA structure, so that intercalation of ethidium bromide between pairs of superposed DNA bases takes place in a lesser degree.

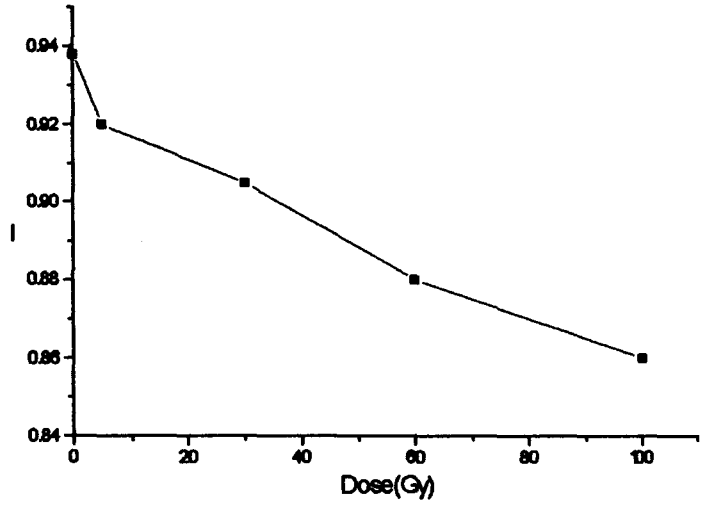

Figure 2. Chromatin-ethidium bromide complexes relative fluorescence versus fast neutron dose.

Intrinsic fluorescence of chromatin tyrosine and of chromatin tryphtofane versus fast neutron dose is presented in figure 3, the diminution of their values indicating a destruction of chromatin proteins.

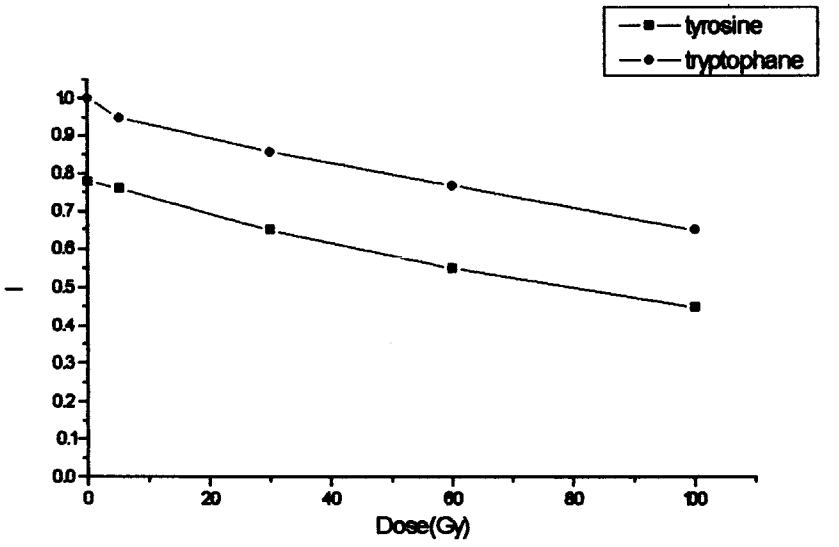

Figure 3. Chromatin tyrosine and tryptophane relative fluorescences versus fast neutron doses.

To study the influence of thyotepa, thyroxine and D_3 vitamin treatments on fast neutron radiolysis in tumour chromatin, 10mg/kg of anticancer drug thyotepa, 40μg/kg of hormonal compound thyroxine and 30,000 UI/kg of D_3 vitamin were administrated separately or associated to Wistar rats bearing Walker carcinosarooma. Similar studies on chromatin modifications as mentioned below were effectued. Preliminary results indicate that the association of thiotepa with thyroxine and D_3 vitamin produce a diminution of chromatin lesions induced by cytostatics and, consequently, differences in chromatin fast neutron radiolysis. Such results could constitute an indication for associated chemotherapy-radiotherapy scheduled in clinical applications. This research is in progress.

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PAPER TITLE Modelling the comparative behaviour of some enzymes at γ- ray and e-beam irradiation

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1 (see page 7)

ABSTRACT (See instructions overleaf) In the study there are consideral some food-interest enzymes, like: proteases, α-amylases, pectic enzymes.
The enzyme preparations were irradiated in dry state and normal conditions(ambient temperature, pressure and humidity); by γ-rays and electron - accelerated beam. The γ - irradiation was performed to a Co - 60 source with approximately 105 TBq activity, in the dose-range of 1 - 50 kGy at different dose-rates between 0.3-1.5 kGy/h. The e - beam irradiation were performed on a liniar accelerator with the next main parameters: electron mean energy 7 MeV; mean beam current 5 μA, pulse perriod 3.5 μS; frequency 100 Hz, at the same doses, with dose-rates between 0.5-1.5 kGy/min

There were investigated, comparatively, the kinetical behaviour of the enzymes in terms of the reaction speed, v; Michaelis - Menten constant K_M and enzyme activity A_e versus irradiation dose.

These is observed a significant influence of the irradiation on the enzyme kinetical parameters, especially for the high doses.

To understand both γ- ray and e-beam effects on the enzymes, their action is described in dose-response terms and equation.

INFLUENCE OF FREE 5-BROMODEOXYURIDINE ON SCE INDUCTION IN HUMAN LYMPHOCYTES IRRADIATED AT THE PRESYNTHETIC STAGE

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INTRODUCTION

Published data on induction of sister chromatid exchange (SCE) in mammalian cells by gamma-irradiation at the G_1 stage contain a contradiction: irradiation at the G_1 of the first replication cycle in the presence of 5-bromodeoxyuridine (BrdU) did not induce SCE (1, 2), while irradiation at the G_1 of the second cycle enhanced its frequency significantly (3, 4). Explanations of this controversy suggested either that chromosomes containing no BrdU were insensitive to gamma-rays, contrary to those BrdU-substituted; or that DNA lesions formed in the first cycle induced isolocus SCE both in the first and in the second cycle, these exchanges annihilating each other. However, an attentive analysis of the published works shows that in all cases BrdU was added to the culture just before the onset of the S stage of the first cell cycle; thus, the cells were exposed in absence of BrdU at the G_1 of the first cycle, and in its presence at the G_1 of the second cycle. In an earlier paper (5) we have shown that this difference in the experimental protocols, which seemed to escape notice in interpretations of the results, was very likely the reason of the above discrepancies. Here we describe the results of experiments directly addressed to this problem.

SCE INDUCTION BY IRRADIATION AT THE G_1 STAGE OF THE FIRST CYCLE

Fig. 1 shows the data on SCE frequencies induced in cultured human lymphocytes at the G_1 stage of the first post stimulation cycle by 2 Gy ^{60}Co gamma-rays for different schedules of BrdU addition to the culture. The cells were irradiated either immediately after PHA-stimulation (early G_1), or after 10 (middle G_1) or 20 h (late G_1) incubation. BrdU (final concentration 9×10^{-5} M) was added either at 23 h after PHA-stimulation (i.e. just before the onset of the S phase), or simultaneously with PHA, or immediately after irradiation. The cells were harvested at 59.5 h after PHA-stimulation. The comparison of the left (grey) columns between four groups displayed in the diagram shows that irradiation in the absence of BrdU did not induce SCE. From the comparison of the dark columns it is evident that, provided BrdU was present in the culture from the very beginning, irradiation enhanced SCE frequency reliably. Moreover, the control frequency was significantly higher in the latter case

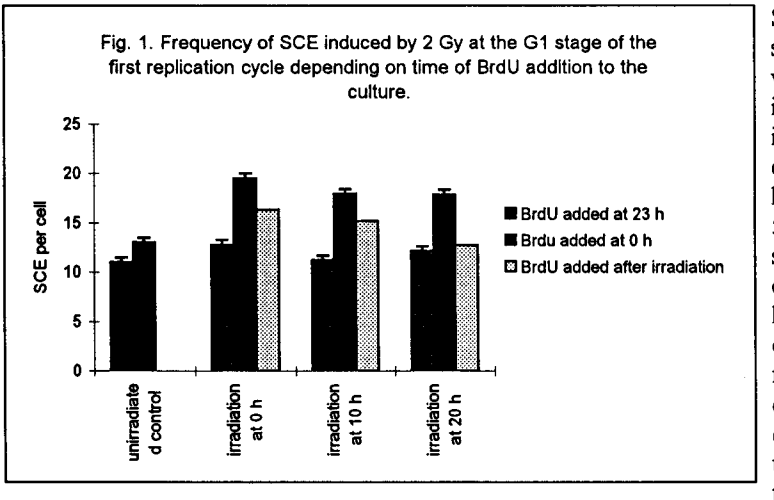

Fig. 1. Frequency of SCE induced by 2 Gy at the G1 stage of the first replication cycle depending on time of BrdU addition to the culture.

than in the former (see the first pair of columns). Comparison between light-coloured columns in three groups and comparison between light and dark columns in each group implies that presence of BrdU in culture medium was essential for SCE formation both at the moment of irradiation and during postradiation incubation of cells.

SCE INDUCTION BY IRRADIATION AT THE G_1 STAGE OF THE SECOND CYCLE

The other set of experiments was aimed to study SCE induction by irradiation at the G_1 stage of the second replication cycle of lymphocytes. BrdU was added to the culture simultaneously with PHA. Three variations of experimental protocol were employed: (1) the cells were irradiated immediately after washing off BrdU; (2) quite the reverse, BrdU was washed off immediately after irradiation; (3) BrdU was washed off 1h after irradiation. It was important to choose the time of BrdU removal so that the bulk of cells would pass only one replication cycle in its presence and the subsequent cycle without it. Under this condition, the time of BrdU removal and that of irradiation being very close to each other, one can be sure that the metaphases with differentially stained sister chromatids, which are to be analysed, are those very cells which were irradiated at the G_1 stage of the second cycle. In preliminary experiments this time was determined as 40 h after PHA-stimulation. Cells were harvested at 62 h after stimulation.

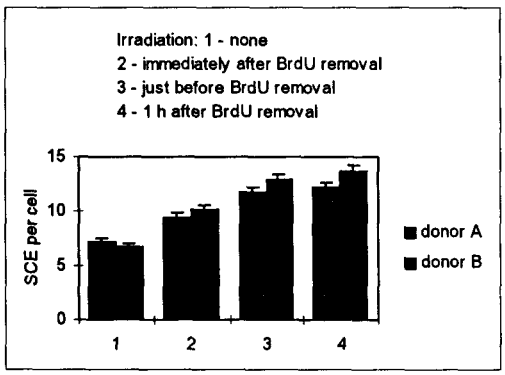

Fig. 2. Frequency of SCE induced by 2 Gy at the G_1 stage of the second cycle depending on time of BrdU removal from the culture.

The results obtained for two donors are illustrated in Fig. 2. Unlike the data for cells irradiated in the first replication cycle, exposure at the G_1 of the second cycle induced SCE even in the absence of BrdU (comparison of the first pair of columns to the second). This may be associated either with the different composition of chromatids (they did not contain BrdU at the G1 of the first cycle, but contained it at the G1 of the second one) or with rather prolonged presence of BrdU in the culture medium before irradiation in the second cycle. However, consideration of all the four groups shown in Fig. 2 evidences that, like in the case of irradiation in the first cycle, the presence of BrdU during the irradiation in the second cycle (and during an hour post irradiation) reliably increased the frequency of radiation induced SCE.

CONCLUSIONS

Irrespective of the number of replication cycle (the first or the second) and of the extent of thymidine substitution with BrdU in chromosomal DNA, irradiation in the presynthetic stage induced significantly more SCE in the presence of BrdU in culture medium than without it. Presence of BrdU was essential for SCE formation both in the moment of irradiation and during postradiation incubation of the cells. It should be underlined that this effect cannot be associated with a direct modification of chromosome radiosensitivity by BrdU, because we are dealing with the presynthetic stage. The influence of free (not incorporated into chromosomes) BrdU can probably be explained by its interplay with mechanisms of repair of initial genetic changes (or lesions). The biochemical possibility of such an interplay had been described (6). Evidences indicating a regular repair process taking place in the G_1 stage were accumulating

over a period of years (see (7) for references) and seem to find a further substantiation recently (8).

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RECOMBINANT PLASMIDS CARRYING MUTANT LOCI *GAM* INCREASE RESISTANCE OF *ESCHERICHIA COLI* WILD TYPE CELLS TO IONIZING RADIATION

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The recessive *gam12* mutation results in constitutive expression of heat-shock regulon as we tested with *lacZ* gene under control of heat-shock promoter p_{htpG}. On the other hand, induction of abnormal protein (human prourokinase) titrating the DnaK increases radioresistance of *E. coli* wild type cells harboring pUK-O2pmO.

Dominant mutation *gam18* compensates defect in *uvrD* gene partly. Random insertions of mini-Tn10-Km^r transposon into pGam18 by hop-transposition from λ_{1105} vector were obtained and plasmid which had lost Gam^r phenotype was selected. Transduction of *kan* marker by Kohara phages revealed homology of plasmid-borne allele with helicases which have 3'→5' polarity. Dominant plasmid of another type pGam43 leads to RecA-independent inhibition of post-irradiation DNA-degradation. Unknown inhibitor is produced constitutively because relative efficiency of plating of T4 2^- mutant phages appeared to be as high as in *recB* strain.

Our data show that enhanced radioresistance of *E. coli* wild type cells harboring recombinant plasmids with mutant loci *gam* is due to unusual involvement of heat-shock proteins in repair and encreased activities of SOS-repair system and specifically RecF-way of recombination-repair.

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ACUTE SKIN LESIONS DUE TO LOCALIZED "HOT PARTICLE" RADIATION EXPOSURES*

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ABSTRACT

The purpose of these studies was to determine the incidence and severity of lesions resulting from very localized deposition of dose to the skin from small (< 0.5 mm), discrete radioactive particles. Hanford mini-swine were exposed to localized doses from 0.2 to over 600 Gy (averaged over 1 cm^2 at 70 μm depth) from isotopes having maximum beta particle energies from about 0.3-3 MeV. The incidence of erythema and scabs (indicating ulceration) were scored routinely for up to 71 days post-irradiation.

The responses followed normal probability distributions, and thus, no true threshold could be defined. Ten and 50% incidence rates were deduced using probit analyses. The lowest dose which produced 10% incidence was about 1 Gy for exposures to Yb-175 (0.5 MeV maximum energy) beta particles. The severity of lesions was estimated by analyzing the results in terms of their diameters and persistence. From preliminary considerations of the probability of induction, size, and persistence of acute lesions, a special limit for hot particle exposures in the range of 5-50 Gy may be reasonable, with an action level between about 1 Gy and the limit.

INTRODUCTION

Exposure to workers to small radioactive particles, typically ranging in size from one micron to a few hundred microns in diameter has been identified as a problem in the nuclear power industry. These particles have become known as "hot particles" due to their relatively high specific activity and small size. When deposited on the skin or inhaled, they can cause intense local irradiation of small areas of tissue and may cause reddening, ulceration, necrosis, and a possible increase in the risk of skin cancer.

In December 1989, the National Council on Radiation Protection and Measurements (NCRP) published a report which reviewed the effects of exposure to hot particles in a human volunteer, in monkeys, and pigs (1). Based on this review, a recommended limit of 10^{10} beta particles (75 μCi h; 10 GBq s) emitted from a "point" particle or a particle of less than 1 mm in diameter was suggested for particles in contact with the skin. This limit was intended to prevent "deep" ulceration but, admittedly, might allow transient effects which frequently disappear in less than a week, as observed by Hopewell et al. (2). For beta particles from activated fuel, this exposure causes a dose of about 5 Sv averaged over 1 cm^2 at a depth of 70 μm in tissue. The risk of skin cancer following irradiations of the skin by hot particles was deemed less than that when extended areas of the skin are irradiated due to the small number of cells exposed and the greater potential for cell killing from the localized beta particle dose.

In 1987, an International Commission on Radiological Protection (ICRP) Task Group reviewed much of the same material reviewed by the NCRP. However, the Task Group based its recommendations on preventing acute transient ulceration and recommended restricting to 1 Sv the dose delivered within a few hours over an area of 1 cm^2 measured between depths of 100-150 μm (10-15 mg/cm^2) (3). The latest recommendations of the ICRP, nevertheless, retain the limit of 500 mSv averaged over any 1 cm^2, at a nominal depth of 7 mg/cm^2, regardless of the area exposed (4). The objectives of this work were to provide additional scientific data to use in the decisions and recommendations of these committees and the regulatory agencies.

MATERIALS AND METHODS

Hanford mini-swine were exposed to four types of particles at Brookhaven National Laboratory (BNL): 1) activated UC_2 particles of 250 μm diameter for comparisons with the data of Forbes and Mikhail used in NCRP Report 106 (1); 2) Tm-170 particles with dimensions of about 230-500 μm for comparisons with Hopewell et al's data used extensively by the ICRP Task Group (3); 3) Yb-175 particles with dimensions of 280-500 μm and maximum beta particle energy of 0.5 MeV (to simulate Co-58 which emits a 0.47 MeV positron); and 4) Sc-46 with beta and gamma emissions very similar to those of Co-60. Cobalt-60, Co-58, and activated fuel are the commonest sources of hot particles in nuclear power plants. Scandium-46 was used rather than Co-60 because its shorter half-life and lower density allowed the production of particles that simulated 100-μm-diameter Co-60 particles with much shorter activation periods. UC_2 particles were 252-μm-diameter spheres coated with 18.5 μm of pyrolytic graphite. Other particles were cut from foils of thickness 258 μm (Tm), or 130 μm (Yb and Sc). The largest dimension of the latter pieces was 535 μm.

*This work was supported by the U.S. Nuclear Regulatory Commission. C. Schaefer was supported by the Health Physics Fellowship Program, administered by Oak Ridge Institute for Science and Education for the U.S. Department of Energy.

Irradiations were made within tattooed grids on the flanks of anestisized pigs. Sources were mounted on styrofoam blocks to minimize backscatter. Exposures of 45 s or less were made using a hand-held cone/rod jig with the source fastened to the end of a styrofoam cone, 8 cm long, attached to a plexiglass rod 22 cm in length. Longer exposures were made using a cloth harness with elastic straps which flexed as the animal breathed, yet held the source in a fixed position for up to two hours.

UC_2 and Tm-170 particles were employed both touching the skin and slightly above the skin to test the hypothesis that more beta particles might be required to induce ulcers in the latter case which minimizes particles on clothing. Animals ranging in age from 3.8 to 15 months were exposed to see if age at exposure had any effect on the outcome. Various scoring procedures were explored to test the sensitivity of results to the number of observations per week, the number of number of observers, and the number of days of observation post-exposure. The sensitivity of results to background corrections also was studied. In addition, the diameter of any lesion and area of erythema was measured, using an overlaid scale, and their persistence recorded.

The following were the scoring procedures for erythema and scabs:

Method A, which detected the most lesions, is a modified Hopewell et al. (2) method in which the mini-swine were scored twice per week for 28 days using two or sometimes three observers (Hopewell, personal communication). To model this scoring criterion, we eliminated data from one of the three weekly observations for the first 21 days post-irradiation. Thereafter, BNL observers made only two observations per week, which was similar to the Hopewell scoring schedule. The BNL lesion-scoring criteria were as follows. If two observers scored an area as a lesion on the same day, or if two observers scored the area as a lesion on one day and one observer scored it as a lesion on the successive day, or visa-versa, the area was scored as one-half of a lesion. If two observers scored the area as a lesion on two consecutive days, then it was considered to be a full lesion.

Method B, or "4-of-6/4-of-4," gave fewer lesions than Method A. For the first three weeks in which the mini-swine were scored three times per week, a lesion was considered to be radiation-induced if two independent observers each scored an area as a lesion a minimum of twice in one week. This method was called 4-out-of-6, since there were two observers and three observation periods, giving six possible outcomes, and four scorings of a lesion were required out of the six possibilities, with each observer seeing the lesion at least twice. After the first 21 days, the pigs were scored twice per week, and a lesion was considered to be radiation-induced if two observers each scored a lesion on two consecutive observation days--which was called 4-out-of-4.

Method C, or "6-of-8/5-of-6," was the most restrictive scoring technique used, and was designed to minimize any spurious apparent background. A lesion was considered radiation-induced if two observers scored an area as a lesion three times each out of a possible four consecutive scoring periods for the first three weeks. After three weeks, when the pigs were only scored twice per week, and a lesion was considered radiation-induced if one observer scored an area as a lesion a minimum of two times and another observer scored the same area as a lesion a minimum of three times in three consecutive scoring periods.

RESULTS

Analysis of the results as a function of the animal's age at exposure revealed no single pattern of changes with age. There was greater variability in the results for specific doses in animals over about 10 months of age, however, individual differences in response were as large as those due to age.

Biopsy samples of normal skin were taken from animals of ages 3.8 months to 15 months. The thickness of the keratin layer ranged from 18-33 μm at ~3 months and increased to 48 μm in pigs aged 15 months. The depth of the basal layer varied from 56-92 μm for these young animals and was 100 μm for a single sample at 15 months of age.

The period of observation (28, 48, or 71 da) was very important. Deduced ED_{50} values (50% incidence of a barely detectable lesion or erythema) were 1.5 to 4 times higher for observations that continued for only 28 days compared to those continued for >71 days. The method of scoring (A, B, or C) typically caused differences of factors of about two, with Method C generally giving higher ED_{50} results; this is attributed to better discrimination against background, or the false positive results obtained using scoring Method C. Background effects (scabs in unexposed areas) were typically about 3%, but varied from 0-14% with large statistical uncertainties due to the small number scored (e.g., 1 of 28 spots). Correcting for background events typically caused 5-10% changes in ED_{50}, and 10-20% changes in ED_{10} results using scoring Method C.

Extrapolation of scab diameter results to the zero diameter intercept to arrive at an apparent threshold yielded results for UC_2 exposures of 4.4-9.1 Gy for exposures to particles on the skin. This compares to about 15 Gy for the related Forbes/Mikhail results. These results are approximately equal to the ED_{10} value of 8.5 Gy derived from the UC_2 incidence results, summarized below.

Table 1 summarizes the percentage scab incidence for the four particles studied here and compares them to the work of Hopewell (5) and Reece et al. (6). The most effective particle in our studies was Yb-175 with maximum beta particle emission of 0.47 MeV that yielded an ED_{10} value of 1.3 Gy (0.36 - 2.6 Gy at 95% confidence). Approximately equal ED_{50} values were obtained for Yb-175 and Tm-170 (0.97 MeV), 5.5 and 5.9 Gy, respectively. These values also are close to the value derived from Hopewell's studies (4.5 Gy) using Tm-170. Both Sc-46 and activated UC_2 had higher ED_{50} values, 12 Gy and 11 Gy, respectively. The ED_{50} values reported by Reece et al. (6) for Co-60,

Table 1. Comparison of the Findings of Hopewell et al., BNL, and EPRI*

Isotope	E_{max} (MeV)	E_{avg} (MeV)	Hopewell		BNL		Reece
			ED_{10} (Gy)	ED_{50} (Gy)	ED_{10} (Gy)	ED_{50}(Gy)	ED_{50} (Gy)
Pm-147	0.22	0.062	2.7	3.9			
Co-60	0.31	0.096					40
Sc-46	0.36	0.11			5.1	12	
Yb-175	0.47	0 12			1.3	5.5	
Tm-170	0.97	0.29	1.9	4.5	2.8	5.9	18
Sr/Y-90	1.4**	0.55	3.4	8.4			
U (act.)	1.8 (1.1)**	0.71 (0.4)			8.5	11.4	40

* Dose averaged over 1 cm^2 at 70 μm depth (Gy).

**Average of the maximum energies emitted by the isotopes in the source. Values in parentheses are for Reece's sources.

Tm-170, and activated U particles were 3-4 times higher than the corresponding results for Sc-46, Tm-170, and UC_2 from our studies and those of Hopewell et al. (5).

DISCUSSION AND CONCLUSION

The lesions being scored in this type of study at low doses are small (e.g., about 1- to 2-mm diameter), and transient (e.g., 1-2 weeks). These lesions are not open festering wounds, but rather, barely noticeable scabs. The question to answer is how should this detriment be weighted in comparison with the debilitating effects normally being avoided in setting limits for deterministic and stochastic risks.

In these studies, over 1,100 individual spots were exposed to hot particles. About 560 of these developed detectable scabs which were assumed to result from a break in the integrity of the skin, and therefore, may lead to infection. Only two of the exposed sites became infected -- these after exposures near 500 Gy. One was treated topically with ointment for a few days and the other was treated topically and systemically with an antibiotic since the lymph nodes seemed to be infected. These large exposures leading to infection certainly are not acceptable detriments and a limit should be set to avoid them. However, the low incidence of infection for animals living under typical experimental conditions, and lack of other serious effects, suggests that a limit set to avoid persistent ulceration may be acceptable even if small transient scabs are produced.

A dose limit could be based on the product of the probability of scab formation, the diameter of the scabs and their persistence. From a preliminary evaluation of our data, for doses in the range of 1-200 Gy, this value is about 1-4 scab-mm-da/Gy. Alternately, a limit could be based on the number of days a scab is expected, regardless of its diameter. For example, preliminary analysis in terms of the number of scab-days for exposures between 0.1 and 50 Gy indicates that an average value for all sources is about 1.6 scab-days/Gy with somewhat higher values at lower doses and perhaps twice as many scab-days per Gy from Tm-170 compared with other sources.

These results together suggest that a limit in the range of 5-50 Gy may be appropriate to avoid effects that are more nearly comparable with other detriments at the dose limits. Based on these results and the relatively small harm associated with the production of lesions by hot particles, it seems reasonable to set a special hot particle limit in the range of 5-50 Gy (dose averaged over 1 cm^2 at 70-μm depth in tissue). In addition, an action level for persons receiving doses in the range 1 Gy to the limit selected could be established involving weekly checks for possible lesions; any ones detected then could be treated to avoid infection.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19, 1996
Vienna, Austria

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PAPER TITLE ANALYSIS OF TREATMENT POTENTIALITIES OF SKIN RADIATION LESIONS BY THE AID OF HYPERBARIC OXIGENATION

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MAJOR SCIENTIFIC TOPIC NUMBER

The experiment was performed on not pedigree male rats weighing 220-280 grams. A section of the animal's back measuring 4.0 x 4.0 cm was exposed to radiation dose of 5000 and 10000 rad on the RUM-140 device. Hairs on that section were removed before the irradiation. Radiation burn developed in 10-14 days. This burn closed up in 3 months and after 4-6 months complete recover of hairs was observed.

Regimens of hyperbaric oxigenation /GBO/ (1 and 3 ATMs hour daily during 8-10 weeks) showed no harmful influence on animals behavior. These GBO regimens promoted more auspicious wound process and more rapid healing of the skin.

Application of GBO straight away irradiation before the development of visible manifestations of radiation lesions was conductive to tardy development of radiation wound and more mild course of wound process. Thus obtained results testify that GBO have prospects in the treatment of radiation skin lesions.

EARLY EFFECTS OF RADON PROGENY EXPOSURE IN ANIMALS

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INTRODUCTION

Life-span carcinogenicity studies are currently being conducted in rats at AEA Harwell to investigate the effects of exposure conditions on lung tumour induction following exposure to radon and radon progeny. Chronic exposures to radon/radon progeny are performed under well defined conditions, in a specially constructed exposure facility (1). Groups of animals have been exposed to levels between 175 and 3000 WLM, at an exposure concentration of 1000 WL. Further studies will investigate the effect of dose-rate on lung tumour incidence at both high and low total doses. Life-span carcinogenicity studies in the rat are expensive and time consuming. In addition, determination of the radiation dose to the lungs is not simple. Various approaches to reduce the time-scale and assess the radiation dose to the lung are being investigated. In France the CEA group have used 5-6-Benzoflavone to increase the incidence of preneoplastic changes and tumours and to reduce the latency period for tumour induction following exposure to radon and radon progeny (2, 3). At AEA, studies on the early effects of radon progeny exposure, namely cell proliferation and induction of chromosome aberrations are being conducted to determine whether these early effects can provide predictions for lung tumour induction. This approach may allow the effects of different exposure conditions to be determined far more rapidly and may also give some insight into the mechanisms of lung tumour induction by radon progeny.

METHODS

For these studies, epithelial cell proliferation and induction of nuclear aberrations in alveolar macrophages were studied in male Sprague Dawley rats 14 days following exposure to radon and radon progeny. Earlier studies (4) indicated that the peak response for these effects is observed at this time, regardless of exposure duration (2-15 days).

Following exposure (duration 1-20 days), animals were housed under standard conditions with food and water available *ad libitum*. At 14 days after the end of exposure, the animals received an intraperitoneal injection of 5-bromo-2'-deoxyuridine (BrdU, Sigma) (4 ml, 5 mg/ml). Four hours later they were killed with an intraperitoneal injection (400 mg/kg) of sodium pentabarbitone (Sagatal, RBM, Animal Health Ltd, Dagenham) followed by exsanguination via the abdominal aorta.

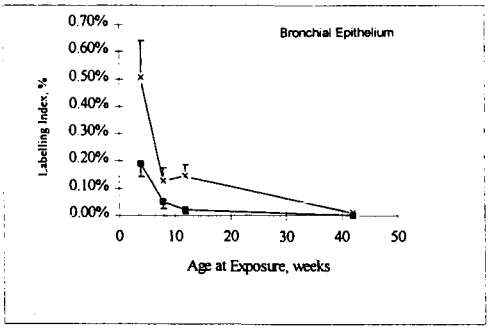
Where measurements of nuclear aberrations were to be made, the lungs were lavaged and cytocentrifuge slides prepared from the lavage fluid (4). Differential cell counts on at least 2000 cells selected at random were made on the these slides. Nuclear aberrations in alveolar macrophages were scored as cells containing micronuclei, binucleated cells, cells containing both types of aberration and cells containing multiple fragments of nuclear material in addition to the main nucleus. Identification of micronuclei was as described previously (4).

For the assessment of epithelial cell proliferation, following lavage, the lungs were fixed, sectioned and serial sections stained for BrdU and with Hand E as described previously (4). For each animal at least 20 fields on the BrdU stained slides were observed and the number of BrdU labelled cells counted. At least 10 fields on the H and E stained slides were observed and the total number of epithelial cells counted. The labelling index was expressed as the percentage of total epithelial cells incorporating BrdU.

EFFECT OF AGE ON RADON INDUCED CELL PROLIFERATION

Before beginning the main life-span studies, the effect of animal age on early effects resulting from radon/radon progeny exposure was investigated. Other groups studying the effect of radon progeny exposure have used animals of widely varying ages. The TNO group in the Netherlands have used animals as young as 4 weeks at exposure (5), whereas the CEA group in France have generally used animals which were 3 months old (6). Groups of 4 animals which were 4, 8, 12 and 40 weeks old were simultaneously exposed in the AEA facility to a total dose of 440 WLM, over 5 days. Groups of similar aged animals were maintained under normal (non-exposed conditions) to act as controls. For each animal, cellular proliferation in the alveolar and bronchial epithelium was determined using BrdU uptake at 14 days post exposure (Figure 1). The youngest animals (both exposed and control) showed the highest proliferation rates (at least 10 fold higher than the adult animals). In both the exposed and control animals, the elevation of proliferation above that in adults, fell with age, and by 12 weeks of age at exposure, proliferation levels were as for the 40 week old animals. Generally, for all age groups and both exposed and control animals, epithelial cell proliferation was low in the bronchial region (less than 0.5% even in the youngest radon exposed animals) and about 3 times higher in the alveolar region. Taya et al (4) found that the bronchial region showed a similar extent of proliferation to that in the alveolar epithelium. Exposure to radon/ radon progeny resulted in an increase in proliferation of between 2 and 6 fold in all age groups. The study demonstrated that the use of animals less than 12

weeks of age may not be appropriate for dosimetric studies and raised doubts about their suitability for life-span studies as the higher proliferation rates could result in higher cell death and lower tumour incidences.



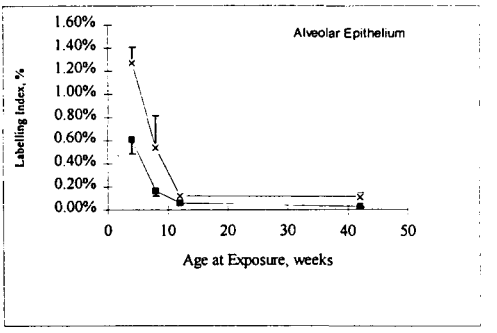
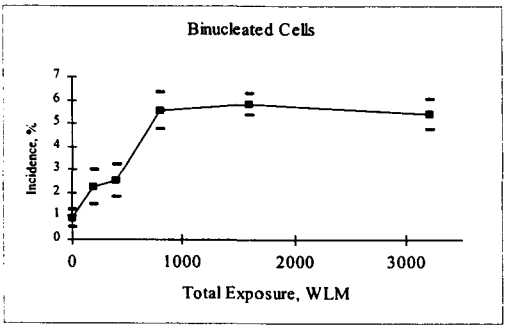
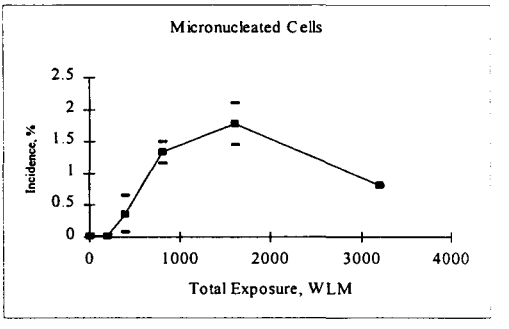


Figure 1. The effect of animal age on epithelial lung cell proliferation, following and without exposure to radon progeny. (x-radon exposed ■-unexposedMean incidence of proliferating cells % ± SEM).

THE EFFECT OF DOSE ON CELL PROLIFERATION AND NUCLEAR ABERRATION INDUCTION

A study to investigate the effect of dose on lung tumour induction is currently being conducted. Fourteen groups of animals were exposed at a constant dose rate (1000 WL) over varying time periods to give total exposures of 200 to 3200 WLM. From each exposure group, 2 animals were killed 14 days after exposure to assess early effects (epithelial proliferation and nuclear aberration induction). The incidences of the three principal types of nuclear aberration are given in Figure 2.





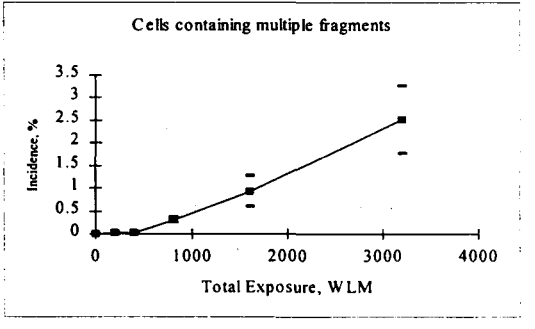


Figure 2. Incidence of nuclear aberrations in alveolar macrophages following exposure of rats to radon progeny at different cumulative doses (Mean incidence % ± SEM).

Cell proliferation was measured in the alveolar and bronchial regions from each of the animals (Figure 3). Alveolar epithelial cell proliferation rose with increasing dose up to 800 WLM. It then fell to levels below that

of control animals at doses of 3200 WLM. In the bronchial region, proliferation peaked between 200 and 400 WLM and also fell to levels below those of control animals at the higher doses.

Early effects (various types of nuclear aberration and cell proliferation) generally showed a dose response with increasing radon/radon progeny exposure levels. For all effects other than multiple micronucleus induction (possibly representative of apoptosis), beyond a peak response dose, the dose response relationship was not maintained and effects either fell at higher doses or remained level. The dose at which peak response was observed was different for different effects. The most sensitive effect was epithelial cell proliferation, with peak responses in the alveolar and bronchial regions occurring at 800 WLM and 200-400 WLM respectively. Aberration induction appeared to be less sensitive and peak responses were observed at 800 and 1500 WLM for binucleus and micronucleus formation.

CONCLUSIONS

The validity of these early effects to accurately predict long term lung cancer incidences will not be known until the current life-span studies are completed. However should one or a combination of the assays used prove to be successful in predicting long term effects from radon exposure, the duration and costs of studies to investigate the effect of exposure conditions on the induction of lung tumours by radon progeny could be significantly reduced. In all future exposures for life-span studies at AEA, animals are being taken to study these early effects.

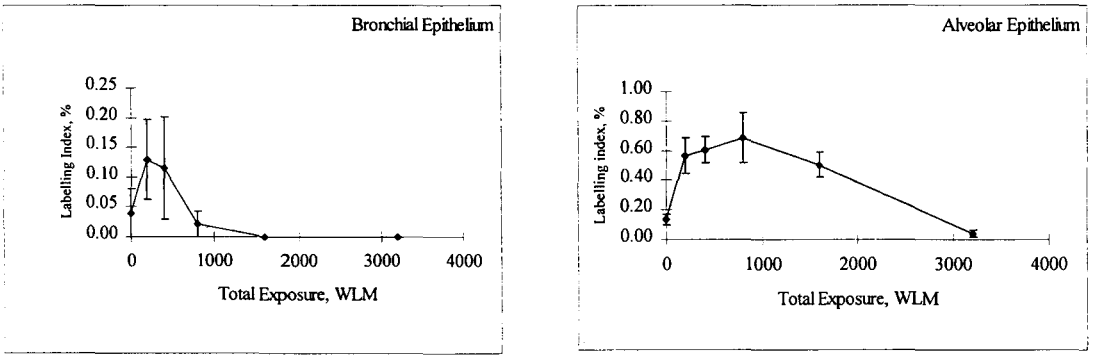


Figure 3. Epithelial lung cell proliferation following exposure of rats to radon progeny at different cummulative doses. (Mean incidence of proliferating cells % ± SEM).

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EFFECT OF RADON-222 AND ITS DAUGHTER INHALATION ON REPRODUCTION IN RATS EXPOSED UNDER NATURAL CONDITIONS

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INTRODUCTION

Radon-222 and its short-lived progeny inhalation is strongly related with a high incidence of lung cancer /1/. The literature did not reveale any other effect produced by Radon-222 and its decay products /2/.
The aim of the study is to present data about sterility induction in rats exposed under natural conditions ia a uranium mine to high level of Radon-222 and Radon-222 progeny concentrations in air.

MATERIAL AND METHOD

The experiments were performed on 60 adult Wistar rats. The animals were devided into 6 groups:

- grups I an II (female and male) exposed for 52 days in a uranium mine to an average concentration of 28,638 Bq/c.m. Radon-222 and 2.76 Working Level of Radon-222 progeny;
- control groups III and IV (female and male) kept underground in the same mine, in the same conditions of temperature and humidity as groups I and II; Radon-222 and Radon-222 progeny concentration were at the level natural background;
- control groups V and VI (female and male) kept in vivarium conditions.

The Kusnetz' and Rock' methods were used to measure Radon-222 and its daughters concentrations in air /3, 4/. The externalgamma dose rate was also measured with a Curiemeter VA-J-100.
All groups were devided into two subgroups; the fertility index was calculated at the end of the expoosure and ten months later, by mating unexposed rats with exposed ones. The weight of the reproductive organs were evaluated and the histological sactions were analysed.

RESULTS

Immediately after exposure and mating of exposed females with unexposed males and unexposed females with expsed females the fertility indices were 82% and 85%, respectively; the histological sections of reproductive organs presented a normal aspect.

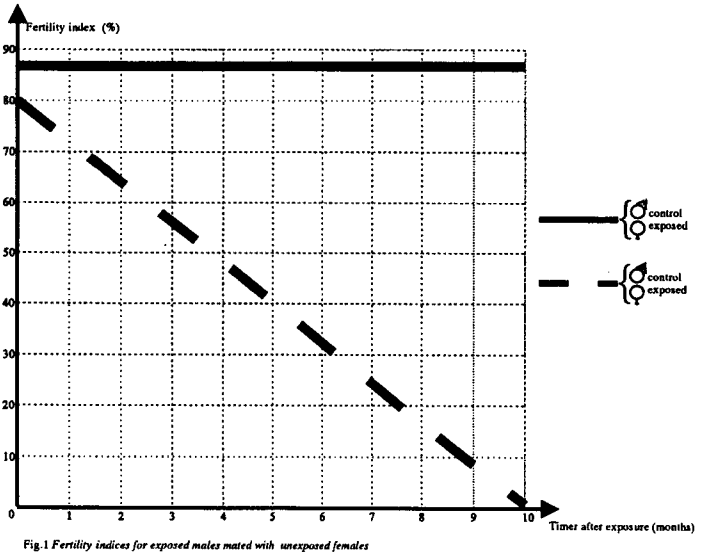


Fig.1 *Fertility indices for exposed males mated with unexposed females*

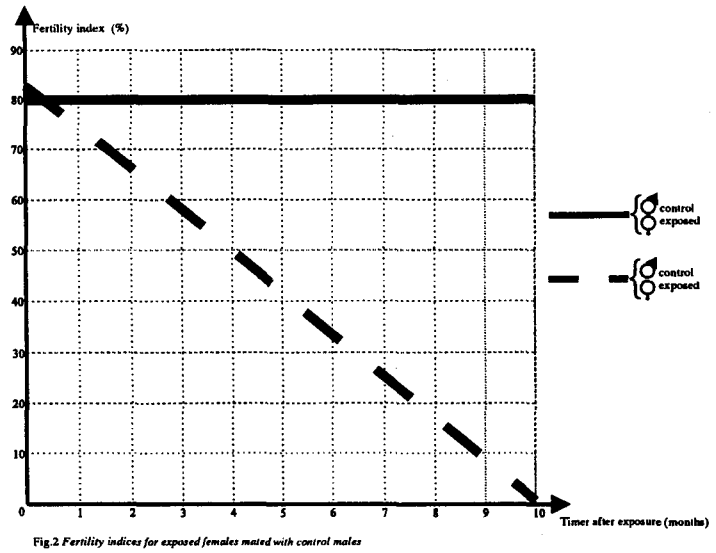


Fig.2 *Fertility indices for exposed females mated with control males*

After a latency period of 10 months the animals presented severe lesions of the reproductive organs manifested by cellular and follicular distrophy at level of the ovaries and atrophy of the germinative epithelium of testis (Fig. 3). Concomitantly to these morphological lesions the fertility index became crossed mating of the exposed and control animals (Fig. 1 and 2). There were no diferesces between the fertility indices of exposed animals mated with underground control groups and vivarium control groups, respectively.

Fig. 3 *The histological section of testis 10 months after exposure*

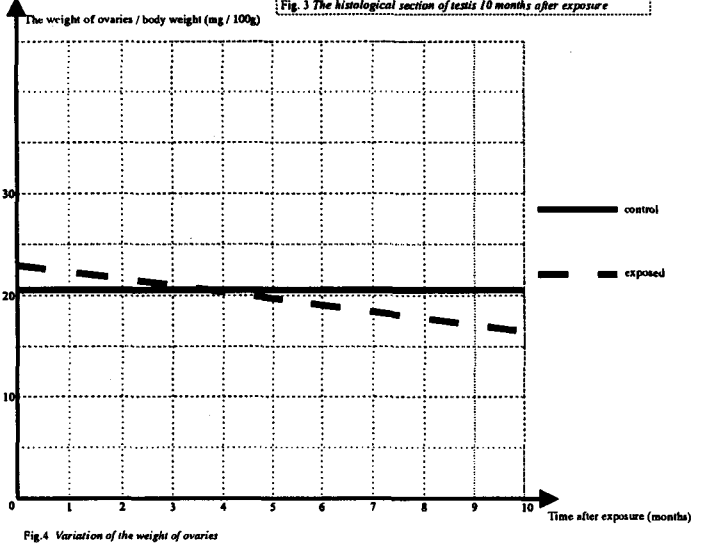


Fig.4 *Variation of the weight of ovaries*

The weight of ovaries for exposed and unexposed rats did not present significant variations in time (Fig. 4 and 5). The weight of testis / 100g body weight was significantly lower ten months after the end of exposure (160 mg/100g. body weight) compared with control underground group IV (476 mg/100g. body weight). (Fig. 6)

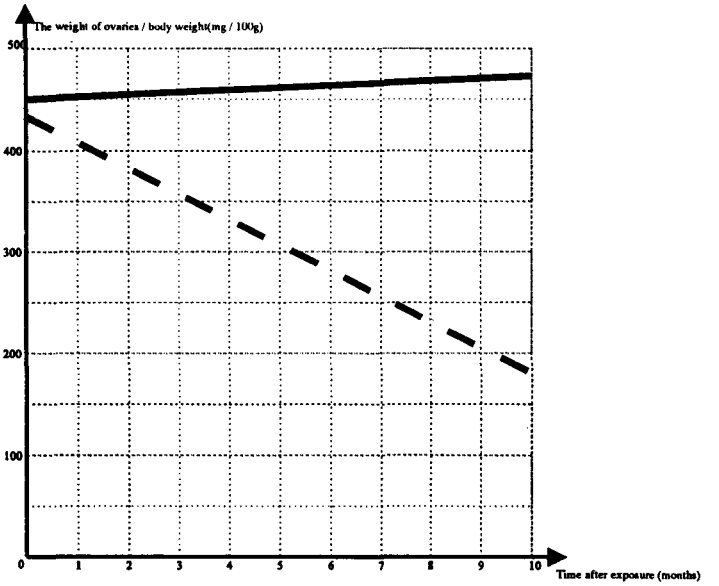
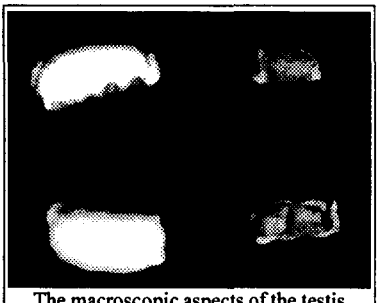


Fig.5 Variation of the weight of ovaries



The macroscopic aspects of the testis 10 months after the exposure compared with control groups

CONCLUSION

As the gamma dose values were too low to produce the rats sterility, the only possible explanation is the inhalation of short - lived Radon-222 progeny.

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Osteosarcoma induction in mice by the alpha-emitting nuclides, plutonium-239, americium-241 and uranium-233

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Introduction

Following either occupational or environmental exposure of individuals to alpha-emitting isotopes of plutonium (Pu), americium (Am) or uranium (U), their entry into the bloodstream will result in long-term retention in the skeleton and a consequent risk of osteosarcoma and leukaemia. The International Commission on Radiological Protection (ICRP) has recently adopted models for the behaviour of radionuclides in the skeleton which take account of the physiological processes of bone remodelling and growth (1,2). The models treat Pu and Am similarly, allowing for burial of surface deposits during mineral deposition and transfer to the marrow and release to the circulation during bone resorption (1). Thus, the models predict that ^{239}Pu and ^{241}Am are similarly effective in inducing osteosarcoma and leukaemia. Uranium behaves more like calcium and the alkaline earth elements and the model assumes more rapid transfer from surfaces to bone volume (2). Because a greater proportion of alpha energy from U isotopes is deposited harmlessly in bone mineral, they are predicted to be less effective in causing malignancy. No changes have been made to the assumptions that the sensitive cells for osteosarcoma induction are uniformly distributed on endosteal bone surfaces and those for leukaemia induction are uniformly distributed through red bone marrow. Furthermore, radionuclides are assumed to initially deposit uniformly on endosteal bone surfaces.

A number of finding are at variance with model predictions. Experimental studies using rats (3) and mice (4) have shown that ^{239}Pu is about three times as effective as ^{241}Am in inducing osteosarcoma per unit average skeletal dose. Autoradiographic studies of mouse (5) and rat (6) bone have shown differences in the initial distribution of ^{239}Pu, ^{241}Am and ^{233}U as well as their subsequent behaviour. The induction of osteosarcoma but absence or very low incidence of leukaemias in humans given radium isotopes suggest that alpha irradiation from bone surfaces may be ineffective in causing leukaemia, possibly due to heterogeneity in the distribution of sensitive cells in the marrow (7).

This paper presents results from a study of the comparative toxicity of ^{239}Pu, ^{241}Am and ^{233}U in mice, comparing tumour incidence and the distribution of alpha dose within the skeleton. Completed results for osteosarcoma induction are reported; leukaemia diagnoses are not yet complete.

Materials and Methods

Male CBA/H mice were obtained from the Medical Research Council (MRC) Radiobiology Unit, Chilton, Didcot, Oxon, OX11 0RD. This strain a very low spontaneous incidence of osteosarcoma and leukaemia. Groups of 5 mice were housed together. Food (type RMI Expanded, Special Diet Services, Witham, Essex) and water were freely available at all times. Animal care and handling conformed to the UK Animals (Scientific Procedures) Act 1986.

Groups of 12 week old mice (50-100) were given intraperitoneal injections of ^{239}Pu, ^{241}Am or ^{233}U as the citrate complexes. Each nuclide was administered to three groups of animals at levels of activity estimated to give average bone doses of about 0.2-0.3 Gy, 0.5-1.0 Gy and 1.3-1.6 Gy, calculated to 500 days after administration; a further control group (100) were given inactive solution. To avoid chemical toxicity to the kidneys from ^{233}U, each nuclide was administered as 9 injections over a 3 week period. All groups were treated similarly. The mice were followed for their life-span and either died or were killed when moribund or a visible tumour was present. Osteosarcomas were identified by x-ray examination of the skeleton at death and the tumours confirmed by histological examination. Other organs and tissues were taken for histological examination as appropriate.

Results

Table 1 summarises tumour diagnoses for each group of animals, counting all observations including cases where individual animals were shown to have more than one tumour or tumour type. High incidences of liver and lung tumours were observed for all experimental groups and the control group, as expected with this mouse strain (8,9). A variety of other soft tissue tumours were identified with low incidences (see footnotes to Table 1). Small numbers of blood cell tumours were also observed; final diagnoses are in progress.

Table 1. Number of tumours by type

Group	^{239}Pu			^{241}Am			^{233}U			Control
Average skeletal dose to 500 days (Gy)	0.2	0.5	1.3	0.3	0.9	1.6	0.3	0.9	1.3	0
No of animals	100	60	50	100	75	50	100	60	50	100
Osteosarcoma	2	5	14	0	3	10	2	1	2	1
Other skeletal tumours[a]	0	1	5	3	4	0	3	0	0	6
Leukaemias / lymphomas[b]	5	6	1	6	0	0	2	1	1	2
Other haemopoietic disorders[c]	2	1	1	3	4	1	4	2	3	5
Liver tumours[d]	94	76	48	86	69	43	83	66	42	80
Lung tumours[e]	40	23	15	36	24	10	40	19	16	37
Other soft tissue tumours[f]	10	12	10	8	11	8	19	18	30	19

[a] skeletal haemangioma, haemangiosarcoma, osteoma, odontoma, sarcoma, mastocytoma ; [b] myeloid leukaemia, lymphatic leukaemia, malignant lymphoma; [c] histiocytic sarcoma, haemangiosarcoma, haemangioma; [d] hepatocellular adenoma and carcinoma; [e] bronchiolo-alveolar adenoma and carcinoma; [f] islet adenoma and carcinoma, tubular adenoma and carcinoma, adrenal cortical adenoma, adrenal phaeochromocytoma, neuroendocrine carcinoma, meningeal sarcoma, malignant paraganglioma, malignant schwannoma, rhabdomyosarcoma, sarcoma, leiomyoma, trichoepithelioma and squamous cell carcinoma (skin)

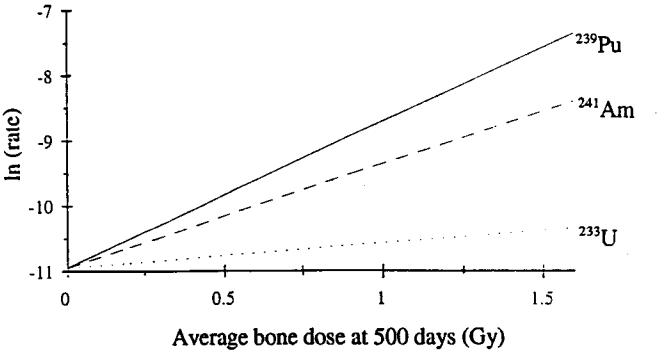

Figure 1. Incidence of osteosarcoma following administration of ^{239}Pu, ^{241}Am or ^{233}U

Osteosarcoma was seen in 2%, 8% and 18% of mice in the 0.2 Gy, 0.5 Gy and 1.3 Gy ^{239}Pu groups (5, 15 and 25 kBq kg^{-1}), respectively. The corresponding incidences of osteosarcoma in ^{241}Am groups were 0, 4% and 18%, respectively, and incidence in each ^{233}U group was about 2%. Only the highest dose groups had animals with more than one osteosarcoma, with one animal having two tumours in each nuclide group and 2 animals having three tumours in the ^{239}Pu group. A difference was apparent in the location of osteosarcomas in the highest dose ^{239}Pu and ^{241}Am groups in that about 70% in the ^{239}Pu group were in the appendicular skeleton, mainly femur, while a similar proportion in the ^{241}Am group were in the axial skeleton, mainly vertebrae. Osteosarcoma incidence as a function of dose was assessed using Poisson regression (Figure 1). Significance was tested by comparing changes in deviance with a χ^2 distribution. Dose was found to have a significant effect on incidence (change in deviance = 20.34 on 1

$p<<0.001$). However, the effect differed significantly between nuclides (change in deviance = 16.21 on 2 d.f.; $p<0.001$). There was evidence for a difference in regression slopes for ^{239}Pu and ^{241}Am (change in deviance = 3.86 on 1 d.f.; $p=0.05$) and a single slope for ^{239}Pu and ^{241}Am differed significantly from a slope for ^{233}U (change in deviance = 12.35 on 1 d.f.; $p<0.001$). For ^{233}U, the slope was consistent with no effect of dose. A further analysis will take account of ages at death and competing causes.

Discussion

The observed incidences of ^{239}Pu induced osteosarcoma in male CBA/H mice are consistent with reported incidences in females of the same strain of 7.5% and 30% after administration of 6 kBq kg^{-1} and 19 kBq kg^{-1} of ^{239}Pu, respectively (8), and the observation that females were more sensitive than males to osteosarcoma induction by a factor of 3 - 4 for ^{239}Pu and 4 - 6 for ^{226}Ra (9). It is considered that the difference may be due to greater turnover of bone in females. The observed difference in osteosarcoma induction by ^{239}Pu and ^{241}Am was smaller than reported previously. Bensted et al (4) reported that 17 out of 22 male rats given 108 kBq kg^{-1} ^{239}Pu developed osteosarcoma compared with 4 out of 19 given 92 kBq kg^{-1} ^{241}Am. Using small groups of animals (10 - 20), Taylor et al (3) showed a significantly greater incidence of osteosarcoma in male and female C57Bl/Do black mice given 105 kBq kg^{-1} and 32 kBq kg^{-1} of ^{239}Pu than in comparable groups given 102 kBq kg^{-1} and 30-34 kBq kg^{-1} of ^{241}Am. Similar studies using C57BL/Do albino mice showed greater osteosarcoma incidence with ^{239}Pu than ^{241}Am in female mice but no difference in male mice.

Autoradiographic studies have shown clear differences in the initial distribution of ^{239}Pu and ^{241}Am on bone surfaces in rodents (5,6). While ^{239}Pu is concentrated mainly on endosteal surfaces of trabecular and cortical bone, ^{241}Am is more widely distributed on bone surfaces including periosteal surfaces and internal cortical bone surfaces. Previous reports of greater sensitivity to osteosarcoma induction by ^{239}Pu than ^{241}Am have been attributed to greater doses from ^{239}Pu to trabecular surfaces (10). It is possible that differences in distribution and osteosarcoma incidence were accentuated by increased bone turnover caused by the deposited nuclides which is apparent, according to Loutit et al (9), at administered activities of 20 kBq kg^{-1} and greater.

Uranium has been shown to deposit initially on bone surfaces, including endosteal and periosteal surfaces, concentrating on growing surfaces (5,6). The low toxicity of ^{233}U compared with ^{239}Pu and ^{241}Am is consistent with more rapid burial of surface deposits, reflecting the chemical similarity of the uranyl ion and those of the alkaline earth elements, including calcium.

In conclusion, it would appear that the observed osteosarcoma incidences, although showing small differences between ^{239}Pu and ^{241}Am, are broadly consistent with ICRP model estimates of equivalent toxicity of ^{239}Pu and ^{241}Am and lower toxicity of ^{233}U.

Acknowledgements

Part funding for this study was provided by the Commission of the European Communities. The assistance of KR Isaacs in pathological diagnoses is gratefully acknowledged.

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RADIATION-INDUCED CHROMOSOME ABERRATIONS IN BONE MARROW CELLS LEADING TO ACUTE MYELOID LEUKEMIA IN MOUSE

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INTRODUCTION

It is well known that radiation-induced acute myeloid leukemia (RI-AML) in mice is characterized by deletion and/or rearrangement of chromosome 2 (1-5). While chromosome 2 has been suspected to be a target of RI-AML, radiation-sensitive site of the chromosome might be implicated in the leukemogenesis (3,6). There were few cytogenetical studies, however, focusing on chromosomal rearrangements shortly after irradiation, and little was known about the frequency and pattern of chromosome 2 aberrations during the early period. In this study, metaphase samples were prepared from whole-body irradiated mice 24 hours after irradiation, most of the cells considered to be in the first mitotic stage. Distribution of chromosomal breakpoints on the metaphase samples were analyzed to study the relationship between chromosome aberrations and RI-AML.

MATERIALS AND METHODS

Eight-week male C3H/He mice were exposed to a single dose of 3 Gy of γ-ray from a ^{137}Cs source, and sacrificed 24 hours after irradiation. Bone marrow cells were extracted from femurs, and metaphase samples were prepared without cultivating process. Chromosome banding was achieved by double staining with DAPI [4',6-diamidino-2-phenylindol] and actinomycin D, which enhances contrast of Q-band with DAPI (7). Banded metaphases were photographed and negatives were scanned with a film scanner connected to a Macintosh computer. Incorporated metaphase images were karyotyped on the Macintosh using Adobe Photoshop software. Chromosome-type aberrations were scored for 5 mice, and breakpoints were identified according to standard idiogram of the banding patterns (8).

RESULTS AND DISCUSSION

Table 1 shows frequencies of chromosome-type aberrations. A total of 250 metaphases was analyzed for 5 mice, and 232 breakpoints were observed in 101 aberrant cells. As an average,

Table 1. Frequencies of chromosome-type aberrations.

Mouse No.	Metaphases analyzed	Metaphases with structural aberrations	Number of breakpoints	Brekapoints per metaphase
#1	50	27 (54%)	70	1.40
#2	50	16 (32%)	43	0.86
#3	50	21 (42%)	46	0.92
#4	50	15 (30%)	29	0.58
#5	50	22 (44%)	44	0.88
Total	250	101 (40%)	232	0.93

approximately one breakpoint per cell was detected.

Number of the breakpoints in each mouse was graphed in Figure 1, where breakpoints on chromosome 2 were distinguished from the other chromosomes. Although there are statistical errors, Mouse #5 seemed to have more breaks on chromosome 2 than the other mice. Hence data of Mice #1 to #4 were pooled, and Mouse #5 was analyzed separately.

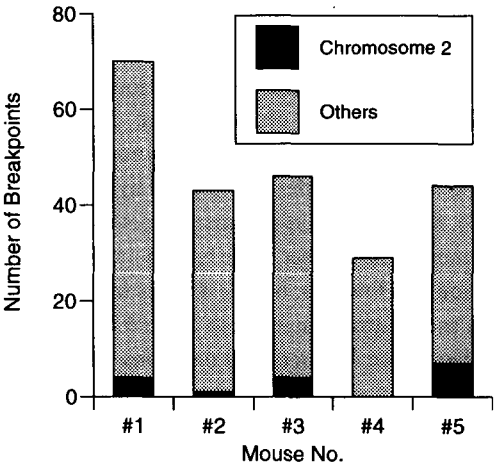

Figure 1. Number of chromosomal breakpoints in each mouse

Figure 2 shows the number of breakpoints in each chromosome for Mice #1 to #4 and Mouse #5. Data of Mouse #5 includes 32 metaphases additionally analyzed. The difference between two graphs is evident, and chromosome 2 of Mouse #5 is conspicuous. Mouse #5 had 13 breakpoints on chromosome 2 in 82 metaphases while Mice #1 to #4 had 9 in 200, 3.5 times higher in relative frequency.

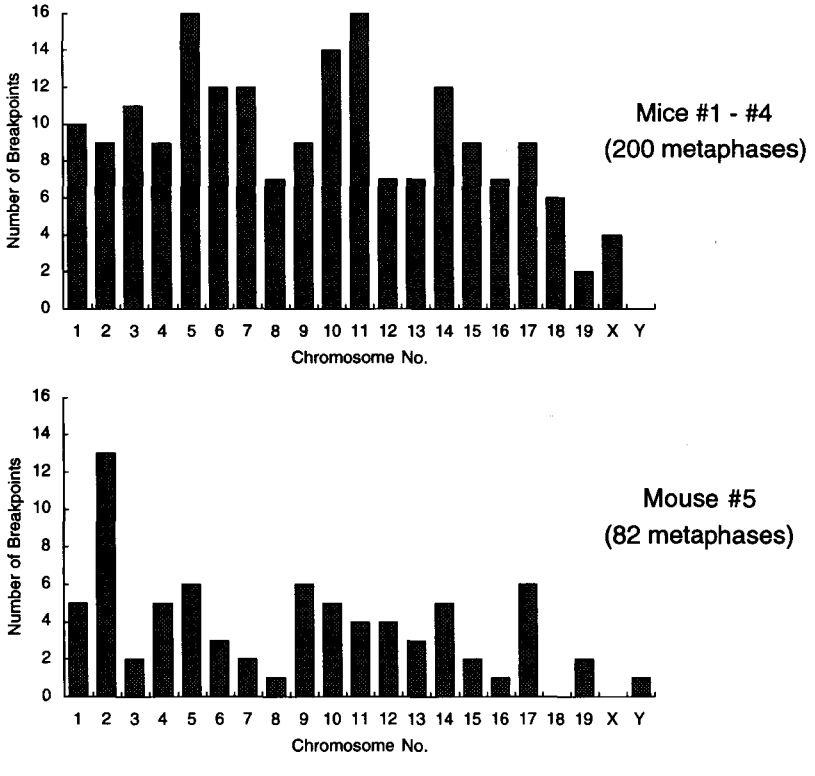

Figure 2. Number of chromosomal breakpoints in each chromosome

Figure 3 illustrates distribution of the breakpoints on chromosome 2. The breakpoints were classified into three aberration categories as described by Savage (9). So far, any sort of cluster is not evident although the number of the breakpoints is not enough. However, Mouse #5 was involved in more intrachanges than Mice #1 to #4. These were all interstitial deletions, typical aberration category of chromosome 2 in murine AML (1-5).

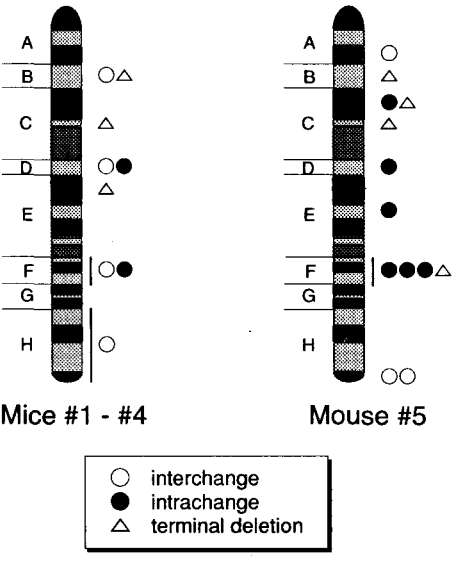

Figure 3. Distribution of breakpoints on chromosome 2

CONCLUSION

Our experiment detected a chromosome 2-sensitive mouse out of 5 mice studied. The result indicates inter-individual difference in chromosome aberration exists even in an inbred strain. It is inferred, if radiation-induced chromosome aberrations are responsible for murine leukemogenesis, there might be AML-sensitive individuals.

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MANIFESTATIONS OF RADIATION SYNDROME IN PIGS

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ABSTRACT

Clinical and hematological changes in pigs after one-time acute radiation by high-energy-X-rays were described.

The radiation of animals was performed by using the linear accelerator of industrial type, with X-rays of 4 MeV, with semi-lethal dose (LD 50/30 = 3,20 Gy) bilaterally (50% of the dose from each side).

The radiation syndrome in pigs developed through 4 stages of the disease: the prodromal stadium (1-3 days), the latent stadium (3-7 days), the stadium of expressed clinical symptoms (7-18 days) and the stadium of recovery (after 25 days). The basic characteristic of the disease was a strong haemoragic diathesis with expressed blood spots on the skin and bleedings from the nose and the anus as well as a rush decrease of blood elements (leukocytes and trombocytes) already 24 hours after radiation.

INTRODUCTION

The studies on the acute radiation syndrome (ARS) in pigs is of great importance because a certain number of morphological lesions are similar to those in people in the same radiation conditions. This refers particularly to the clinical picture and pathomorphological changes on the skin and haematopoietic organs of the breeds of "white" pigs (1-3).

The aim of our investigations was to study the effects of semilethal radiation doses on the manifestation of the radiation syndrome in pigs.

MATERIAL AND METHODS

Six pigs of the "Landras" breed were used in the experiment. They were of both sexes, 4 months old and with average weight of 40 kg. The animals were irradiated with the semilethal dose of high-energy-X-rays (3,20 Gy) on a linear accelerator of industrial type. The irradiation was performed one time bilaterally (50% of the dose on each side). In the course of the 30 post-irradiation days the animals were clinically observed every day and their haematological status was followed on days 1, 3, 7, 14, 21 and 28 after irradiation.

RESULTS AND DISCUSSION

Clinical manifestations of the radiation syndrome in semilethally irradiated pigs can be divided in 4 phases: the prodromal phase, the latent phase, the phase of expressed clinical symptoms and the phase of recovery.

The prodromal phase begins immediately after irradiation and lasts to day 3. It is characterized by slight apathy, inappetence and conjunctivitis of lower degree.

The latent phase attach to the prodromal phase and lasts to day 7. It is characterized by absence of visible clinical symptoms, animals look illusory healthy and consume feed and water normally.

The phase of expressed clinical symptoms begins approximately on day 7 after irradiation and lasts the next 12-15 days. It is characterized by strong apathy, animals mostly lie, refuse feed and water, diarrhea appears, which obtains a profuse character with jelly and blood, the temperature rises and also bronchopneumonia with expressive cough attacks appears. Between days 7 and 10 after irradiation, a strong erythema and hyperaemia occur on the skin of animals, which, in the later phase, transform into dermatitis that has sometimes a pustulous character. Dermatitis is mostly expressed on ears, on the back, on the inside of extremities as well as on the stomach. About day 15, bloody spots and diffused bleedings appear, the skin aches when touched and breaks easily. In their further development, skin lesions convert into bloody stained dark crusts that give a "sooty" appearance to the whole change. In this phase also bleedings from the nose and the anus are manifested and on day 18 the first deaths were noticed.

The phase of recovery begins after the day 25 in animals that survive. It is characterized by a gradual disappearance of symptoms that were expressed in the previous phase, but even after day 30 the survived animals couldn't be characterized as clinically healthy.

The frequency and the period of appearance of clinical symptoms of the radiation syndrome are shown in figure 1.

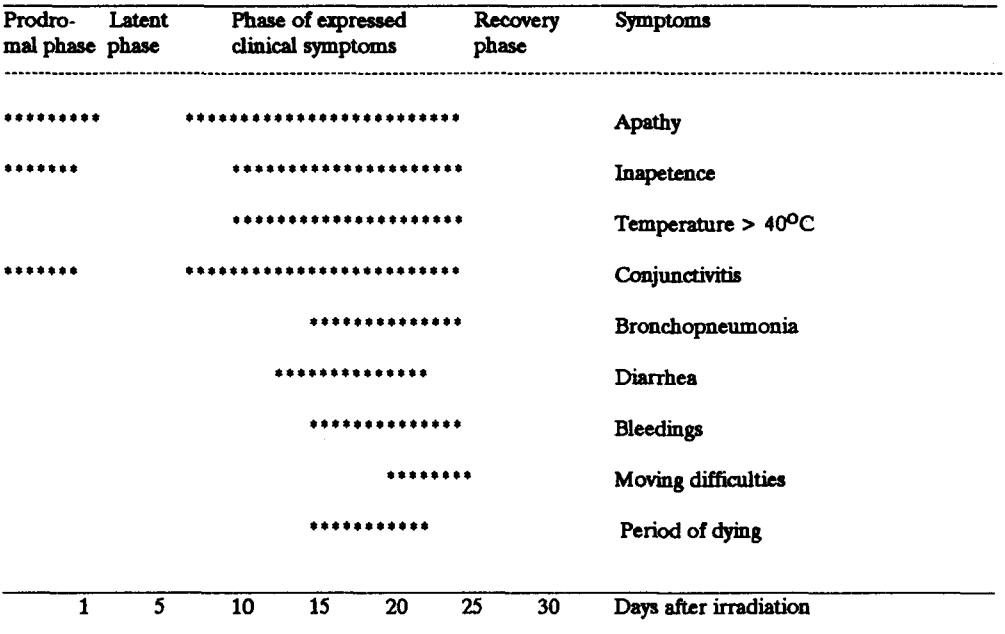

Figure 1. Frequency and period of appearance of clinical symptoms in semilethally irradiated pigs.

Table 1. Relative values of leukocyte and trombocyte number in semilethally irradiated pigs.

Days after irradiation	Relative number of leukocytes, % in relation to the normal	Relative number of trombocytes, % in relation to the normal
0	100	100
1	70	80
3	50	60
7	40	45
14	20	15
21	30	20
28	40	35

Haemathological changes were reflected in a rush decrease of the formed blood elements (leukocytes and trombocytes) already 24 hours after irradiation (table 1).

CONCLUSION

The main characteristic of the radiation syndrome in semilethally irradiated pigs was a strong haemorrhagic diathesis with expressed bleedings on the skin and from the nose and the anus. The changes on the skin were manifested by strong hyperaemia, erythema and dermatitis, especially on the inside of thighs and on the stomach. In the later phase, blood spots and diffused bleedings on the skin and from the nose and the anus appeared. Changes in the haemathological status were obvious already 24 hours after irradiation, with a drastic decrease of leukocyte and trombocyte number. On day 14 after irradiation, the leukocyte and trombocyte number decreased to only 10-20% of their normal value.

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April 14-19,1996
Vienna, Austria

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PAPER TITLE ESTIMATION OF RADIOPROTECTION EFFECTS OF UKRAIN BY MATHEMATICAL THEORY OF EXPERIMENTS

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MAJOR SCIENTIFIC TOPIC NUMBER 6.3 (see page 7)

ABSTRACT (See instructions overleaf)

The problem of choosing the optimal conditions using drugs is a cardinal one for any experimental research. Basing on principles of mathematical theory of experiments, optimal doses and periods of drug administration were determined using both the rate of survival at day 30 following as response indices.

The ability of Ukrain (Uk), a cytostatic and immunomodulating semi-synthetic compound of thiophosphate-modified alkaloids of Chelidonium Majus L., to modify the effects of irradiation was studies in CBA mice.

The doses of irradiation from 6.0 to 7.5 Gy at the dose rate of 1.5 Gy/min, the dose of Uk from 0.2 to 10 mg/kg and time: from 24 h before t
o 24 h after irradiation of its administration were chosen as the most significant factors that may influence the survival of irradiated animals. The drug (0,2 -1,4 mg/kg) was found to increase the survival rate of mice by 50 - 60 persent at irradiation doses from 6.00 to 6.75 Gy with no effect at the dose of 7.50 Gy. Maximal increase in the survival rate of the mice was noted when the drug was administreted within the interval from 6 h before to 3 h after irradiation.

To conclude, the main outcome of the work presented is the finding that Ukrain is capable of modifying the effects of irradiation when applied in both prophylactic and curative regimes. The drug was also found to be able of modifying the effects of prolonged irradiation with the cumulative dose of 8.75 Gy (LD 85/30) at the dose rate of 0.02 Gy/min.

Abstract no.: 90258
Poster No.: P24-9

AUTOIMMUNE PROCESS IN CNS UNDER Cs-137 INNER IRRADIATION

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INTRODUCTION

Autoimmune hypothesis as to the development of radiation-induced brain injuries stands high among the concepts of the CNS post-radiation damage pathogenesis.

To study the changes occurring in a living organism affected by a small-dose radiation due to incorporated radionuclides as well as to create adequate models are of critical importance in the post- Chernobyl period.

The effects of chronic small-dose inner radiation on the development of autoimmune responses were evaluated by determining the level of the CNS proteins and protein-induced antibodies to the CNS components.

MATERIALS AND METHODS

Forty eight ordinary mice weighing 140-180 g were used throughout the experiments. The mice were kept on isocaloric diet in vivarium and received 600 Bk of Cs-137 with food daily. The control group mice (n=16) were kept under similar conditions out of isotopic unit being given similar food free of Cs-137.

The tests were performed after 1,3,6,9 months since the beginning of experiments and in 3 months after a 6-month Cs-137 intake has been terminated.

After 1,3,6,9 months the animals were decapitated, and the blood were taken for serum. In the serum of experimental mice we studied the content of autoantibodies to neurospecific proteins (NSP): MBP, S-100 and NSE (neurospecific enolase) as well as the availability of blood-circulating MBP and NSE. NSP were obtained from calf brain. MBP was isolated using Palladin et al.(1970) technique while S-100 and NSE were isolated after G.A.Berezhnoy (1978). Antibodies to NSP were determinated by the Voller et al (1979) method modified by T.M. Cherenko (1989) using indirect immunoferment technique. To determine the MBP content in the blood serum standard ELISA modified by T.M.Cherenko was used. NSE serum concentration was measured using a set of reagents for NSE assay in human blood serum (NSE EIA <DIAplus>, F.Hoffman-La Rosch & Co., Basel, Switzerland).

RESULTS AND DISCUSSION

Analysis of mean levels of antibodies to MBP shows that at the initial stage of inner ionizing radiation their concentration holds low. However, starting from 3-6 and 9 month particularly, it increases definitely. Titres of autoantibodies to S-100 and NSE were seen to grow gradually within 1-9 months.

In mice with a 3-month interval after a 6-month radiation exposure mean levels of antibodies to all the three NSP have reached control values. The level of autoantibodies to MBP in these mice slightly exceeded the control values, whereas levels of autoantibodies to S-100 and NSE were indefinitely lower.

The most statistically confident autoantibody accumulation against standard value was observed within 6 months (to S-100) and 9 months (to MBP and NSE).

While studying the development of neurosensitization it appeared that autoantibodies to the three NSP: glia markers, neurons and myelin were identified not in every mouse. Out of the mice

having a 1 - month Cs-137 intake only 30 % revealed autoantibodies to NSE and S-100. In this period autoantibodies to myelin structures are not yet registered.

After 3 months of radiation exposure autoantibodies to MBP are also revealed in 80 % of animals. In 9 months autoantibodies to all the three NSP are identified. Practically half of the animals under test exhibit autoantibodies to MBP and NSE while autoantibodies to S-100 appear less frequently.

Thus, it may be concluded that: 1) Not all the animals develop autoimmune responses; 2) Autoantibodies to different antigens are revealed with different frequency. Throughout all experimental periods autoantibodies appear mostly to glia and neuron proteins, whereas autoantibodies to MBP are seen in more prolonged time (from the 3-rd month), being found, however, in the majority of mice.

In early period (1 month) autoantibodies appear not to all antigens which fact allows a suggestion that immune responses to different brain structures are triggered step-by-step. This may evidence for a stepped character of the brain damage under small-dose radiation. The most sensitive brain structures (glial and nervous cells) are the first to suffer, while less sensitive ones (myelin and oligodendroglia) undergo damage later.

Of particular interest are data obtained as to specificity of autoimmune responses in mice having a 3-month interval after a 6-month radiation exposure. 66 % of this animals reveal autoantibodies to NSE, whereas only 16,7 % show autoantibodies to MBP, this value being lower than that in the group receiving isotopes during 9 months.

Thus, autoimmune responses remain to exist after isotopic exposure and nuclide withdrawal from the blood and brain, acquiring the features of a chain reaction.

Different frequency of autoantibody appearance in the groups of animals receiving and not receiving isotopes testifies to the dynamical character of the process and its relatioship with the brain and immune system injuries. The fact that autoantibodies to NSP were identified not in all but in a part of the animals under test points to the existence of an individual sensivity (threshhold) to inner radiation. It cannot be excluded that such a sensitivity is controlled by histocompatibility gens or by other censor systems.

Considering the reasons why autoantibodies are inavailable in some animals" blood the following assumptions can be made: 1) Autoantibodies are almost entirely bound to the blood-circulating antigens; 2) Autoantibodies are produced through the immune system suppression; 3) There exists either cell or humoral inhibitor to slow down the development of immune responses, and whose nature is to be studied farther on.

To understand more completely the mechanisms of radiation effects on the brain structure and function, we determined the serum content of MBP. The results tell of a wave-like protein content in the blood. After a month of radiation exposure the highest MBP content amounted to 258 ng. Later on the MBP level decreased and remained within 100-123 ng.

The data obtained may be compared to the level of autoantibodies to MBP. In early period (1 month) no autoantibodies to MBP are seen in the animals" blood, although this is a period of the highest MBP concentration. In more prolonged periods there appear autoantibodies which bind and remove the antigens from circulation thereby reducing their content in the blood. It may be suggested that in early periods (1-st month) radionuclides destroy nervous cells most actively. Then the compensatory mechanisms are triggered to prevent further brain damage and antigen outflow into the blood. Both mechanisms are likely to be involved: on the one hand the processes of primary and secondary brain damage are taking place, and on the other hand the compensatory mechanisms are being triggered.

Of special interest seem to be the data on the MBP content in the animals' blood after a 3-month restoration period when the animals received no radionuclides. The MBP content in these animals was found to lower practically to the control values pointing to regenerative processes and destruction intensity decrease.

When determining the NSE content, its level was noticed to grow after the 6-th month of radiation exposure reaching maximum after 9 months. In the group of animals having a 3-month interval after a 6-month Cs-137 consumption this level was lower than in the group exposed to radiation during 6 months but higher than in the control group, this fact being probably indicative of the onset of reparative processes in the brain.

CONCLUSIONS

After 1,3,6 and 9 months of Cs-137 dietary intake 50-60% of experimental animals releaved the availability in the blood of antibodies to the brain structures as well as blood-circulating neuroantigens MBP and NSE, this evidencing for the development of destructive processes in the brain due to incorporated radionucludes.

The antibody level was found to grow gradually and continuously from the 1-st to the 9-th month. However, a 3-month interval after 6 months of Cs-137 intake involves statistically significant decrease of neuroantibody and autoantigen blood levels as compared to the animals fed with Cs-137 without intervals, this testifying to reparative processes in the brain.

Inner radiation disturbs blood-brain barrier permeability, allows the brain transbarrier tissues to contact the immune system thereby enabling a stopped autoimmune process to be developed in the CNS. It follows that autoimmune responses to the brain antigens may prove the existence of the brain organic damage because of inner radiation due to incorporated radionuclides.

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PAPER TITLE NITRIC OXIDE AND RADIOINDUCED DAMAGE IN DEVELOPING BRAIN

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1 (see page 7)

ABSTRACT (See instructions overleaf)

A key role for nitric oxide (NO) has been postulated in the pathogenesis of brain damage following acute insultes. To ascertain the relationship of NO to glutamate neurotoxicity and its correlation with morphological and biochemical changes observed after prenatal exposure to ionizing radiation, an animal model was used.

Fetuses of Wistar rats were irradiated intra-utero on 13th day of gestation with 0.4 Gy, 0.7 Gy and 1 Gy with gamma rays, and sacrificed at postnatal days 3 and 10.Brain samples were taken for histopathological and biochemical studies: superoxide dismutase (SOD), catalase, guanylyl ciclase, D-T diaphorase and NO-sinthase activities and lipoperoxides and DNA levels.

It was observed a dose-dependent loss in neuronal density and a decrease of total brain DNA. Correlated changes in oxidative stress were found. We report here an increased production of NO and higher guanylyl ciclase, NO sinthase and D-T diaphorase activities at the higher dose.

These data suggest that the NO and the intracellular signaling molecules may play a possible role in the radioinduced damage in developing brain.-

MECHANISMS OF NEURO-ENDOCRINE REGULATION IN THE CONDITIONS OF SMALL IRRADIATION DOSES AND POST-CHERNOBYL RADIATION SITUATION

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Study was held of function of thyroid, parathyroids, gonads, pancreas, adrenal and pituitary glands, transport of hormones in blood, interaction with membrane, cytoplasma and nuclear receptors of different in radiosensitivity and hormone-dependency tissues under the action of small doses of acute and chronic external and internal irradiation. The state of central and peripheral neuro-humoral regulation of cardiovascular system was assessed. The analogous researches were carried out in the conditions of radionuclides pollution after the Chernobyl accident. The obtained data reveal an expressed disturbance of processes of formation, regulation, transport, reception of hormones and neuromediators, which is conditioned by the character, dose and time period after the irradiation. The results serve as the basis of formation of nearest and remote radiation effects and pathology.

NEUROREGULATORY PROCESSES IN THE CNS STRUCTURES LONG AFTER CHRONIC EXPOSURE TO LOW DOSES OF IONIZING RADIATION AND LEAD AND CORRECTION OF THE DISORDERS

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In experiments on mature Wistar male rats exposed to different modes of ionizing radiation at relatively low doses and in combination with chronic probe-assisted application of lead acetate (5 mg/kg daily) we found changes in the intensity of neuronal uptake of some neurotransmitters that characterize the functional state of the corresponding synaptic formations of the diencephalic region and other brain structures long after radiation. A decrease in the functional activity of the mediobasal hypothalamus was accompanied by disorders of neurohumoral regulatory mechanisms in the hypothalamus-pituitary-adrenocortical system and hypothalamus-thyroid system. Application of therapeutic doses of neurotropin (Nipon Zoki, Japan) and thymaline 3 months after cessation of radiation and lead stimulated an increase in specific binding of glucocorticoids by brain structures, activation of the hypothalamus-pituitary-adrenocortical system, and some normalisation of neurotransmitter relations in centres regulating autonomic functions 6 months after.

A NEW CONCEPT ON THE BASIC MECHANISM OF EFFECTS OF PRENATAL EXPOSURE IN THE DEVELOPING NERVOUS SYSTEM OF A NEW ORGANISM.

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INTRODUCTION

Congenital nervous deficiency is known to be the only meaningful teratogenic end point ascertained in prenatally irradiated children, for instance, in survivors of atomic bombing [1]. Similar effect is known to be the aftermath of prenatal exposure to alcohol [2] or lead [3]. Besides, an opinion exists [4] that prenatal exposure to any agent affecting physical development can have such a consequence. Consequently, any proposed mechanism through which exposure of an embryo (to radiation) produces alterations of nervous system function of a newborn must disclose the nature of this "overall" vulnerability of the nervous reactivity of a newborn to any prenatal exposure.

We have found such a mechanism theoretically and then verified it in animal experiments.

A BRIEF THEORY OF EFFECTS OF PRENATAL IRRADIATION

Teratogenic action of prenatal exposure starts from impinging upon proliferative and other characteristics of embryo cells. Embryonic development is stabilized by morphogenetic homeostasis, so that this impingement induces homeostatic (adaptive) response aimed at normalization of the development. An irradiated embryo consists of cells with decreased proliferative capacities and increased variety of all their characteristics. Systemic analysis shows [5] that under these conditions, a certain overdevelopment of the endocrine component of regulation of a growing organism is produced by adaptive response to normalize the process of the development. If this endocrine "overregulation"remained to be noncompensated in the course of the subsequent development, a certain overdevelopment of the entire neuroimmunoendocrine regulation in a newborn could occur, therefore further adaptive response induces coadaptive underdevelopment of nervous and immune components of regulation which are ontogenetically later than the endocrine component. These side effects of the adaptive response appear to be just this power systemic motivity for producing nervous (and immune) deficiencies in a newborn. So, induction of the nervous deficiency appears to be an element of the normal adaptive response to embryo cells damage, and this fact is just an explanation of the overall vulnerability of nervous system of a newborn to any deleterious prenatal exposure.

According to this mechanism, prenatal irradiation can induce malformed structure of neuroimmunoendocrine regulation in progeny (e.g. the existence of excess endocrine regulation accompanied with nervous and immune deficiencies), even if it occurs during the pre-implantation stage of embryogenesis. This statement stands in a strong contradiction with the common scientific opinion [6] that irradiation at this stage leads to "all or none effect": it can result in an embryonic death, however those that survive appear to be normal. Hence, the proposed mechanism would be proved if regulation effects of pre-implantation irradiation are found in alive progeny.

RESULTS OF VERIFYING EXPERIMENTS

We have found just these effects in progeny of pregnant Wistar rats exposed to 2 Gy of gamma-irradiation protracted at the initial 6 days of pregnancy. Comparatively to controls, increased body mass and fatness of irradiated in utero progeny was observed from 42-th day of postnatal life and later, with these patterns being decreased from birth up to 28-th day. So, an endocrine overcompensation of the aftermath of radiation damage occurred in the late period of postnatal life of progeny. After acute gamma-irradiation at the 100-th day of postnatal life, irradiated *in utero* progeny had much more profound body mass loss than controls did. Decreased systemic response to acute irradiation is known in animals having an obvious endocrine regulation deficiency [7]. In light of this fact, more profound response which appeared to be characteristic for those irradiated *in utero* is a clear indication that pre-implantation irradiation had really induced excess endocrine component of regulation. Hence, predicted existence of "endocrine regulation imprinting" of damaged properties of embryo cells have met the experimental corroboration.

The question is: has this endocrine imprinting been accompanied with the production of alteration of the nervous system function?

First, comparatively to controls, increased relative mass of adrenals on 42-th, 49-th and 120-th days of postnatal life was observed in irradiated *in utero* animals. It is an indication that syndrome of chronic adaptation have been induced. Second, these animals had the increased level of autoantibodies to S-100 brain protein. It is an indication of their diminished training abilities. Hence, alteration in the nervous reactivity was induced by irradiation at the pre-implantation stage of embryogenesis.

However, some more surprising is the fact that pre-implantation irradiation also altered immune reactivity of progeny.

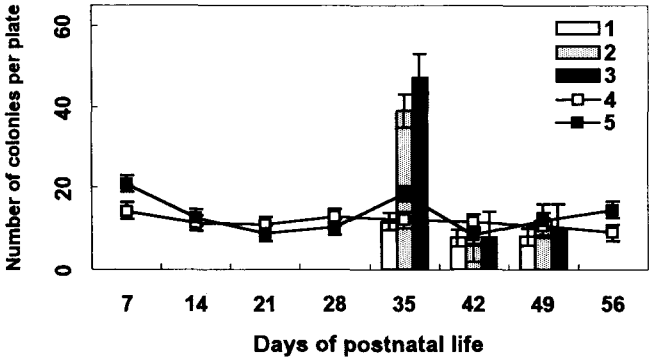

Figure 1. The temporal dynamics of automicrophlora in control animals and those irradiated *in utero* at the pre-implantation stage of embryogenesis.

1 - Control animals. 2 - Animals irradiated *in utero*.
3 - Control animals injected with ram erythrocytes or horse serum.
4 - Animals irradiated *in utero* and injected with ram erythrocytes on 28-th day of postnatal life.
5 - Animals irradiated *in utero* and injected with horse serum on 28-th day of postnatal life.
Vertical bars represent 95-% confidence intarvals.

First, the titre of antibodies in irradiated animals in response to injection of ram erythrocytes or horse serum was increased comparatively to controls. Second, abnormal temporal dynamics of skin automicroflora was observed in irradiated *in itero* progeny after its weaning on 28-th day of postnatal life, especially if weaning was

accompanied with injection of ram erythrocytes or horse serum (see Fig 1). Thirdly, peritoneal anaphylaxis test has shown that increased vulnerability to allergy was characteristic for prenatally irradiated animals.

Significant correlation of distorted immune regulation patterns was ascertained. Thus, in the full accordance with the proposed systemic(!) mechanism of induction of immune deficiency, observed alterations in immune reactivity were credible to belong to the unique pathological process launched by pre-implantation irradiation.

CONCLUSION

Ectopic gray matter is known to be the major brain defect responsible for nervous deficiency induced by prenatal irradiation in humans [1] and rodents [8]. This defect is also known to be induced by in utero action of alcohol [2] being a cytotoxic agent. It seems incomprehensible that similar brain defects are due to in utero exposure to agents having absolutely different mechanisms of primary action at (embryo) cells: radiation is a typical genotoxic agent. But this fact appears to be just corresponding with the proposed systemic mechanism of induction of the nervous deficiency. Ectopic gray matter is known to be due to arrest in migration of immature neurons. Instead of being an aftermath of the direct damage, this effect is much more credible to be an end point of a certain systemic response of a growing organism aimed at diminution of potencies of the nervous system of a newborn.

The proposed mechanism has predicted effects in immune systems of alive newborns due to irradiation at pre-implantation stage of embryogenesis. Immune deficiencies were never observed before as (radiation) teratogenic effect. Nevertheless, we have observed such effects in experiment. This means that our concept appears to pass through a "difficult examination" in the light of the world's radiobilogical knowledge.

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PAPER TITLE ENHANCED FREQUENCY OF ABNORMAL SPERMATOZOA AFTER COMBINED X-RAYS EXPOSURE

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MAJOR SCIENTIFIC TOPIC NUMBER 2 (see page 7)

ABSTRACT (See instructions overleaf)

ENHANCED FREQUENCY OF ABNORMAL SPERMATOZOA AFTER COMBINED X-RAYS-ACRYLAMIDE EXPOSURE

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Living organisms are exposed to a various types of radiation and different types of chemicals. Combined X-rays-acrylamide exposure is possible in industry and in scientific laboratories. Study was performed using sperm morphology test. Male mice were treated with 0.25 Gy of X-rays, 75 mg/kg bw of acrylamide (AA) and with combination of both agents. All stages of spermatogenesis were exposed. Spermatozoa were examined and classified as normal, banana like, lacking in hook, amorphus, folded and twin tailed forms.

Acrylamide induced sperm-head abnormalities after exposure late spermatids and early spermatocytes. X-rays clearly increased the frequency of abnormal spermatozoa after exposure spermatozoa, late spermatids and spermatocytes. Combined X-rays-AA treatments increased the frequency of misshapen sperm head after exposure all stages of spermatogenesis except late spermatocytes and spermatogonia.

Effect observed after combined exposure even in low doses significantly exceeted the sum of effects produced alone by the agents.

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PAPER TITLE Effects of 239-Pu on foetal haemopoietic cells.

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1. (see page 7)

ABSTRACT (See instructions overleaf)

When considering risks for radiation protection purposes the embryo\foetus is considered as an especially sensitive subgroup. To establish radiation risk estimates for the embryo\foetus relevant organ systems, exposed to ionising radiation at low doses, need to be studied. There are data for low LET radiation, but the sensitivity of the foetus and, more particularly, foetal haemopoietic cells to high LET radiation has not been fully investigated. This scenario has assumed some importance in relation to discharges into the environment.

The experiments presented here, part of larger carcinogenesis study, were designed to evaluate the effect of intrauterine exposure to ^{239}Pu on foetal haemopoietic cells. Groups of pregnant mice were infused with ^{239}Pu at concentrations of 0, 80, and 160 kBq\kg throughout gestation. Both these activity groups resulted in doses to the foetus considerably higher than that received from environmental contamination but at a dose rate much less than used in other reported work.

Cell culture studies have shown that, up to 1 year post partum the cellularity of the femoral bone marrow was undisturbed. The number of stem cells, myeloid cells and microenvironmental fibroblasts - as measured by CFU-A, CFC and CFU-f respectively - were also unaffected. This study will be compared with those which have shown that a single injection of ^{239}Pu or ^{241}Am during gestation led to significant alterations in haemopoiesis (Van Den Heuvel, 1990, Int. J. Radiat. Biol., 57, 103; Mason et al, 1992, Int. J. Radiat. Biol., 61, 393.).

DOSE ESTIMATION OF ENRICHED URANIUM IN TESTES ON INDUCTION OF DOMINANT LETHALITY AND SKELETAL ABNORMALITIES

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INTRODUCTION

Enriched uranium is one of the principal fuels of nuclear power stations (1). Since now in the sphere of radiological medicine what is concerned about the environmental pollution and damage to human beings by nuclear fuels and its fission products released by nuclear tests and plants. Especially in recent years nnuclear power stations are increasing rapidly, the production and consume of enriched uranium expand year by year. Consequently, observations of its harming effect on environment and in the body become a significant task. Especially its action on induction of dominant lethality and skeletal abnormalities showed a close relation on retentive peculiarity in testes. So we paid attention to the relationship between the cumulative absorption dose and the incidence of the dominamt lethality and skeletal abnormalities in the offsprings.

EXPERIMENTAL METHODS

(1) Radiation dose estimation of enriched uranium in testes.

According to the retention function rs(t), with respect to time accumulation of 28d, the cumulative activities $A_s(t)$ (Bq) of enriched uranium in testes were obtained:

$$A_s(t) = \int_0^t A_0, s. rs(t). dt$$

Here A_0, S=early radioactivities of enriched uranium in testes.

Again from the cumulative radioactivities of enriched uranimm in testes calculated the absorption dose (2, 3). Dose estimation may be according to the formula as follows:

$$D_r = 1.6 \times 10^{-10} \sum_s \sum_j [A_s \Sigma SE(T \leftarrow S)_i] \text{ (Gy)}$$

Here $SE(T \leftarrow S)_i = \dfrac{Y_i E_i AF_i (T \leftarrow S)}{M_T}$ [(MeV/g/nucleus conversion)

Y_i=produce of α radiation

E_i=average energy of α particles

M_T=mass of the organs

$AF_i(T \leftarrow S)$=absorption fraction=1

So that the cumnlative absorption dose may be calculated by contribution of three uranium radionuclides.

(2) The rate of dominant lethals and skeletal abnolmalities induced by enriched uranium.

Sexually mature male and virgin female BACB/c strain mice, 10 weeks old and weighing 21±1g, were used in this study, Experimental animals received intratesticular internal irradiation by enriched uranium with doses of 0.4, 2, 10, 20,

40 and 60ug. After 13 days treatment, each male was allowed to mate with 2 virgin females for a period to 49 days. The females were killed 18-20 days after the beginning of the mating, and the number of corpora lutea, as well as living and dead embryos were counted. So that the rate of dominant lethals were calculated.

Then living and dead embryos are taken out of the mating females, and are fixed in 95% ethyl alcohol for 7 days. One after another are turned into 1% KOH solution for 6 days (5). After a while embryos are stained with 0.05% of alizarin red for 1 day. Experimental specimen are infiltrated with a miscible liquid with ethyl alcohol-glycerin-KOH for 8 days. Dominant skeletal mutations were detected in the skeleton specimen.

EXPERIMENTAL RESULTS

(1) Contribution of the cumulative absorption dose in testes by three uranium radioisotopes

Uranyl fluoride containing ^{235}U of 18.9% (60mg/ml) was used in this study. Sexually mature male BALB/c mice, 10 weeks old and weighing 21±1g, were randomly divided into 6 experimental groups. There were 5 mice in each group. Animals treated by single intratesticular injection(4) with different doses of enriched UO_2F_2 ranging from 0.4 to 60ug. We estimate the cumulative absorption dose of the three radioisotopes of the experimental enriched uranium with 18.9% abundance through 28d. It was deducted from the data shown in Table 1.

Table 1. Contribution of the cumulative absorption dose through 28d in testes by three uranium radioisotopes after injection of enriched uranium with different doses.

Injected	Cumulative absorption dose in testes, Gy			
doses, ug	^{234}U	^{235}U	^{238}U	Total
0.4	8.639×10^{-5}	2.903×10^{-6}	2.103×10^{-6}	9.140×10^{-5}
2	4.320×10^{-4}	1.451×10^{-5}	1.052×10^{-5}	4.570×10^{-4}
10	2.160×10^{-3}	7.257×10^{-5}	5.257×10^{-5}	2.851×10^{-3}
20	4.320×10^{-3}	1.451×10^{-4}	1.052×10^{-4}	4.570×10^{-3}
40	8.639×10^{-3}	2.903×10^{-4}	2.103×10^{-4}	9.140×10^{-3}
60	1.304×10^{-2}	4.354×10^{-4}	3.154×10^{-4}	1.380×10^{-2}

It should be noted, that the contribution of total α radiation by ^{235}U was only <5% and >90% fraction of total α radiation came from ^{234}U. Although the content of ^{234}U in natural uranium is less, but physical half life of ^{234}U is comparatively shorter, and then the α radioactivity is higher. Therefore the contribution of α radiation is chiefly by ^{234}U.

(2) The rate of dominant lethals induced by enriched uranium

Experimental results indicated, that therateofdominant lethals induced byenriched uranium was elevated with the increasing doses of enriched uranium.

Table 2 gives the distribution of the females according to the number of living and dead embryos induced by different doses of enriched uranicum. It shows that the increase in the mean number of dead embryos is the result of an increase of the radiation dose of enriched uranium. Our present results show that doses of

euriched uranium given to mouse mainly increase the rate of intrauterine deaths. Whereas the mean number of implantations is only slightly altered.

Table 2. The rate of dominant lethals induced by different doses of enriched uranium

Injected doses, ug	Pregnant females(%)	Live embryos/♀	Dead embryos/♀	Ratio with 1 dead embryos/♀	Ratio with >2 dead embryos/♀
0	83	7.00	0.63	0.5	0.13
0.4	100	7.50	1.25	0.75	0.38
2	100	6.22	1.33	1.00	0.33
10	100	5.73*	1.63	0.81	0.36
20	83	4.80*	3.20**	1.00	0.80
40	77*	5.63*	4.88**	1.00	0.88
60	71*	4.60*	2.60**	1.00	1.00

*p>0.05 **p<0.01

(3) The rate of dominant skeletal abnormalities induced by enriched uranium.

In mutation-rate experiments, dominantskeletal mutations were confirmed by breeding tests. Skeletal abnormalities were detected in the skeletons of some of the sons of irradiated males. Table 3 summarizes certain parts of the skeleton appear to be especially subject to change by dominant mutations such as ribs fusion, extra rib on one side, and extra ribs on both sides,as well as extra point ribs. Especially gaps in skull, and spine defect, were foundby large doses of enriched uranium.

Table 3 The rate of dominant skeletal abnormalities induced by different doses of enriched uranium

Injected doses, ug	Total embryos	Embryos with dominant skeletal abnormalities				Rate of sheletal abnormalities
		Extra rib on one side	Extra ribs on both sides	Extra point ribs	Other	
0	55	2	8	0	0	18.2
0.4	66	7	6	7	1	31.9
2	59	5	5	6	0	27.1
10	66	12	9	4	1	39.4**
20	43	3	9	8	0	46.5**
40	41	2	23	3	0	68.3**
60	17	1	9	0	3	76.5**

**p<0.01

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PAPER TITLE Effects of ionizing radiation and early treatment agents on hepatic regeneration

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MAJOR SCIENTIFIC TOPIC NUMBER 2..... (see page 7)

ABSTRACT (See instructions overleaf) The purpose of this work was to study the action of radiomodifying drugs of detoxicators (gluconeodes) and immunomodulators (interleukin-1β) classes on reparation (a model of regenerating liver). Outbred white male rats weighing 160-200 g were used in the experiments. The animals were irradiated at a dose of 6.75 Gy using IGUR-1 source at a dose rate of 1.58 Gy/min. 1,4,24 or 72 hours after irradiation the left and central lobes of the liver of the animals have been removed. The intensity of regeneration was judged by insertion into their DNAs of ^{3}H-thymidine, wich was administered to the rats 20 hours after hepatectomy. As a result of the experiments, it has been established that hepatic regeneration in irradiated rats with hepatectomy having been performed 1 or 4 hours after radiation exposure has been completely inhibited. One day following radiation exposure, the intensity of regeneration was 60% of the norm. On day 3 after radiation exposure, the rate of regeneration normalized. Single administration of gluconeodes for therapeutic purposes led to an insignificant acceleration of the regeneration process as compared to irradiated control (with hepatectomy 1 and 4 hours after irradiation). A significant effect of this agent (after single or 3 times a day administration) has been revealed on days 1 and 3 of radiation sickness when the rate of hepatic regeneration 1.5-2 times exceeded the norm. Similar effect has been revealed after a single dose of interleukin-1β (25ug/kg). Thus, the mechanisms of action to different classes of compounds have a similar positive effect on the processes of regeneration.

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PAPER TITLE EFFECT OF RADIOACTIVE PHOSPHORUS P-32 ON CALCIUM, PHOSPHORUS AND MAGNESIUM CONCENTRATION AND ALKALINE PHOSPHATASE ACTIVITY IN BLOO PLASMA OF CHICKENS

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MAJOR SCIENTIFIC TOPIC NUMBER 2. (see page 7)

ABSTRACT (See instructions overleaf)

The aim of this study was to investigate the effects of internal contamination with lethal amount of bone-seeking isotopes P-32 on mineral metabolism and alkaline phosphatase activity for detecting the damage caused by P-32 on the chicken's organism. The experiments were carried out on a fifty-day old hybrid chichens of "JATA" breed on both sexes. The concentration of calcium, magnesium and inorganic phosphorus as well as activity of alkaline phosphatase were determined in blood plasma. Obtanined results have shown as follows:

1) magnesium concentration was significantly reduced on the first and the seventh day after contamination,
2) two significant drops of calcium values on the first and the third day after contamination,
3) the phosphorus concentration was significantly reduced on the seventh day after contamination, and
4) alkaline phosphatase activity was significantly increased on the first day after injection of P-32.

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PAPER TITLE Red-pigment-producing Monascus ruber strains obtained by radiomutagenesis

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MAJOR SCIENTIFIC TOPIC NUMBER 2.1. (see page 7)

ABSTRACT (See instructions overleaf) A strain of Monascus ruber from the Collection of the Institute of Food Research, ICA 3.250 and another two radioinduced mutant strains, ICA 3.250 M_1 and ICA 3.250 M_2 were used.

Culture was performed from the stock culture on corn flour-solid media with intermittent stirrings for 7 - 10 days at 30°C and the pigments were extracted by Et - OH.

The pigment biosynthesis on liquid media was performed using the synthetic medium containing 4 % glucose; 0,1 % KH_2PO_4; 0,05 % $MgSO_4$. 7 H_2O; 0,05 % NaCl and 0,01 % $FeSO_4$. 7 H_2O in the shaker - type incubator (240 rpm), at 30°C for 6 days. The samples were analysed by UV - VIS spectrophometry.

The irradiations were performed at a Co - 60 source with an activity of approximately 105 TBq, the irradiations being carried out statically, in a cylindrical geometry, in the dose range of D = 1 + 10 kGy, at 1 kGy/h dose - rate.

There was determined the viability percentage versus the unirradiated (control) sample, separately for every irradiation dose, to obtain the dose-effect curves, the radiosensitivity parameters and the optimum mutagenesis dose range.

There were obtained two mutants (at 4 kGy and 6 kGy) which were analysed for pigment biosynthesis.

RADIOSENSITIVITY OF THE GUINEA-PIG OOCYTE AT DIFFERENT STAGES OF FOLLICULAR DEVELOPMENT

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INTRODUCTION

Recently, we have shown that the guinea-pig represents an excellent model for studies on genetic hazard of radiation in man (1). We also developed techniques for culturing oocytes of this species and preparing their meiotic chromosomes for cytogenetic examination (2). In the experiments reported here, we examined the radiosensitivity of oocytes at two different stages of maturation, separated by only one week.

MATERIAL AND METHODS

In a first series of experiments, ovaries of adult females were X-irradiated with 1 or 2 Gy on day 3 of the 17-day estrous cycle. Their meiotically competent oocytes were punctured from the growing follicles on day 10. They were cultured for 6 h to the MI stage, and fixed for cytogenetic analysis.
In a second series of experiments, ovaries were irradiated with the same doses on day 10 of the estrous cycle. In this case, oocytes were collected and cultured either immediately after irradiation, or at various time intervals thereafter.

RESULTS

As shown in Table 1, important differences were found in the yields of chromosome aberrations, according to the time of the estrous cycle at which irradiation had occurred.
Irradiation at the beginning of the cycle induced low numbers of translocations and slightly higher numbers of other aberrations. These effects were dose-dependent. The frequency of oocytes showing translocations was about 1.5 Gy^{-1}, a value comparable to that obtained after irradiation of the immature oocyte (3).
When irradiation was delivered one week later, and oocytes were collected and cultured immediately thereafter, the frequencies of oocytes showing chromosome anomalies revealed much higher. This was true for translocations as well as for other types of chromosome aberrations. Multivalents implied variable numbers of chromosomes and in many cases the number of chromosomes implied could not be determined. These highly damaged metaphases were frequently associated with chromatid breaks. The frequency of oocytes carrying translocations was 30-35 % Gy^{-1}.

In some experiments, oocytes which had been irradiated on day 10 were collected and cultured one day later. Surprisingly, MI preparations revealed impossible to obtain: all fixed oocytes were already in MII, whether they had been irradiated with 1 or 2 Gy. This phenomenon was studied more in details, using various doses of X-rays and different times of culture.
It appeared that most if not all oocytes which were in MII on day 11 had been stimulated to divide *in vivo*, before removal of the ovaries. The lowest dose of radiation which was able to induce such a rapid stimulation of the first meiotic division in meiotically competent

oocytes was apparently in the range of 0.25- 0.50 Gy. Further experiments showed that all MII oocytes had been eliminated from the ovaries by day 15, i.e. 2 days before ovulation, and been replaced by others that were contained in smaller follicles at the time of irradiation. Those were in diplotene and were able to evolute normally to the MI stage, when cultured for 6 h.

Dose (Gy)	Day of irradiation	Oocytes examined	N^{ber} abnorm. (%)	N^{ber} with transloc. (%)	N^{ber} with break/frag. (%)
0	-	78	0 (0)	0 (0)	0 (0)
1	3	132	5 (3.78)	2 (1.51)	4 (3.03)
2	3	147	15 (10.20)	5 (3.40)	11 (7.48)
1	10	122	50 (40.98)	44 (36.06)	27 (22.13)
2	10	120	89 (74.16)	76 (63.33)	53 (44.16)

TABLE I
Chromosome aberrations detected in metaphase I oocytes irradiated on days 3 or 10 of the estrous cycle (examination 1 week or directly after irradiation).

DISCUSSION

Our data evidenced a dramatic increase in the radiosensitivity of the guinea-pig oocyte during a time interval of only 1 week : oocytes irradiated at the beginning of the estrous cycle (day 3) had a low frequency of chromosome aberrations, while those irradiated at the middle of the estrus cycle (day 10, when growing Graafian follicles are clearly visible at the surface of the ovaries) exhibited heavy chromosome damage.
Earlier, Oakberg and Clark (4) noted a marked decrease in the number of large oocytes, during the first ten days following irradiation of guinea-pig ovaries by 2 Gy. In agreement with this, we found that oocytes irradiated at the middle of the estrous cycle were eliminated from the ovaries in a few days, after their evolution to the MII stage. The stimulation of the first meiotic division by radiation required less than 24 h after doses of 1 or 2 Gy, and was probably due to a very rapid killing or inactivation of follicular cells which normally exert an inhibitory effect on this process. When examined on day 15 of the estrous cycle, i.e. 2 days before ovulation, ovaries of animals irradiated on day 10 again showed high numbers of growing Graafian follicles. Oocytes contained in those were in diplotene, and behaved normally in culture, reaching the MI stage after 6 h. Thus, the ovaries had apparently compensated for the loss of all large follicles by an acceleration of the maturation and growth of smaller oocytes.
An increase in the number of corpora lutea per female was noted by Cox and Lyon (5), in guinea-pigs irradiated with 4 Gy (the only dose tested) between days 6-12 of the cycle and mated at the first estrous post-treatment. This effect was attributed to the ovulation of abnormally high numbers of oocytes in irradiated animals ("superovulation effect"). Concommitantly, there was an apparent increase of the preimplantation loss in these animals, an effect possibly resulting from the induction of dominant lethals by radiation. It will be important, therefore, to have a precise idea of the level of chromosome damage present in oocytes surviving an irradiation near the middle of the estrous cycle, and able to reach ovulation after an accelerated growth. Such study is in progress in our laboratory.

Data on the mouse indicated that, in this species, the radiosensitivity of the oocytes is low during weeks 1-2 before ovulation, if one excepts the 12 hours immediately preceding ovulation (6,7). On the basis of our results, it can be concluded that the radiosensitivity of the growing guinea-pig oocyte (1 week before presumed ovulation) is clearly much higher than that of the corresponding stage in the mouse, both in term of sensitivity to killing and to induction of chromosome aberrations.
Recent results have also shown that the mouse immature oocyte was sensitive to the induction of translocations and other chromosome aberrations by radiation (8-10), and data obtained in our laboratory allowed to extend this conclusion to the immature oocyte of the guinea-pig (3). However, the results of Straume et al. (10) suggested that the radiosensitivity of the mouse immature oocyte should be rather high, while our data indicated that the radiosensitivity of the guinea-pig immature oocyte is rather low.

All together, our results suggest that important differences exist between the mouse and the guinea-pig, with regard to the radiosensitivity of their female germ cells. This underlines the necessity of performing studies in other mammalian species, in order to better define the genetic risks associated with an exposure of women to radiation.

ACKNOWLEDGEMENTS

This work was supported by Research Contracts from the European Communities (Contract No. FI3P-CT920005) and from the Belgian Government (Contract No. ST/52/030)

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MODIFICATION OF RAT INTESTINAL MUSCARINIC CHOLINERGIC RECEPTORS BY IONIZING RADIATION

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ABSTRACT

Evidence exists for increased fluid and chloride secretion induced by ionising radiation. Acetylcholine via stimulation of muscarinic epithelial receptors may be implicated in such increases. This study addresses the question of whether ionising radiation modifies cholinergic muscarinic receptors.
Male Wistar rats were exposed to total body gamma irradiation (8 Gy; 1.3 Gy.min^{-1}; ^{60}Co source). Plasma membranes were prepared from small intestinal mucosal scrapings and marker enzyme activities (sucrase, Na+/K+-ATPase) measured. Muscarinic receptor binding characteristics (Kd, Bmax) were determined using the non-selective muscarinic antagonist ^{3}H-QNB (quinuclidinylbenzilate).
Sucrase and Na+/K+-ATPase activities are maximally decreased 3 and 4 days (D3, D4) post irradiation (3 fold). Myeloperoxidase activity is markedly reduced (10 fold at D4) and is still lower than control values 21 days after irradiation (5 fold). Two ^{3}H-QNB binding sites are observed for control rats. For the high affinity site, Kd is decreased up to 7 days after irradiation with a maximum at D4 (4 fold decrease) whereas Bmax is unchanged. For the low affinity binding site, both Kd and Bmax are reduced. The significance of these sites remains unclear.
The increase of the affinity of the high affinity binding site is in agreement with an increased response to cholinergic stimulation, and so with increased fluid and chloride secretion. These results suggest a possible implication of a dysfunction of cholinergic regulation in irradiation-induced diarrhoea. Thus it is of interest to determine the mechanisms by which radiation can modify muscarinic mucosal receptors.

INTRODUCTION

Acetylcholine, released from parasympathetic nerve endings, has a major role in the control of intestinal function. In particular, increased stimulation of muscarinic epithelial receptors may result in fluid hypersecretion. (1). Several reports show that ionising radiation induces intestinal fluid and chloride secretion (2,3).
In addition Otterson (4) observed a decrease of the amount of acetylcholine esterase (degradative enzyme) in the mucosa after irradiation. This suggests that irradiation may alter cholinergic regulation of mucosal functions so leading to increased fluid and electrolyte transport.
This study addresses the question of whether ionising radiation modifies the characteristics of cholinergic muscarinic receptors in the mucosa of rat.

METHODS

Male Wistar rats (Laboratory C.E.R.Janvier), weighing between 280 and 300g, were exposed to whole body gamma irradiation and received a dose of 8 Gy (1.3Gy.min^{-1}, ^{60}Co source). Control rats were sham-irradiated during the same period. The small intestine was removed under anaesthesia (sodium pentobarbital: 60mg.kg^{-1}) up to 21 days after irradiation and animals euthanised following an overdose of anaesthetic. Small intestinal plasma membranes were prepared from mucosal scrapings (both ileum and jejunum were pooled for each rat) and kept at -80°C. Enzyme activities were determined by spectrophotometric assays : Sucrase (5), Na^+/K^+-ATPase (6).

Determination of muscarinic receptor characteristics was performed using a non-selective muscarinic antagonist ^{3}H-QNB (quinuclidinylbenzilate). Membrane (200µg protein) were incubated with increasing concentrations of ^{3}H-QNB ranging from 50pM to 15nM. Non specific binding was determined in the presence of atropine (50µM). Separation of bound and free ligand was by rapid filtration. The experiments were performed 1, 3, 4 and 7 days after irradiation on either irradiated or sham-irradiated rats.

Analysis of binding data was by Scatchard analysis.

Statistics: A one way Anova test was used to test populations of control rats and for enzyme activities experiments, and a one way Anova Dunnett's test for binding experiments. (ns = non statistically significant; * = $p<0.05$)

RESULTS

Determination of enzyme activities

Sucrase is an enzyme associated with the apical membrane and is essentially located on the top of the villi. The results presented in the Table 1 show that sucrase activity was greatly decreased to 17% of control values 3 and 4 days after irradiation ($p<0.05$). Nine and 21 days after irradiation the sucrase activity went back to control values (ns). In parallel, the activity of the basolateral enzyme, Na^+/K^+-ATPase was determined and it is clear that irradiation modifies the Na^+/K^+-ATPase activity with a time dependent pattern similar to the pattern observed for the sucrase. Na^+/K^+-ATPase activity was unchanged at D0 but falls to 66 and 60% of basal level 3 and 4 days after irradiation respectively ($p<0.05$), and returned to basal values at D9 (ns). At D21 a second decrease in activity of 72% is observed.

Table 1: Determination of sucrase and Na^+/K^+-ATPase activities

Day after irradiation or sham irradiation	Sucrase (U/mg protein)				Na+/K+ ATPase (U/mg protein)			
	control rats	Irradiated rats	n	p	control rats	Irradiated rats	n	p
D0 at 6hours	1.26 ± 0.12	1.02 ± 0.13	2	ns	0.98 ± 0.06	0.99 ± 0.04	2	ns
D3	(n=19)	0.22 ± 0.08	5	*	(n=19)	0.33 ± 0.14	5	*
D4		0.22 ± 0.02	4	*		0.39 ± 0.06	4	*
D9		1.01 ± 0.08	6	ns		0.73 ± 0.09	6	ns
D21		0.79 ± 0.09	4	ns		0.27 ± 0.09	4	*

Characterisation of muscarinic receptors

Analysis of 3H-QNB binding data of control rats indicated the presence of 2 binding sites, one of high affinity and low capacity and one of lower affinity but higher capacity. No difference was observed for control animals during the 7 days following the simulation of irradiation (ns) and the results obtained for all control rats were pooled.

Figure 1 summarises the results obtained for binding characteristics tested up to 7 days after an 8 Gy irradiation. The Kd of the high affinity binding site (Kd1) is decreased to 24.4% of the basal value at D4 after irradiation. This decrease is still observed at D7 (25.8, $p<0.05$). No significant change in the number of sites (Bmax1) was observed for this high affinity binding site (ns) even 4 and 7 days after irradiation. For the lower affinity binding site, both Kd (Kd2) and Bmax (Bmax2) were decreased at D4 (11.5% for Kd and 14.1% for Bmax). Seven days after irradiation only the Kd2 of this site was significantly decreased (13.1%, $p<0.05$).

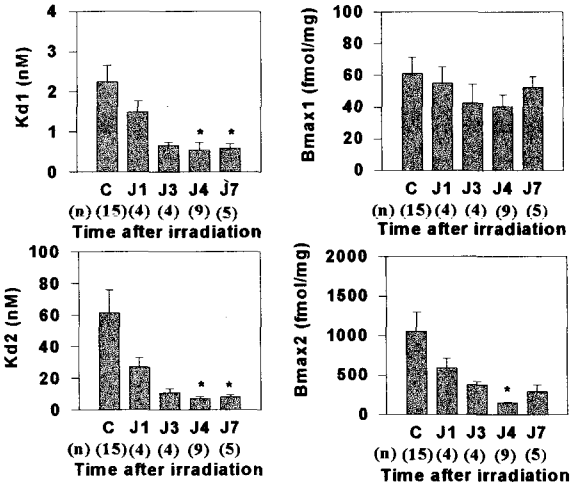

Figure.1: Characterisation of 2 muscarinic binding sites

CONCLUSION

The present data show that gamma irradiation markedly attenuates both apical (sucrase) and basolateral (Na^+/K^+-ATPase) enzyme activities. This may reflect a decrease of number and size of villosities linked to mucosal denudation (7). Such changes may explain decreased nutrient absorption following irradiation.
The modification of muscarinic receptor characteristics, in particular the effect on the high affinity binding site is in agreement with an increase of sensitivity of the small intestine to cholinergic regulation. Thus irradiation may affect regulatory processes of water and electrolyte transport. The effect on the second binding site is more difficult to interpret in terms of increased sensitivity. More experiments are required to see whether changes in both sites are also related to increased fluid and electrolyte transport after irradiation. To this end further experiments are in progress in order to test the functional capacity of muscarinic receptors using isolated intestine placed in Ussing chambers.

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INFLUENCE OF GAMMA IRRADIATION ON A GASTROINTESTINAL PEPTIDE: NEUROTENSIN

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INTRODUCTION

Exposure to ionising radiation induces gastrointestinal dysfunction. These injuries may be classified into two main categories: the initial syndrome characterised by acute effects such as nausea, vomiting, anorexia and diarrhoea observed during the first 24 h after irradiation for doses as low as 1 Gy. Later effects and the onset of the gastrointestinal syndrome (GIS) are seen at higher doses (>10 Gy) and include intestinal haemorrhages, diarrhoea and septicaemia, the severity of which depends on the nature of irradiation and on the dose received. These modifications of digestive system function are associated with a general deterioration in the epithelial cell-lining resulting in loss of functional capacity and integrity (1). In addition epithelial ion transport is modified inducing an electrolyte imbalance with a decrease in water absorption combined with excessive secretion and loss of sodium and chloride (2). On the other hand, ionising radiation dramatically affects gastrointestinal motility resulting in modified frequency and amplitude of contractions (3). The precise contribution of each of these factors in the manifestation of irradiation-specific effects like radio-induced diarrhoea is not known. The aim of this study was to investigate the effects of whole body gamma irradiation on neurotensin (NT), a gastrointestinal neuropeptide implicated in gut electrolyte transport and motility regulation. This study carried out in the rat, focused on plasma NT levels as well as on colonic smooth muscle NT specific receptors.

METHODS

Male Wistar rats (250g) were exposed to whole body gamma (^{60}Co source) irradiation with doses ranging from 1 to 8 Gray (Gy) at a dose rate of 0.25 Gy.min^{-1} or sham-irradiated. Peripheral blood samples and the colon were removed under anaesthesia (sodium pentobarbitone 60mg.kg^{-1}) from 1 to 10 days after irradiation. Plasma NT was measured by radioimmunoassay (Amersham) and plasma membranes were prepared from the colon smooth muscle after removal of the mucosal layer. Analysis NT receptor binding sites was carried out on these membranes 3 days after irradiation and the Kd and Bmax determined using ^{125}I-labelled NT in the presence of increasing concentrations of unlabelled NT. Specific binding was measured from the difference between ^{125}I-NT binding in the presence and absence of 1µM unlabelled NT.

RESULTS

Plasma NT levels increased rapidly 24 hours after gamma irradiation with a peak of 32.2 ± 7.6 fmol/ml ($n=6$; $p<0.01$) for a dose of 8 Gy as compared with sham-irradiated animals (5.9 ± 1.0 fmol.ml^{-1}) . The maximal secretion was observed 2 days after irradiation and remained elevated for 4 days following a dose of 6 Gy (42.7 ± 3.5 fmol.ml^{-1}; $n=6$; $p<0.005$). The NT levels decreased progressively and returned to basal values at day 7 for all doses (Figure 1).

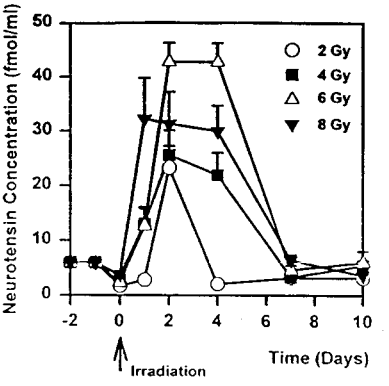

Figure 1: Variation in blood neurotensin levels after gamma irradiation in the rat. The NT concentration was measured by radioimmunoassay after irradiation in the dose ranging from 2 to 8 Gy. Results are means ± SEM; n=6

Increased plasma NT levels appeared to be dose dependent at day 2 in the dose range 2-8 Gy (Figure 2).

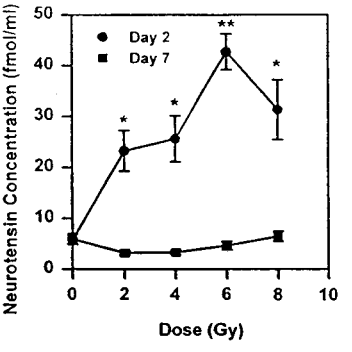

Figure 2: Dose response curves for blood neurotensin levels.
Results are means ± SEM; n=6; *p<0.005, **p<0.001

In parallel, NT receptor binding in colonic smooth muscles was modified. Three days after 6 Gy irradiation, NT specific receptors were characterised by the appearance of a second class of site of low affinity and high capacity. An increase (3 fold) of the total capacity of binding sites was observed as compared with controls (Table 1).

Table 1: Binding sites capacity

	Control	6 Gy Day 3
Affinity Kd (nM)	Kd=0.66 ± 0.04	Kd1= 0.28 ± 0.03 Kd2= 4.05 ± 0.38
Capacity Bmax (fmol.mg^{-1})	Bmax= 86 ± 4	Bmax1= 60 ± 9 Bmax2= 316 ± 36

Data are expressed as mean ± SEM; n=3

CONCLUSION

Whole body gamma irradiation of rats in dose range of 2 to 8 Gy results in a dose dependent increase in plasma NT levels. Variation of gastrointestinal peptide plasma profiles may provide new and better site specific indicators of irradiation induced gut damage. This increase is concomitant with and may be related to a modification of NT specific receptors in colonic smooth muscle. These new observations demonstrate that ionising radiation at doses of less than 10 Gy induces modifications of some gastrointestinal peptide functions.

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PROLIFERATIVE ACTIVITY OF CULTURED RAT GLIAL CELLS AFTER IRRADIATION OF PROGENITORS OR MULTIPLYING AND DIFFERENTIATING GLIAL CELLS.

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INTRODUCTION

Both glial and mesodermal cells participate in the formation of scar tissue after injury in the adult central nervous system. According to most authors, the glial response to brain injury is present in fetus and new born animals but incomplete. The mature scarring would be effective only on day 8 after birth in rat (1). The GFAP+ (Glial acidic Fibrillary Protein) cells in the rat fetuses cortex were first detectable on day 19 (2). Are there, early in gestation, target cells sensitive to the deleterious effects of irradiation?

MATERIAL and METHODS

Irradiation (1 Gy) were carried out on 15 or 21 days pregnant rats or on nerve cells just after the beginning of culture. On day 1 after birth, the cortex cells were isolated and cultured in medium with 10% fetal calf serum (FCS) until 30% of confluency. The cells were synchronized in the G1 phase of the cell cycle by a 48 hours culture in medium with 0.1% FCS. This arrest was verified by BrdU incubation. The medium was then replaced by one with 10% FCS for the cells to enter the S phase and 20 hours later ^{3}H thymidine or BrdU were added for 6 hours. The radioactivity was measured in washed cells and BrdU+, GFAP+ cells by immunohistochemistry were counted on 5 microscopic fields. The MTT survival test was carried out on all cultures.

RESULTS

The culture with 10% FCS promoted the survival of astrocytes (GFAP+) which represented 99% of the cells. After irradiation, the GFAP+ cells survived (MTT test) as well as the controls (figure 1). The cell cycle arrest with 0.1% FCS was effective as only 7% of cells were BrdU+. After return to 10% FCS, 35% of control glial cells were BrdU+. The proliferative activity of cells from fetuses irradiated on day 15 was increased by 100% and on day 21 to about 50% of the control

values (figure 2). The proliferative activity of the cells irradiated *in vitro* was similar to the control values.

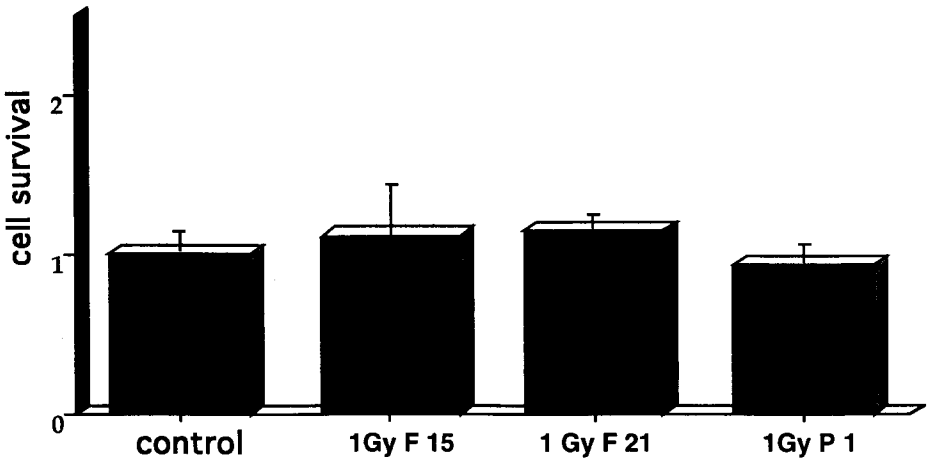

Figure1: Cell survival measured by MTT test relative to control survival. P 1: isolated nerve cells irradiated on day one postnatal; F 15 or F 21: nerve cells isolated from 1 day old animals exposed when they were 15 or 21 day old fetuses

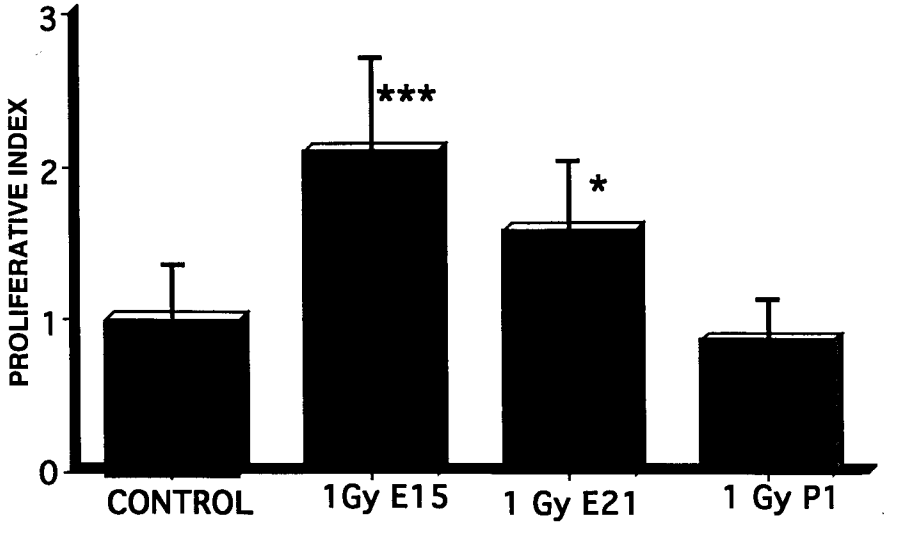

Figure 2: GFAP+ glial cells proliferation relative to the control one.
*** $p < 0.001$,
* $p < 0.05$

DISCUSSION

The neurons death induced by irradiation was most effective on gestational day 15 when these cells were dividing actively (3) but glial cells present as progenitors were radioresistant (unpublished results) . On the other hand, few neurons were dividing on day 21 and few were killed by irradiation. In spite of glial cells division from progenitors, these cells were rather radioresistant in comparison with the proliferative neurons (unpublished results). The increase of proliferative activity induced by the irradiation on day 15 of gestation was very interesting because at that time the glial cells were mainly as progenitors. The increase in proliferative activity was lower on day 21 when glial cells started dividing. The progenitors of glial cells could be the target of ionizing radiation to induce proliferation. We can hypothetize that the exposure on day 15, the most radiosensitive period for the cortex neurons, kill a great number of these cells which may release directly or indirectly (via mesenchymal cells) factors like FGF (4) that stimulate progenitor glial cells to proliferate. On day 21pc, the decrease of both neurons radiosensitivity and the number of progenitor glial cells (sensitive to released growth factors) could explain the lower increase of the proliferative activity of the glial cells. On day one after birth, no change was observed with the cells irradiated *in vitro*. This could be either the consequence that glial cells and neurons had lost their close relationships or glial cells could not respond to irradiation at this age.

CONCLUSION

This increase of proliferative activity after prenatal irradiation is very important because:

- the gliomas incidence could be enhanced
- a postnatal brain injury could recruit more reactive glial cells to proliferate.

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PAPER TITLE RADIATION-INDUCED ALTERATIONS OF CELL MEMBRANES AND BLOOD CELLS: THE PHYSIOLOGICAL CONSEQUENCES AND DIAGNOSIS VALUE

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RADIATION-INDUCED ALTERATIONS OF CELL MEMBRANES AND BLOOD CELLS: THE PHYSIOLOGICAL CONSEQUENCES AND DIAGNOSIS VALUE

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The blood-forming organs and blood are the most sensitive targets to radiation damage. A number of structure-function parameters of the cells of blood and blood-forming organs in different time intervals after acute sublethal doses and chronic irradiation during one year under the background of radioactive iodine and cesium injections were determined. These were the intensity of intrinsic fluorescence of cellular components closely related to the metabolic activity of the cells; the fluorescence intensity of membrane-bound probes, pyrene, ANS and merocyanine 540 (Imc) which characterize the properties of the membrane structure; Con A binding and velocity of lectin-induced agglutination of the cells (Vagg) which were changed in bone morrow cells (BMC), splenocytes, thymocytes and peripheral blood lymphocytes (PBL) after irradiation. The changes in Imc and Vagg were revealed at lover doses than it was found for other parameters. BMC and PBL were the most sensitive to either acute or chronic irradiation.
The analysis of the data obtained testify that the observed changes are mainly due to their shifts in population composition of the cells. The perspective of application of the spectral methods for early elucidation of radiation-induced cancerous and immunodeficient changes in the organism are discussed.

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PAPER TITLE CHANGES OF STRUCTURAL AND FUNCTIONAL PARAMETERS OF BLOOD AND BLOOD-FORMING ORGANS CELLS MEMBRANE UNDER IRRADIATION OF ORGANISM

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CHANGES OF STRUCTURAL AND FUNCTIONAL PARAMETERS OF BLOOD AND BLOOD-FORMING ORGANS CELLS MEMBRANE UNDER IRRADIATION OF ORGANISM

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Cell membranes are one of the high sensitive to radiation targets. In this study structural and functional changes of membranes of rat thymocytes, splenocytes and peripheral blood lymphocytes (PBL) after acute treatment of animals by γ-irradiation in doses 0,25-2,0 Gy were investigated. For fluorescent intensity of membrane probe merocyane 540 (IMc),binding with cells of concanavalin A (Con A) and velocity of Con A-induced agglutination (*V*agg) double-phase dependencies on time after irradiation was shown for all types of cells. In first 1-3 days after irradiation IMc, *V*agg and lectin binding to cells were increased, but in 5-10 days they turned to norm. Under lowering of irradiation dose the effect values decreased and terms of their appearance increased. Both strongly pronounced effects on PBL and correlations among immunological characteristics and biophysical test give the opportunity to use the last one for express-control functional-important radiation changes of immunocompetent cells. From comparison of the data described with results received in vitro we can make a conclusion that observed changes were caused by the influence of radiation on population composition of lymphocytes. Therefore, the fluorescent-microscopic study of cells seems to be very perspective for the early detection of radiation damages in blood system. Application of the method gives an opportunities for carrying out of high-sensitive express-control for blood system state during radiation influence and efficiency of used radioprotective means.

β-CAROTENE AND VITAMIN PROTECTION OF MOUSE THYMUS AGAINST RADIATION EFFECTS

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Devastation of lymphoid organs is the promptest and the most sensitive response of organism to ionising radiation. Lympholytic action of this factor is followed by violation of immunity system (1), thus increasing the hazard of various diseases. An analysis of effects protective against degradation of lymphoid tissue is a promising test in a search for effective means enhancing the resistance of organism to radiation action. It has been reported that β-carotene, a provitamin A having antioxidant properties, is very promising as a radioprotector (2). As β-carotene is water-insoluble, it is preferable to use its water-dispersed forms with addition of α-tocopherol, tocopherol acetate (vitamin E), ascorbic acid (vitamin C) and its derivatives, proteins and emulsifying agents. These additions allow to achieve the stability of the water-dispersed form and improve the assimilation of β-carotene in organism (3). Below we describe the data on radioprotective ability of β-carotene and vitamins.

White mice (genetical type Swiss, males, weight 18-22 g) were exposed to ^{60}Co γ-rays at 100 rad/min. 24 h later, the mice were sacrificed, and the number of cells in thymus was determined. As a radioprotector, we tested Vetoron, a water dispersed commercial preparation (AQWA-MTD, Moscow, Russia) containing 2% β-carotene and vitamins E and C, each in concentration of 0.67%. The animals were fed with Vetoron enriched feed for 7 days prior to irradiation. The Vetoron enriched feed (31.2 mg of β-carotene per 100 g of the feed) was prepared daily.

The obtained data are shown in the Figure. At 24 h post exposure, a dose-dependent decline of the number of cells in thymus was observed. This dependence can be described, in semilogarithmic co-ordinates, by the equation

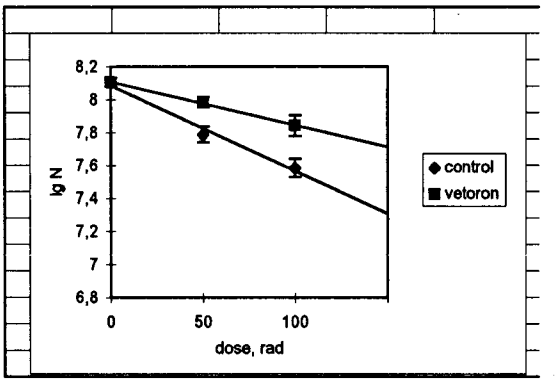
$$\lg N = (8.102 \pm 0.026) - (0.0053 \pm 0.0007)\, R$$

where N stands for the number of cells in thymus of exposed control animals, R is radiation dose (in rads). Preliminary feeding of mice with the Vetoron enriched feed exerted a significant ($P<0.05$) protective action. The dose dependence of thymus devastation value obtained for this case can be described, in semilogarithmic co-ordinates, as

$$\lg N_v = (8.106 \pm 0.025) - (0.0026 \pm 0.0006)\, R$$

where N_v stands for the number of cells in thymus of exposed animals fed with Vetoron enriched feed, R is radiation dose (in rads). Comparison of the slopes of the dose curves obtained for control animals and for animals fed for a week with Vetoron enriched feed shows that this preparation reduced the radiation induced decrease of the number of cells in thymus by 50%.

Data on β-carotene induced increase of viability of irradiated animals can be found in literature (4, 5). Also it has been shown that decrease of concentration of β-carotene and other antioxidants, including vitamin C, in blood enhanced the hazard of cancer and cardio-vascular diseases (2, 6). Our results, together with these data, allow to conclude that composite water



Action of the Vetoron preparation on the number N of cells in thymus of the irradiated mice.

dispersed preparations containing β-carotene and vitamins E and C can be used for protection of organism against action of radiation and other unfavourable environmental factors.

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WAYS OF PHARMACOLOGICAL PROPHYLAXIS OF STOCHASTIC AND DETERMINISTIC EFFECTS OF CHRONICAL RADIATION EXPOSURE

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INTRODUCTION

The prophylactics of late effects of exposure is the actual medico-social problem, because now there are large groups of persons who were exposed during occupational contact and living on territories contaminated by radionuclides. Most probable consequences of external and internal chronic influence of radiation may be the increase of malignant tumour frequency, the development of secondary myelo- and immuno-depressions, the earlier forming of sclerous and destructive processes, and the acceleration of senescence. The role of damages in immune system was not yet understood in pathogenesis of the late effects of radiation, but there are evidences that the decreasing of the immunologic supervision in period of forming the consequences of radiation influence enables to realize the cancerogenic effect of radiation (1,2).

The purposes of this investigation are to decrease the frequency or to prevent the development of radiation consequences dangerous for health and life by using the method of modification of radiogenic damages in hemopoietic and immune systems by applying the pharmacological preparations with immunomodulating effects. The ivestigation tasks include: the study of modifying influence of pharmacological substances with different mechanisms of effect: myelopid (immunomodulating, and regulatory), β-carotin, *Calendula officinalis* (immunomodulating, and antioxidant), lipamid (detoxicating); the separate or complex applications of these substances; and the development of the optimum medico-prophylactic schemes. The advantages of these indicated preparations in comparison with the known (T-activin, thymogen, cytokines, etc.) are the absence of contraindications and the possibility to use *per os*.

MATERIALS AND METHODS

The work was conducted on 2 species of laboratory animals (*Wistar* rats, and *CBA* mice) under three models of chronic radiation damages, following the incorporation of plutonium-239 (polymer nitrate or citrate, single injection by amount of 92.5 kBq/kg), and of tritium oxide (HTO, daily drinking water intake, total dose of 9 Gy for 6 or 3 months, dose rates of 4.5 and 9.2 cGy/day), or the long-time fractionational external γ-exposure to cesium-137 (dose of 6 Gy for 6 weeks, or for 3 weeks with dose rate 20 cGy per day for 3 hours, the exposure during 5 day per week with interval of 3 or 10 days.

The medico-prophylactic courses were begun in 0.7, 2, or 5 months after beginning of external exposure; in 1 or 6 months after HTO intake; and in 0.2, 2, or 6 months after plutonium-239 intake. Myelopid (MP), produced at the Immunology Institute, Moscow, was injected hypodermicly one-time per week or per month giving 0.6 mg/kg or 2.0 mg/kg, respectively, and making 3-14 injections per 1 or 2 courses. β-carotin (produced by "Vitamins" SPJ, Moscow) was received by animals together with foods (curds, olive oil) in amount of 7 mg/kg during 4 months. The *Calendula officinalis* was received with drinking water by mice in amount of 12 ml/kg, and by rats in amount of 5.6 ml/kg. The tablets of lipamid were taken with the crude for mice in amount of 15

mg/kg per day, and for rats in amount of 7.2 mg/kg per day. The complex of myelopid, lipamid, and Calendula was received by animals in the same amount, as separately.

The evaluation of modifying effects of medicinal preparations was conducted after exposure and 1 or 2 courses of immuno-therapy for the following criteria: the number of stem cells (SFU_{s-8}, SFU_{s-12}); of karyocytes in bone marrow and in immunity organs; of T- and B-lymphocytes, plaque-forming units in spleen (PFU - integral index of humoral immunity); of peritoneal macrophages, by taking the functional activity of lymphocytes, macrophages, natural killers, RBTL, the content of leukocytes in blood as well as by results of autopsy study after animal death.

The processing of data was conducted by general statistical methods with estimate of statistical reliability of differences equal to $p \leq 0.05$, as threshold of unerring prognoses.

RESULTS AND DISCUSSION

Chronic radiation damage under indicated models of influence may be characterized as having the phases. The traits of damages in hemopoietic and immune systems depended from kinds of ionizing radiation or from plutonium compositions, dose rate, and the total dose of exposure, the duration of influence, evenness of exposure. The development of myelodepression, hypoplasia of lymphoid organs, and immuno-deficiency state were being observed under impact of HTO or plutonium-239 polymer nitrate. It was found out the deep suppression of humoral immunity (decrease of PFU up to 10% of intact control), the decreasing of activity of cellular immunity effectors by 10-30%. The damages of immune processes were less expressed, and recovered to normal values in 3-6 months of observing in case of external exposure or of plutonium citrate injection. For all kinds of ionizing radiation, the most deep and stable changes (up to 10% of norm and below) were indicated in the stem cell compartment under considering all studied models of radiation damage, and full recovery of these cells' number was not registered for period of observation.

Leukopenia was not long after all kinds of radiation influence, the lymphopenia and hypoplasia of lymphoid tissue depended from the decreasing of population of both T- and B-lymphocytes. The value of proliferative ability damage increased with increasing dose of even exposure.

The application of pharmacological preparation under long-term influence of radiation allows to decrease the depth of immuno-deficiency and myelodepression, and all the tested medicinal preparations have immunomodulating effect: the increase of stem cell content at 1.5-2.5 times; the increase of the antibody-producers at 1.5-4 times; the decrease of hypoplasia of lymphoid organs at 1.5-2.5 times; the increase of functional activity of cellular immunity effectors by 10-50% in comparison with exposed animals that were not treated). The stimulating effect of immunomodulators was more expressed under significant damage of radiosensitive organs and tissues, but the same effect was indicated in case of using substances when there was full recovery of immune reactions. It was found out that the indicated modulating effect of preparations was pronounced mainly in 2-3 months after medico-prophylactic course, and more constant influence of medicinal substances on recovery processes was registered in the stem cell compartment. The regulatory effect of myelopid was being observed during the recovery of immunity indexes, and it was higher than for the intact control. The most expressed effect was determined in case of complex application of preparations (myelopid, *Calendula officinalis*, and lypamid; or myelopid and β-caratin) for both external and internal exposures. Chronic effect of incorporated tritium or plutonium polymer nitrate caused the decrease of the average lifetime (ALT) by 10% and the increase of malignant tumour frequency at 2 times due to mainly the increasing

of outcomes of myeloleukemia, lung cancer (HTO), liver tumour, and ostersarcoma (plutoniume-239).

The medico-prophylactic course of myelopid provided the decrease of common number of malignant tumours at 1.5 times ($p<0.05$), including the frequency of lung cancer that decreased at 1.7 times up to spontaneous level; the myeloleukemia decreasing at 3.6 times; and tumours of liver and ostersarcomas decreasing up to the spontaneous level. The receiving of β-carotin enabled to decrease at 1.5 times (lower than spontaneous level) the frequency of lung cancer, and at 3 times the myeloleukosis induced by tritium. In case of fractionational γ-exposure (dose of 3 Gy), the decrease of ALT was not find out, however, the malignant tumour frequency increased at 6 times due to the increase of leukemia and liver cancer. Myelopid applications, separately or in complex with β-carotin, led to the decrease of leukemia frequency up to spontaneous level, and of liver cancer at 2.5 times.

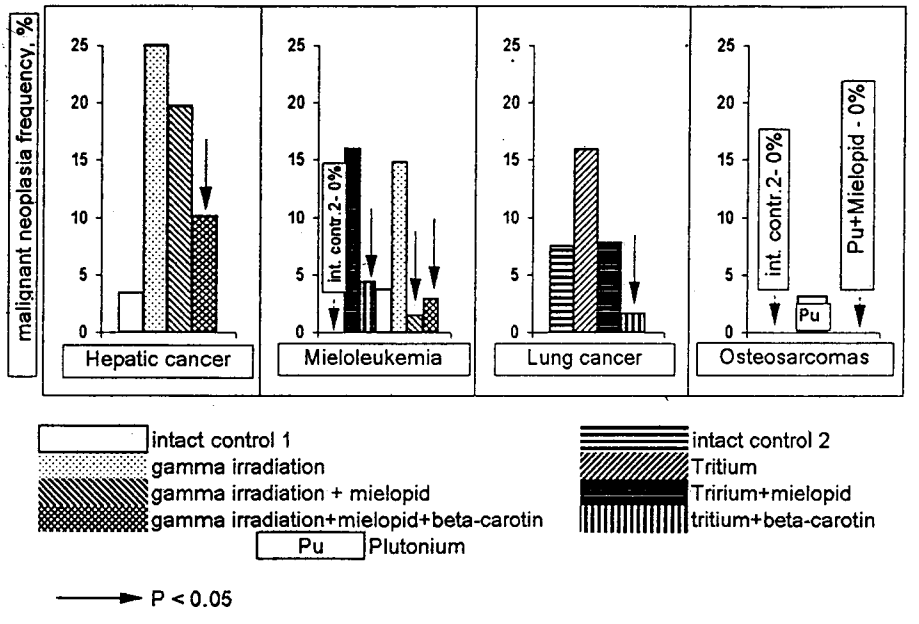

Figure 1

Pharmacological prophylaxis of malignant neoplasia after internal (Pu and Tritium) and external (Cs-137) irradiation in mice

Thus, pharmacological preparations with immunomodulating effect may decrease the level of damages of immunity organs, promote the more full recovery of hemopoietic and immune systems, and these functions, increasing the immune control that allows to decrease the malignant tumoural growth or to prevent the developing of the radiogenic malignant neoplasms.

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PAPER TITLE EFFECT OF DIETARY SELENIUM INTAKE ON RADIATION INDUCED OXIDATIVE STRESS IN ERYTHROCYTES

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MAJOR SCIENTIFIC TOPIC NUMBER 2. (see page 7)

The key mechanism responsible for producing oxidative damage by ionizing radiation are free radicals and H_2O_2 formation. Our recent findings suggest that supplementation with selenium enriched yeast (SeY) could in rats exposed to ionizing radiation increase survival and prevent or minimize changes in content and distribution of Se, Cu, Zn and Mn in most of the rat tissues, as well as oxidative changes in blood. The exception is changes in tissue Fe content and distributions induced in SeY supplemented irradiated rats, that are not unidirective. Other studies indicate that Se compounds may initiate formation of reactive oxygen species and manifest toxicity when oxidative damage exceeds antioxidant defenses or the ability to form selenoproteins, selenoethers or elemental Se.

To study the effects of supplementation with SeY on oxidative stress induced by ionizing radiation and antioxidative defense systems we employed erythrocytes. Since erythrocytes are constantly circulated through tissues producing higher magnitude of free radicals and H_2O_2 flux they require a relatively high level of selenium containing enzymes, glutathione preoxidase (GSHPx), that is the main H_2O_2 catabolizing enzyme in them and other antioxidative enzymes. Male Wistar rats received selenium (0.5 μg/d) as SeY or pure yeast (Y) for 28 days before exposure to gamma rays (single dose of 4.2 Gy and 8.6 Gy from ^{60}Co source). Erythrocytes were analyzed for changes in the content of trace elements (Se, Cu, Zn, Mn and Fe), thiobarbituric acid reactive substances (TBARS), reduced (GSH) and oxidized (GSSG) glutathione, as well as the activity of enzymes, GSHPx, glutathion reductase (GR), catalase (CAT) and superoxid-dismutase (SOD) 8 and 30 days after the exposure.

Taking into account all the studied parameters in erythrocytes of the rats, it may be concluded that supplementation with SeY may reduce the changes induced by ionizing radiation in trace elements and alteration of the antioxidative defense. The ratio of reduction caused by supplementation with SeY depends on the involvement of enzymes in initiated antioxidant defense, related to the dose of radiation obtained and time.

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1996 International Congress on Radiation Protection
April 14-19,1996
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PAPER TITLE EFFECTS OF SUPPLEMENTATION WITH SELENOUS YEAST ON RADIOPROTECTION

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MAJOR SCIENTIFIC TOPIC NUMBER 2. (see page 7)

ABSTRACT (See instructions overleaf)

EFFECTS OF SUPPLEMENTATION WITH SELENOUS YEAST ON RADIOPROTECTION

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Ionizing radiation induced in the living system an increased production of free radicals, mainly oxygen reactive species, that could cause unrevesible damage of biomoleculs. Endogenous antioxidant defense system (AODS) developed in the aerobic cell with the role to protect against damaging reactivity of oxyradicals. Essential element Selenium is an important part of AODS.

In our study we investigated weather the supplementation with Se in a form of protein bound element, could exert an protective effect against harm of whole-body irradiation. A groups of male Wistar rats were exposed to gamma rays (single dose of 4.2 Gy from Co-60 source) and supplemented with Se-enriched yeast (0.5 ug/d for 4 weeks before and whole time after irradiation) or with pure yeast - controls . We analyzed the alterations in : amount of reduced (GSH), oxidized (GSSG) glutathione, Selenium, malondialdehyde-like products; activities of enzymes catalase (CAT), glutathione peroxidase (GSH-Px), superoxide dismutase (SOD) in red blood cells of rats 8, 30, 150 days and 8 months after irradiation.

The obtained results suggest that Se supplementation reduced changes in AODS, helping the organism to overcome harmful effects of ionizing radiation.

THE EFFECT OF SELENIUM AND GREEN ALGA MOMOTARU E-25 ON RADIATION CARCINOGENESIS AND LEUKAEMIA IN RATS EXPOSED TO IONIZING RADIATION

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INTRODUCTION

During the past decades several studies of radioprotective agents indicated that the antioxidative properties of selenium provide protection against free radical damage resulting from peroxidative conditions or ionizing radiation.

SELENIUM EXPERIMENT

In a long-term study four groups of 80 white mongrel rats each were exposed to radiation following the Chernobyl pattern. A control group of 80 animals was not irradiated. The fodder for three of the irradiated groups was supplemented with selenium containing yeast (1) to ensure a selenium content of 0.5 mg/kg, 1.5 mg/kg and 5 mg/kg, respectively.

During the 2.5-year study, the total cancer incidence was 7 (8.75%) cases in the control group. There were 53 (66.25%) cancer cases in the irradiated but not treated group, whereas there were 21 (26.25%), 16 (20.0%) and 19 (23.75%) cancer cases, respectively, in the selenium supplemented groups (Figure 1).

Over the same period there were 2 (2.5%) cases of leukaemia in the control group. In the irradiated but not supplemented group leukaemias totaled 13 (16.25%), while in the supplemented groups there were 4 (5.0%), 1 (1.25%) and 5 (6.25%) cases, respectively (Figure 2).

Radiation is mediated through formation of free radicals. When tissues are exposed to high energy radiation, most of the energy is absorbed by cell water. Radiation causes one of the oxygen-hydrogen covalent bonds in water to split, creating hydrogen and hydroxyl radicals. The hydrogen radical is the most reactive one known in chemistry, it can readily cause DNA damage. The selenium depending enzyme, glutathione peroxidaze, is supposed to be the endogenous scavenger of the hydroxyl radical (2).

Recently, an anticarcinogenic activity of beta-carotene and some other biological agents of vegetable origin has been reported. This effect is, however, unlikely to be associated directly with antioxidants. Besides, there is no evidence of any superiority of these substances to selenium as means of cancer prophylaxis.

In the study to follow we wished to investigate possible effects (synergism, antagonism) of selenium and Green alga Momotaru E-25, a biologically active strain obtained in Japan, when used in combination.

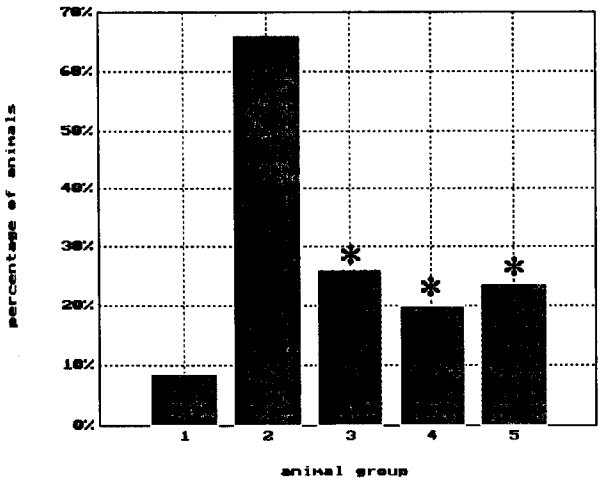

Figure 1. Total cancer incidence (pituitary, thyroid, breast, lung, ovary, uterus, kidney, liver plus leukaemia).
* $P < 0.05$ for comparison with group 2.

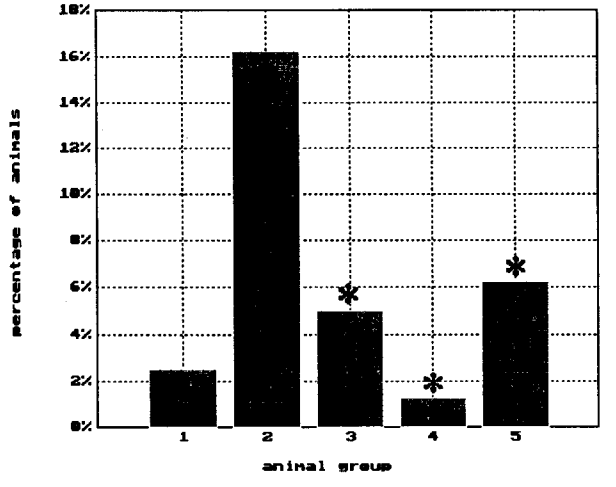

Figure 2. Leukaemia incidence.
* $P < 0.05$ for comparison with group 2.

SELENIUM PLUS GREEN ALGA MOMOTARU E-25 EXPERIMENT

The experiment involved 200 white rats divided into four groups. Group 1 was given a standard diet and no treatment (control). Groups 2, 3 and 4 were exposed to 6.5 Gy of gamma radiation and to incorporated iodine-131. After external irradiation was completed, groups 3 and 4 began to receive Momotaru E-25 and group 4 additionally got Selena (a Finnish preparation) at a dose of 30 ug selenium a day per capita. The animals were followed up for 2.5 years till their natural death. The exposed rats from groups 2, 3 and 4 had leukaemias, cancers and benign tumours, with the death rate increased.

Selenium supplementation of the diet (group 4) resulted in a longer lifespan (Figure 3). No antagonism between selenium and Momotaru E-25 was observed.

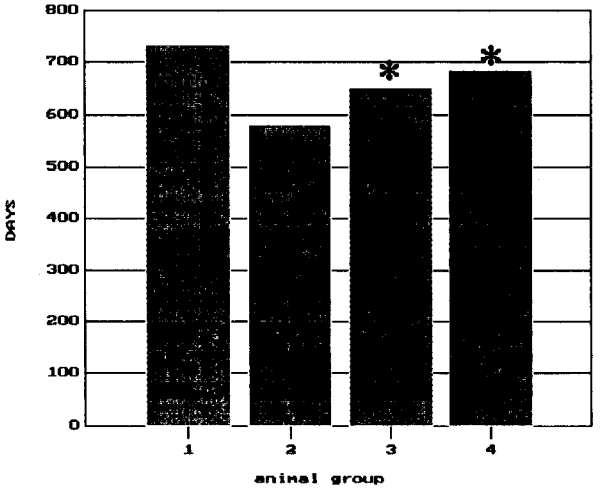

Figure 3. Rat lifespan.
* $P < 0.05$ for comparison with group 2.

CONCLUSION

A diet enriched with non-toxic doses of selenium and Momotaru E-25 caused a longer average lifespan and a 2 - 4 fold decrease of leukaemias and other malignancies, e.g. breast, thyroid and lung cancers, etc., at late times. In the groups with selenium and Momotaru E-25 supplementations, latent periods were longer than in the exposed control given no preparations.

The anticarcinogenic effect of selenium doses applied is on the average equal to the prevention of excess tumors from an effective dose equivalent of at least 1.0 Sv.

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RADIOPROTECTIVE AND RADIOTHERAPEUTIC PROPERTIES OF BIOTECHNOLOGICAL AGENT MD2

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INTRODUCTION

In recent years as the result of nuclear testing and accidents at nuclear power plants such as Chernobyl, etc. radiation exposure has become a major issue in various parts of the world. Experience of recent nuclear accidents has shown there is no effective treatment for patients exposed to doses of radiation that result in fatal hematopoietic failure and/or secondary infections (1). Therefore, agents that are effective when administered after irradiation, are of great interest.

In this study, the possibility of using biotechnological agent MD2 after lethal total body irradiation (TBI) and radiotherapy has been demonstrated. In addition, the considerable radioprotection without toxic effect can be obtained.

MATERIALS AND METHODS

Mongrel male mice were ip. injected with the 20% solution (in PBS) of MD2, 25 ml/kg, (0.5 ml/20 g mouse). Control mice were ip. injected 0.5 ml PBS per mouse. Dogs (18±2 kg) were orally administered with MD2, 1ml/kg, twice a day for 15 days following irradiation. The 20 year old patient admitted to the KMMA, was operated in 1976. Diagnosis: hard tissue sarcoma with desintegration localized on VI-VIIth ribs near the spine. The operation involved the resection of VI-VIIth ribs. The patient was orally administered with MD2, 0.4 ml/kg, three times a day. In addition, MD2 was applied to the skin. This supporting therapy lasted 3 months and was repeated after one year. *Irradiation.* Mice and dogs received 7.5 Gy (0.7 Gy/min) and 3.3 Gy (0.3 Gy/min) cobalt-60 TBI, respectively. Some of these experiments were carried out in 1976. The patient received 60 Gy/15/3 weeks on to hem after the operation. When 45 Gy was given, MD2 was administered. *The production of MD2* (polycomponent liquid, pH=5) is a high-technology biotehnological process which includes the bacterial fermentation of barries and vegetables (2). MD2 contains all essential amino acids; vitamins: "D", "B"; nicotinic, lactic and acetic acids; glucose; metals: Ca^{2+}, Mg^{2+}, Na^{+}, K^{+}, Fe^{3+}, Cu^{2+}; E-coli and lactobacillusis (2, 3). The agent has been used in the food (to increase bacterial fermentation), pharmaceutical industry, and in stock-raising in the former Soviet Union (4).

RESULTS

In the mice treated with MD2 (0.1 ml/mouse), no animals showed signs of impending toxicity during the 1-24 h preceding irradiation, and no signs other than those attributable to radiation damage were apparent subsequently. Maximum protection was attained after a single injection

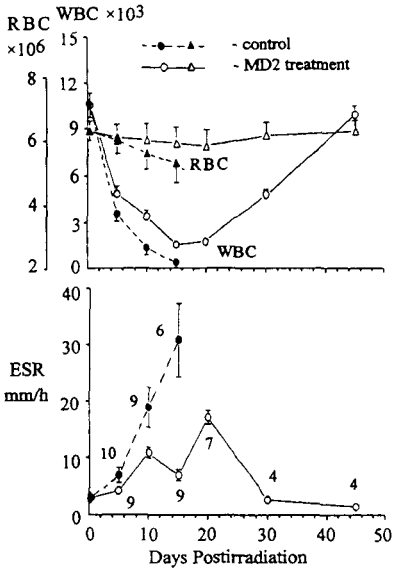
Table 1. Survival of TBI mice after treatment with ip. administered MD2.

Pre-TBI conditioning	No. of mice	No. of mice (%) Alive on Day 30-post TBI	Thymus, mg	Spleen, mg
PBS	80	4 (5)	21±4	98±15
MD2 1 h before TBI	40	18 (45)*	37±3*	210±36*
MD2 5 h before TBI	40	23 (57)*	43±4*	207±24*
MD2 24 h before TBI	43	13 (30)	-	-
MD2 1 h after TBI	30	7 (23)	-	-

* - $p < 0.05$ compared with saline.

of MD2 1-5 h before TBI (Table 1). Administration of MD2 24 h before or 1 h after TBI was less effective. Protected mice remained healthy until day 30 post-TBI when they were killed for examination of their thymus and spleen. These organs were preserved by MD2 because in protected mice the thymus and spleen were twice as heavy as those in the control (Table 1).

To assess the toxic effect of MD2, mice were injected ip. with the intact agent undiluted. The dose of MD2 at which about 50% of the mice died within 30 days was 200 ml/kg (4 ml/mouse). Consequently, the therapeutic index is equal to about 40 (4/0.1).



In the treated dogs with MD2 45-day survivors were 44% (n=4/9, the median survival time (MST) was 17.4±1.6 days), whereas in the control - 0% (n=0/10, MST was 14.7±1.5 days). With increasing rates of survival, MD2 administration ameliorated leukopenia, reduced the intensification of erythrocyte sedimentation rate (ESR) (Fig. 1). Post-TBI MD2 administration reduced the decline of white blood cells (WBC) and the gradual normalization of WBC was evident from day 20 onward. In addition, red blood cells (RBC) remined virtually unchanged. The treated dogs died on 15 day post-TBI, the day of nadir for WBC, when MD2 administration was stopped. In addition, ESR of survivals increased by 2.4 times on 20th day post-TBI and then decreased without MD2 administration. All dogs that survived lived at least one year. Finally, treated dogs had a better appetite, were much more active and their hair was not as badly damaged as in the control.

Fig. 1. WBC, RBC and ESR in irradiated dogs. Figures are the No. of dogs.

As for the patient, 14 days after surgery (29/09, 1976) a course of radiotherapy was carried out (13/10 to 5/11, 1976). The doses were increased during radiotherapy resulting in a decrease

Table 2. Changes in peripheral blood in patient admitted to the KMMA in 1976.

Date of examination	21/09	6/10	19/10	26/10	28/10	31/10	4/11	12/11	15/11
Erythrocyte count ×10³	5300	4300	4700	4500	4700	4600	4800	4800	5000
Leukocyte count	9100	11000	7700	6200	6500	6000	6500	6300	6600
Lymphocytes	2910	2970	1850	1670	910	640	1170	1260	1310
(%)	(32)	(27)	(24)	(27)	(14)	(10)	(18)	(20)	(21)
ESR (mm/h)	3	23	15	9	8	8	8	10	13

in the number of lymphocytes in the blood (Table 2). This effect was most pronounced. During radiotherapy the patient complained of fatigue, general malaise, loss of appetite, and some days before MD2 administration, of aggravated pain that was not relieved by drugs. To ameliorate the patient's state MD2 was administered (2/11, 1976). Two days later, sleep and appetite had normalized, lymphocytes had increased considerably (Table 2) and the pain had abated. On 14th day after MD2 administration (16/11, 1976), the patient was discharged home and continued to use MD2 according to the scheme mentioned above during the following 2,5 months. Then this course was repeated after one year. Now, 19 years after intervention with radiotherapy, the

patient feels well and does not complain of any long-term complications associated with radiotherapy or with the original disease. The patient graduated from Unversity immediatively after the operation, is now working, and continues to be mentally active.

DISCUSSION

At present, the majority of effective radioprotectors, e.g. WR 2721 or WR-3689 with DRF about 2-3, cannot be widely used, especially in humans, because of their toxicity (5). In contrast, MD2, having a therapeutic index of about 40, is not toxic at doses relevant to radioprotection. Doses of about 1-5 ml/kg body wt did not induce undesirable side effects such as diarrhea or nausea either in animals or in humans. Furthermore, as previously revealed, these doses both enhanced physical performance in mice and in humans, and improved the functional state of the central nervous system of the latter (6).

It should be noted that MD2 prevents hematopoietic organs from radiation-induced damage and appears to recover hematopoietic cells since post-TBI treatment with MD2 leads to a slower decline in cells of both myeloid and erythroid origin (Fig. 1). Nevertheless, the mechanism(s) underlying the MD2-mediated radioprotective effect is at present not understood and is a matter of speculation. On the one hand the fact that MD2 contains bacterial endotoxin, a well known radioprotector (5), may in part account for the radioprotective property of this agent. However, in contrast to MD2, the endotoxin is most effective when adninistered 24 h before irradiation and provides slight protection when administered shortly before or after radiation exposure (5). On the other hand MD2 has immunomodulator and bactericidal activity (3). This may account for the extreme effectiveness of post-irradiation MD2 adminstration. As an immunomodulator, MD2 may induce cytokines which are protective, and therapeutic agents against radiaton (1). Additional experiments are necessary to find out. Nevertheless, it is clear that other factor(s) are involved in the MD2-mediated therapeutic effect, because cytokines are not effective after lethal TBI (1) and have no bactericidal properties. It should be noted that MD2 displays antitumor activity (3) that makes this agent useful in clinical radiation therapy, as shown above.

At the cellular level, it is proposed that protein kinase C (PKC) is the target of MD-like agents (2). In contrast to known activators of PKC (phorbol esters, bryostantin) MD1, analog MD2, induces calcium-release from intracellular stores which is necessary for full cellular response (2). Furthermore, it was recently revealed that MD1 reduced MN formation in human lymphocytes with DRF=1.4, and the accumulation of cells in the G2-phase, when MD1 was added 1 h after irradiation (4). In the case of MD2 these effects may underlie the preservation or recovery of hematopoietic cells. This remains to be seen.

CONCLUSIONS

1) When administered before sublethal TBI, MD2 protects mice without toxic effects.
2) The advantage of MD2 is the possiblity of its use after lethal TBI. The effectiveness of post irradiation therapy depends on the duration of MD2 administration.
3) MD2 may be administered per os and applied to the skin.
4) MD2 protects and causes a recovery in hemopoiesis at hematopoietic syndrome.
5) MD2 may be used in clinical radiation therapy to ameliorate the patient's condition.

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Vienna, Austria

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PAPER TITLE RADIOPROTECTIVE EFFECTS OF LECTINS

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The goal of the study was to determine the radiosensitivity modification effect of lectins. Experiments were performed on mice following mathematically planned scheme.

Animals were observed after intraperitoneal injection of Pisum sativum lectin (PSA), Ricinus Communis agglutinins (RCA), Glycine soja lectin (SBA), Arachis hypogaea (AH) in a single dose 0.1 - 5.0 mg/kg b.w. to check differences in lethality and survive time.

The lethal doses (LD16, LD50, LD84) in mice found respectivly for PSA at the rate of 1,5; 1,8 and 2 mg/kg b.w.; for RCA - 4,3; 4,6; 4,8 mg/kg b.w.. Intraperitoneal administration of SBA and AH in dose 0.1 - 2 mg/kg b.w. does not produce a significant increase of lethality.

Further, four groups of animals were exposed to LD 80-100/30 (6.5-7.5 Gy, 1.5 Gy/min) gamma-irradiation at 24 h time-point after administration of the lectins in a single dose 0.1 - 1 mg/kg b.w.

The collected data showed that lethality, induced by LD80 radiation exposure, gets in PSA-pretreated group 30% shift downward. In case of LD100 radiation exposure, there were no significant differences in the rate of lethality to compared with non-treated irradiated (LD100) control.

No radiosensitivity midification effect had been found in the cases of SBA- , AH- and RCA-pretreatment no matter of a dose and a time-point of lectine administration.

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PAPER TITLE ESTIMATE OF ANTIRADIATION ACTIVITY OF RADIOPROTECTORS BELONGING TO DIFFERENT PHARMACOLOGICAL GROOPS WITH THE HELP OF INTEGRAL PROTECTION INDEX (IPI)

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

IPI is a criterion of radioprotectors efficiency. Its biological essence is to determine a share of "protected" (according to a selected index) biological objects over the given dose range while using one ore another preparation. In this work the mortality of experimental animals was used as a radiation injury index.

White nonbred mice were irradiated with 5.25 - 7.5 Gy that is with doses which result in 50 - 100 % mortality. In all cases intraperitoneal administration was used. In preliminary multifactorial experiments the doses and time of radioprotector administration were determined as optimal.

On the base of IPI values the radioprotecors investigated were ranged according to their radioprotective efficiency in the following manner: high-efficient preparations (Actovegin, Naphazolin, Cystamine, Thimogen, Limphokinin) and low-efficient preparations (Prodigiosan, Polyribonate).

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PAPER TITLE Effect of interleukin - 1β on the state of hemopoiesis in irradiated mice and dogs

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MAJOR SCIENTIFIC TOPIC NUMBER 4 or 6 (see page 7)

ABSTRACT (See instructions overleaf) The paper is devoted to the study of the possibility of using interleukin - 1β as a therapeutic drug in acute radiation sickness. Mice were irradiated at a dose of 6.0 Gy ($LD_{70/30}$) and dogs at a dose of 2.9 Gy ($LD_{70/45}$). The drug was administered to the mice intraperitoneally at a dose of 50 or 500 ug/kg 1 hour following exposure, whereas the dogs received an intravenous dose of 2 ug/kg 3,24,48 and 72 hours following exposure. The use of the drug has accelerated the recovery of hemopoiesis in mice. Even at day 7 after radiation exposure, the number of leukocytes in treated animals ($0.74 \pm 0.10 \cdot 10^9/l$) was higher than the critical level, whereas severe leukopenia ($0.30 \pm 0.09 \cdot 10^9/l$) was still noted in controls. The drug had a positive effect on the content of myelocaryocytes and granulocytes. The process of bone marrow cells maturation was more intense. A group of treated dogs differed from the controls by the slow development of leukopenia and the high content of granulocytes in the blood at 1-3 days after irradiation. This effect may be associated with the enhanced recruiting of mature cells from the medullary reserve. During the height of the disease, the intravascular aggregation of thrombocytes in treated dogs and the outer indications of hemorrhagic diathesis were significantly less pronounced as compared to the control. It is conceivable to use interleukin as a therapeutic agent in combination with other classes of drugs.

FUNCTIONAL APPROACH TO THE STUDY OF ANIMALS POPULATIONS (RODENTS - ADAPTATIONS TO HARMFUL FACTORS).

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Radiosensitivity problem is the part of problem of biological variability and physiological individuality, adaptation and total adaptability. But ionizing radiation influence prognosis is not beyond the theoretical studies yet [1, 2] and hasn't been developed even for external irradiation (acute and chronic) and known dose loads. Information concerning the responses of variety of organisms to radiation exposure is required to make an assessment of potential environmental impact. There are many data about a various small mammals' radioresistance, but all of them were fulfilled without the account of the functional status of animals.

The possibility of two alternative pathways of growth and development regularly realized during ontogenesis (single-phase and biphase growth) was demonstrated in a natural bank vole (Clethrionomys glareolus Screb.) population (South Urals, South taiga subzone). The functional approach, offered by Olenev [3, 4] and used in the zoological investigations, is based upon the functional status (the functional condition, associated with the specificity of growth, development, reproductive condition and succession in time). It has been suggested that three Physiological Functional Groups (PFG) should be distinguished.

FIRST PATHWAY OF ONTOGENETIC DEVELOPMENT. (corresponding to PFG 3).Youngs of year breeding during their Year of Birth are characterized by monophasic rapid growth, reaching an average body weight of 25 g, and enter breeding. Initial stage of formation of true roots started at 65-75 days of age [4]. The high level of metabolic processes . Rapid aging (by our data on age changes in teeth), life span is 3-5 months. These animals serve to provide the increase in numbers during their year of birth.

SECOND PATHWAY OF ONTOGENETIC DEVELOPMENT.(corresponding to PFG 2). Youngs of the year, which do not breed during the Year of Birth are characterized by biphase growth. At approximately one month of age, at a body weight of 16-18 g, growth is suspended, and the first phase of growth is concluded. The initial stage of formation of true roots in them is noted at age of 120-130 days. Activity of metabolic processes is reduced. Lifespan the Youngs of the year (with regard the second phase of growth) is 13-14 months. Aging is delayed as compared to PFG 3 by almost two times. The spring period of "conservation" is concluded by short-term growth and maturation within the course of two-three weeks, and body weight reaches 24-27 g during this. Almost all animals. which overwintered, mature. Activated processes of metabolism are similar in level to those for PFG 3 individuals. although according to absolute age animals are much older than representatives of PFG 3.

It is known that radioresistance to a considerable extent depends on previous to irradiation functional organism condition, the most significant indices of which are CNS and immune system condition, metabolism and energy processes level and also hormonal status of the organism. So the study aimed to reveal the peculiarities of the PFG 2 and PFG 3 radioresistance.

MATERIAL AND METHODS

The laboratory experiments were carried out it the two intrapopulational functional groups of bank voles (Cl. glareolus), captured from the natural environment in Southern Urals. Voles were subjected to acute total gamma-irradiation ^{137}Cs in the range 9,0 - 15,5 Gy (exposure rate of 110 s Gy/min). The lethality, average lifespan and effects of radiation on the haemopoietic system (bone marrow from femur and blood) were estimated after the dose 12.7 Gy, chosen so that it would cause the enough high level mortality of animals. The quantity of leucocytes and granulocytes of bone marrow were determined at various postirradiation intervals on 1, 4, 8, 16, 30 days from 7 different voles in each functional groupings. These experimental values were also obtained from 10 unirradiated control voles of the same grouping for making a comparison. A total of 307 individuals was utilized in this study: 177 animals from PFG 2, 130 from PFG 3. All hematological tests were measured by the standard micromethods. All animals were maintained in our laboratory conditions under the constant temperature and humidity during 30 days observation period.

RESULTS AND DISCUSSION

LETHALITY.

It is apparent from Table 1 that PFG 2 voles are more radioresistance than those of PFG 3 (LD 50/30 -13,2 ± 0,2 Gy and 12,7 ± 0,2 Gy, relatively, p≤0,05) [5]. It should be noted, the range of doses when the death of PFG 2 animals was observed is 12,7 - 15,5 Gy, while for PFG 3 voles is 11,0 - 14,0 Gy. The mortality of the breeding year's youngs were almost in 3 time greater than the nonbreeding year's young after the exposure by the dose 12.7 Gy.

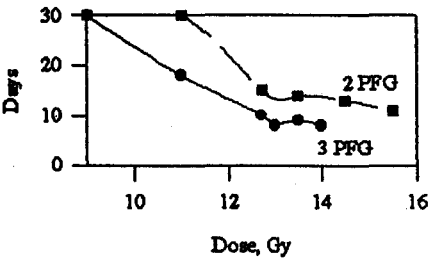
AVERAGE LIFESPAN.

The lifespan from the moment of exposures to the death during 30 days of observation is a criterion of radiation effect' assessment [6]..It is well known that this indicator is inversely proportional to quantity the dose. This seems to be supported by the data from Tabl.1 which demonstrated that after irradiation by dose 12,7 Gy the lifespan of PFG-3 voles is 9.8 days and period of death with 4-th to 12-th days whereas the lifespan of PFG 2 animals is 13,5 days (the difference is significant) and period of lethality with 11-th to 18-th postirradiation days. Dose 14,0 Gy causes 100% mortality of animals PFG 3, while the 100% level of lethality for voles PFG 2 are registered after dose 15,5 Gy (Fig.1).

Tab. 1. Lethality and average lifespan of different PFG voles, subjected to irradiation by 12.7 Gy.

Parameters	PFG 2	PFG 3
Number of animals	34	41
Number of dead animals	6	22
Lethality, (%)	17.6	53
Average lifespan	13.5±0.7	9.8±1.0

Fig. 1 The relationship between dose and average lifespan of intrapopulational bank voles groupings.



A features of the of radiation sickness among animals different functional groupings are noted. The intestinal symptoms such as diminished food and water intake, diarrhoea, loss of fluids, and another clinics manifestation, such as a decreased activity, conjunctivitis were expressed more stronger for animals PFG 3. has already been noted, part of voles PFG 3 were died in early period (within the first 7 days following exposure). It may be believed, that the main couse of their lethality is the damage to the gastro-intestinal tract. But animals PFG 2, were died in more late period. Therefore, one can suppose, that their death was occured by the bone marrow syndrome.

REACTION OF THE HAEMOPOIETIC SYSTEM:

Radioresistance of mammals depend upon the radiation damage of radiosensitive systems, such as the haemopoietic system and the gastro-intestinal mucosa. The stem cells are most radiosensitive and thus predominately influence radiation response. Symptoms occur when lack of replacement of end cells occur [7]. The kinetic of a leucocytes and bone marrow cells quantity, demonstrated in the Fig.2, shows the essential differences between the two investigated groupings by the depth of parameters decreasing. The rate and the level of recovery of the nonbreeding year's youngs were also significantly greather. Our experimental data correspond to a literature ones that cells systems, undergoing to continual renewal or rapid growth, are likely to exhibit the greatest radiosensitivity . It can assume the radioresistance of the nonbreeding part of a population are conditioned by minimizing of a metabolism processes and a reducing of the energy supplying.

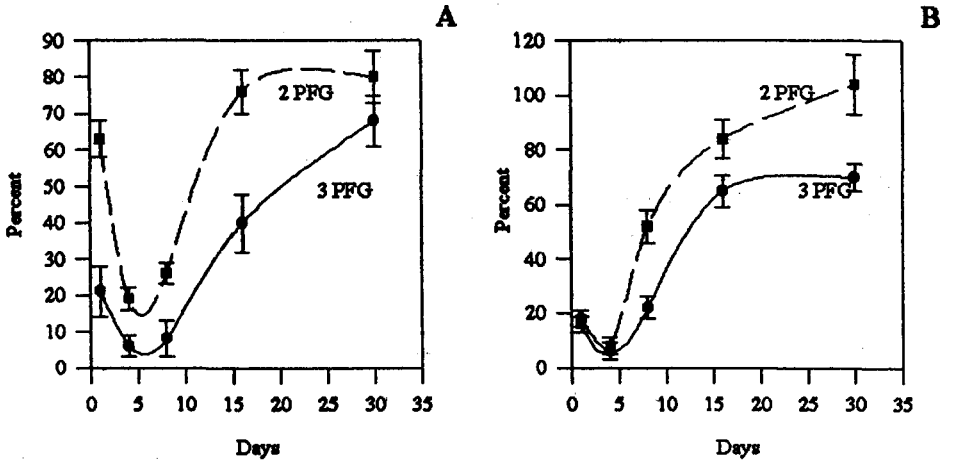


Fig. 2 Comparative haemopoietic cytokinetic in two intrapopulational functional groupings of bank voles exposed to the dose 12.7 Gy. A - LEUCOCYTES, B - BONE MARROW CELLS.

CONCLUSIONS

1. Specimens of the same absolute age, but of a different functional status , are differed essentially in the response to the radiation' action by the next criteria: LD50/30, the average lifespan, the reaction of the haemopoietic system. In the natural environment (exemplified on a rodents populations) it can change the relation between breeding and nonbreeding parts of the population and, thus, its density.

2. It seems reasonable to apply the functional approach in the practice of radioecological investigations. It permits substantial reduction of errors and provides a more precise methodological basis for analyzing of the radioresistance of a natural animals populations, for a conducting of experimental investigations and also for the health protection.

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MELANIN DECREASES REMOTE CONSEQUENCES OF LONG-TERM IRRADIATION.

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INTRODUCTION

Radiocontamination of biosphere results in chronic influence of ionizing radiation in low doses on large groups of living organisms, including human populations.

Earlier we have revealed that melanin was able to reduce the percentage of different mutation types induced by acute irradiation in animals (Drosophila, mice)(1,2). Our experimental data have also shown that melanin decreases the mutation load accumulated in Drosophila populations as a result of X-ray irradiation for 115 generations (3). The investigation of melanin possibility to influence chronic irradiation effects was very important and urgent. It was interesting to study melanin action in human cells too.

METHODS

The influence of melanin isolated from animal hair on genetic effects of irradiation in mice and human lymphocytes has been studied. Mice males of 2.5 months and 22g weight were used. The starch gel or melanin suspension in it were injected into stomach every day with a special needle. Melanin was supplied in concentrations from 0.3 to 30 mg/kg. Mice were exposed to 1-3 Gy of γ-rays of Cs^{137} at the dose rate of 0.007Gy/h (chronic irradiation) and 420Gy/h (acute one). Animals were killed 2.5-3.0 months later when the exposure was stopped. This interval was necessary for repairing irradiated spermatogonia. The levels of reciprocal translocations in metaphase of spermatocytes were analysed cytologically by the method (4) which is a modified Ivens' method (5).

Human cells were cultured according to a standard method. The culturing time was 52h at 37°C. Colchicin was injected 2 hours before cells fixation with ethyl alcohol and glacial acetic acid mixture. Blood was taken from practically healthy people of 25-45 years old in special medical hospital.

Melanin was added to culture media at G_1 and G_2 stages in the following concentrations: 0.1; 0.3; 1.0; 3.0; 10.0; 30.0 mg/l. Human cells were exposed to acute irradiation (0.5Gy of γ-rays) 40 min after melanin injection. The cytological test included dicentric-ring- and fragment- analysis.

RESULTS AND DISCUSSION

Investigations of melanin influence on spontaneous mutation level in mice has demonstrated that melanin itself doesn't possess a mutagenic activity in all concentrations used, even being supplied for 30 days.

Melanin in all concentrations was shown to reduce effectively mutagenic action of acute γ-irradiation. The melanin influence on genetic effect of chronic irradiation was even more effective. The data presented in fig.1 show that the pigment in the concentration of 3 mg/kg greatly reduced the percentage of induced mutations at different doses of chronic irradiation. The same effects were shown if melanin was supplied in other concentrations - it was revealed that melanin activity doesn't depend on concentration used.

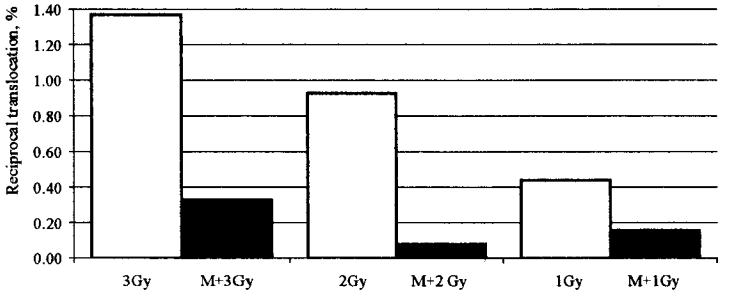

Figure 1. The melanin influence on chronic irradiation induced mutation frequency in mice germ cells (M - melanin).

It's very difficult to compare antimutagenic activity of melanin under acute and chronic irradiation because in the first case one injection of melanin has been used, but in the second case melanin has been injected many times (once a day for 10-20 days). Nevertheless it is possible to draw a conclusion that melanin is no less and even more effective under chronic irradiation than under acute one.

Radioprotective action of this pigment is connected with its high ability to accept and to give back electrons and with anti-radical activity (6,7). It's clear that when low-dose irradiation is used, the possibility for melanin to catch free radicals or electrons is better.

The investigations of melanin action in human lymphocytes have shown, that the aberration frequency in intact cells (control) was ranging from 0.42±0.19 to 1.0±0.3%. These values agree with the literature data. It means that melanin doesn't increase the control level of aberration and has no mutagenic ability.

The study on radioprotective action of melanin has demonstrated, that it is effective in reducing of aberration frequencies, induced by radiation. It was shown, that melanin in all used concentrations (from 0.3 to 30 mg/l) was able to decrease the mutation level (fig.2), but not so effectively as in mice. Strict concentration effect correlation was not observed either in human cells or in mice. There are some proofs that only small amount of melanin can penetrate into cells, and melanin quantity inside cells doesn't increase with rise in outside melanin concentration - this fact can explain absence of such correlation.

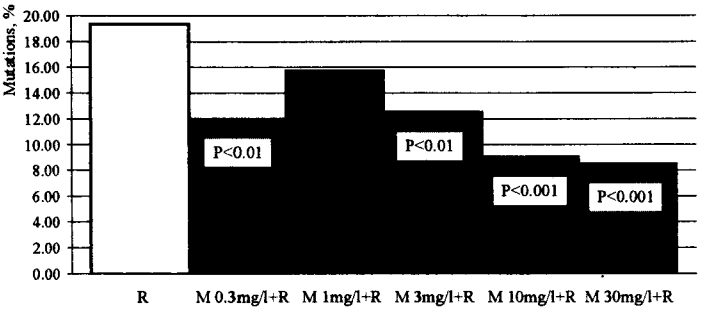

Figure 2. The influence of melanin on radiation induced mutation frequency in human lumphocytes (R - radiation, M - melanin).

Antimutagenic effect of melanin has been revealed to be the same under irradiation at G_2 stage as at G_1 one. These results demonstrate that melanin action doesn't depend (or little depends) on the repair system. The same conclusion was drawn earlier, when melanin had been investigated in drosophila and mice (1, 2).

Complete toxicological tests have been conducted. Melanin could be used in medicine for people protection against genetic consequences of long-term irradiation. We are ready to present this pigment for clinical tests.

CONCLUSIONS

The pigment melanin is not toxic and doesn't possess a mutagenic activity. It reduced the level of mutations, induced by chronic irradiation with low dose rate even more effectively than by acute one. Melanin radioprotective effect is more pronounced in vivo (drosophila, mice), than in vitro (human lymphocytes). Melanin could be used in medicine for people protection.

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EXPERIMENTAL DEVELOPMENT OF COMBINED TREATMENT FOR ACUTE RADIATION SICKNESS

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INTRODUCTION

A wide application of nuclear energy in various areas of human activity and a growing number of workers occupationally exposed to radiation increase the risk of emergency (1-3). The most severe impact results from nuclear power plant accidents, especially those with a reactor core damage, as in the case of Chernobyl accident, the largest one, when many persons received doses which caused acute radiation sickness. Complicated and varied pathogenetic mechanisms of acute radiation sickness as well as the fact that there are certain stages in its course suggest that a combined treatment is required. Our studies aimed at mitigating the cardinal symptoms of acute bone-marrow radiation syndrome were carried out for years along several lines:

- development of drugs and procedures for early treatment to be effective within the first 24 h postirradiation;
- search for drugs to intensify hemopoietic recovery when used in the latent period;
- investigation of drugs for prophylaxis or incidence reduction of infectious syndrome
- in the agranulocytosis period.

The practical results of the studies are presented in this paper.

MATERIALS AND METHODS

Animals: Male and female mongrel dogs, 1 to 5 years old, weighing 10-20 kg and maintained under the vivarium conditions on a standard diet.

Irradiation: A single uniform total-body exposure to ^{60}Co gamma-rays from the experimental generator EGO-2 at a dose rate >0.6 Gy/min and doses of 3.46 Gy (LD_{90}) and 3.65 Gy (LD_{95}). In special series of experiments to assess the preventive agents for early radiation syndrome, radiation doses were 8 Gy and 20 Gy.

Remedies investigated for modification of radiation damage manifestations: Dimetcarb and dixaphen, complex drugs comprising dimetpramide - an antiemetic, and sydnocarb or caffeine with ephedrine - central nervous system stimulators; prodigiosan, a lipopolysaccharide obtained from *Bacterium prodigiosum*; proteus vaccine; derinate, a DNA-based drug; estradiol dipropionate; hemosorption, a detoxication procedure; and an anti-infectious complex including broad-spectrum antibiotics and vitamins.

Efficiency criteria: The severity of acute radiation sickness and its symptoms, changes in the peripheral blood (erythrocytes, hemoglobin, reticulocytes, ESR, thrombocytes, leukocytes, differential blood count), 45-day dog survival.

Statistical processing: ^{2} test, Student's t-test, Wilcoxon-Mann-Whitney test.

RESULTS AND DISCUSSION

Early radiation responce shows as nausea, vomiting, hypodynamia,asthenia, undue fatiguability. To prevent these symptoms, dimetcarb and dixaphen were used. In the dog experiments, at a dose of 8 Gy which causes 91% nausea, dimetcarb orally administered 1.5 h before or immediately after irradiation led to a 40% reduction, with less pronounced hypodynamia. The other drug, dixaphen, was injected intramuscularly. It is more efficient, having the ability not only to prevent but also to stop actual vomiting in 50-60% of animals at a dose of 20 Gy.

To decrease the early postradiation intoxication symptoms and to mitigate the subsequent acute radiation sickness course, the hemosorption efficiency was studied, with optimal application conditions specified. The results clearly demonstrate the procedure to be appropriate within the first 24 h postirradiation. Hemosorption makes the damage far less severe and increases the dog survival by 50% at LD_{95} (3.65 Gy).

The detoxicating effect of hemosorption manifested itself, in particular, in removal from the exposed dogs' blood of potential radiotoxins, medium-molecular compounds of peptide nature, their amount growing substantially in the early postradiation period. Special experiments showed that isolated polypeptides suppress the hemopoietic precursor cell pool at the level of granulocyte-macrophage committed cells. Hemosorption treatment resulted in a significant increase of these cells in the bone marrow and peripheral blood at the acute stage of radiation sickness as compared with the control.

Acute bone marrow syndrome was treated for with the drugs we combined into a group called "early therapy agents". These can stimulate cytokine production, step up repairing the initial radiation injuries, intensify restoration of the hemopoietic stem cell count and microenvironment. The drugs are efficient when used in the first 24 h postradiation. The dog experiments demonstrated that the drugs which belong to high-molecular compounds (derinate) or proteolypopolysaccharides (proteus vaccine, prodigiosan) mitigate distinctly the radiation sickness course and promote the peripheral blood reconstitution following a single subcutaneous or intramuscular injection in 24 h after irradiation. At a dose of 3.46 Gy (LD_{90}), the survival rate exceeded that of the control by 31-60% ($p<0.05$). In addition to higher survival, the treated dogs exhibited a much lower incidence of severe forms of acute radiation sickness (40-50%, with 96% in the control), less pronounced hemorrhagic syndrome and a less profound drop in some peripheral blood parameters than the control did. The best results were obtained for proteus vaccine: it was also efficient at a higher radiation dose (3.65 Gy), close to the lowest absolute lethal one, with a survival rate of 50-60% (5% in the control).

It is known that the principal critical point in the bone marrow of acute radiation sickness is the hemopoietic system and its status determines the outcome. Investigated as a stimulator of postradiation hemopoietic recovery was estradiol dipropionate, a steroid hormone involved in the regulation of hemopoiesis. It was injected intramuscularly every other day starting from the midlatent period, i.e. from 5 to day 19 postradiation. With this scheme, estradiol raised the survival rate up to 47% at LD_{95} ($p<0.05$), slowed down a drop of leukocytes and promoted their clear-cut earlier restoration compared to the control, thus shortening the agranulocytosis period. At the height of disease, 20 days postradiation, the leukocyte count was 0,5 10^9 and 1,0 10^9/l blood ($p<0.05$) in the control and treated animals, respectively.

The leukopoiesis-stimulating effect of estradiol is associated with increased number of myeloid series cells in the bone marrow, due to mature and young divisible cells, and of granulocyte-macrophage precursor cells in the bone marrow and peripheral blood.

To prevent agranulocytosis-related infectious complications, we developed rational complexes by combining antibiotics of different classes to be used in cycles. The application of these complexes of systemic-action antibiotics or poorly absorbed aminoglycosides with delayed administration (starting from day 6-7, twice a day) for microbial decontamination of the intestine alongside with vitamins increases the dog survival up to 60-100% at LD_{50}-LD_{90}. At the absolute lethal dose the effect drops down to 30-40% (Fig.). The antibiotic therapy significantly reduced the severity of clinical symptoms, successfully prevented infections and mitigated the course of complications developed. However, the treatment produced little or no beneficial effect on the damaged hemopoiesis.

With one of the drugs under investigation (derinate, proteus vaccine, estradiol) or hemosorption included into the antibacterial complex, the efficiency grew from 30-40% to 70-90% at a near-absolute lethal dose and the length of antibiotic therapy was shortened (Fig.).

This is of great importance since the therapeutic agents applied together with antibiotics reduce an adverse effect of the latter on hemopoiesis and immunity by shortening their use. Our data indicate that a combined application of the remedies studied and an antibiotic complex not only mitigated the damage but also enhanced hemopoiesis. The difference was most prominent for estradiol used in combination with antibiotics.

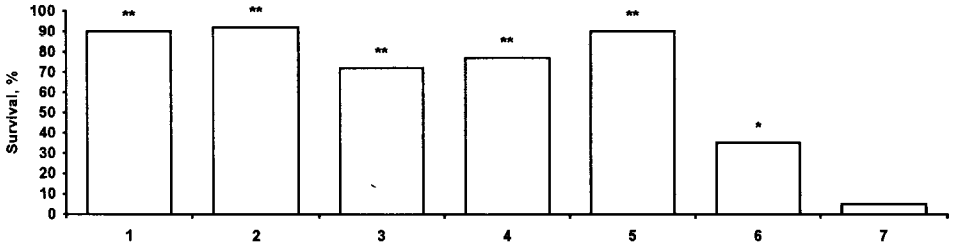

Figure. Efficiency of different remedies used in combination with antibiotics in dogs irradiated at a dose of 3.65 Gy (LD_{95}): 1 - hemosorption; 2 - proteus vaccine; 3 -prodigiosan; 4 - derinate; 5 - estradiol; 6 - antibiotics; 7 - control. Each group includes at least 10-12 dogs.
* significant difference from the control ($p < 0.05$);
** significant difference from antibiotics ($p < 0.05$).

Thus, the studies have demonstrated the efficiency of various remedies which should be included into the antibacterial complex to make it more effective in the treatment of acute radiation sickness. The results obtained conform with the evidence (4) that multiple administration of a cytokine (CSF) to exposed dogs adds to the antibiotic efficiency. The early therapy agents capable of stimulating the endogenic cytokine production and of enhancing the natural body resistance have the advantage of a single intake which facilitates essentially their practical application.

If the early therapy agents or hemosorption cannot be used within the first 24 h postradiation, we would recommend, as an alternative, that a hemopoietic stimulator (estradiol) should be included into the antibiotic complex starting from the midlatent period. The symptomatic drugs (dimetcarb or dixaphen) are also shown to be appropriate for prophylaxis and cure of early radiation response manifestation.

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PAPER TITLE INVESTIGATION OF AN ANTIRADIATION ACTIVITY OF ACIDI GLUTAMINICI AND ITS DERIVATES

AUTHORS NAMES V.V.ZNAMENSKY, R.A.SHCHEGOLEVA, N.I.LISINA.

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MAJOR SCIENTIFIC TOPIC NUMBER

It is known, that acidi glutaminici and its derivates have wide biological activity, directly take part in the regulation of energy, protein and lipid changes in human being in some pathological and stress conditions.

On the othes hand pharmaceutical industry produces somi drugs with acidi glutaminici for treating sone desorders. Beside this Natrii glutamat is used in food industry.

In this report there are data about toxical and antiradiation beateres of acidi glutaminici and its derivates: glutamini, glutamat Kalii and Calcii, Magnii, Lizine in experiments on small laboratory animals.

Natrii glutamat was used as food adition in total gamma-radiation.

At first an antiradiation activiti of the drug was investigated in different combinations: during 15 days before, after, before and after the radiation. At the second turn of experiments such groops of animals were influenced with acute radiation in subletal doses 600 and 750 rad.

The conditions of prolonged radiation with low pawer were investigated too.

The results allow to consider that Natrii glutamat in experiments on small animals in conditions with acute or prolonged radiation increase their survival with different doses on 35-40%.

To make final conclusions we must experiens Natrii glutamati on big laboratory animals - dogs.

ANTIRADIATION EFFECTIVENESS OF THE CHLORINE C

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INTRODUCTION

At present ever more attention of the experimenters in the field of search of high-effective antiray means - is directed to development of preparations from bio-active substances of a natural origin. In this connection all greater interest is caused by researches of antiray activity of these compounds, distinguished, as a rule, from known preparations of synthetic manufacture of low toxicity, absence of expressed collateral effects and possibility of course application.

It has biological (antiray) activity in dozes 5-10 mg/kg and clorine C which is derivative of chlorophil A. At present it passes tests in oncology.

Porphyrines (synthetic and natural) are recently subjected to wide study as potential medicinal means (1-2), due to their ability to be accumulated in bodies of the reticulo-endothelial system and proliferous tissues (3), as well as their physical-chemical characteristics (fluorescence, photosensitizing action, colouring).

All this testifies for the benefit of perspective use of porphyrin for treatment and diagnostics of tumors (4). According to the abovedescribed properties of porphyrines there is that fact, that for some of them radioprotective properties are revealed during the injections as well as before and after radiation treatment (5).

The abovesaid has formed the basis for study of antiray properties of the chlorine C during the experiments on small-sized laboratory animals.

The researches of sharp toxicity of chlorine C were conducted on white non-bred and linear (CBA x C57 Bl) F1 mice of both sex weighing 20-26 g. Chlorine C has been injected once (intraperitoneally, intramuscularly, subcutaneously and per os) in a wide range of dozes, registering death of the animals during 7 days after the injections. Parameters of sharp toxicity (LD_{16}, LD_{50}, LD_{84}) were defined by the method of Lichfield and Wilcocson.

Facts on sharp chlorine C toxicity are indicated in the Table.

Table Sharp chlorine C toxicity in experiments on mice.

Mice	Way of injection	LD_{16} (mg/kg)	LD_{50} (mg/kg)	LD_{84} (mg/kg)
non-bred	intraperitoneally	105	200	340
(CBA x C57 Bl) F1	intraperitoneally		270	
	intramuscularly		1100	
	subcutaneously		1400	
	per os		2000	

The facts, submitted in the Table show that toxicity of chlorine C is considerably below at the injection per os, subcutaneously and intramuscularly, than intraperitoneally. These facts permit to relate chlorine C to substances of the average toxicity.

Antiradiation effectivity of chlorine C was studied on the mice (CBA x C57 Bl) F1. Chlorine C was applied in a wide range of dozes with its' use in 3 variants: before radiation treatment, after radiation treatment, combined (before and after radiation treatment).

Studing radioprotective effectiveness of clorine C the mice have been subjected to general radiation treatment on Y-installation " IGUR " with capacity 1.68 Gy/min. 30-days-long survival of mice, dynamics of their death, changes in their weight as well as average duration of life of dead ones depending on a doze of clorine C, time and way of the injection before radiation treatment in the doze of $LD_{90\text{-}95/30}$, that has made for sharp radiation treatement 9 Gy, has been studied.

Experiments have shown, that chlorine C possesses high radioprotective àctivity in experiments on mice during intraperitoneally injection 15-60 minutes before radiation treatment in dozes, close to minimum absolutely lethal. At the injection 90 mg/ kg 15-60 minutes before the radiation treatment antiray effectiveness was 63 % and 45 %, accordingly (control - 4 %).

In optimum radioeffective doze 40 mg/kg chlorine C by its' effectiveness (80%, the control - 13 %) comes nearer to the widely known radioprotector - mercamine.

Radioprotective activity of chlorine C reduces at an increase of a time of the injection before radiation treatment and at other ways of injection (intramuscularly, subcutaneously, per os).

Studies of medical activity of chlorine C in experiments on mice have shown, that the compound does not possess medical activity. The death of the animals during the experiments did not differ in terms from their death in the control of radiation treatment.

Combined application of chlorine C (before and after radiation treatment) in the experiments on mice did not provide antiray activity. Chlorine C was injected intraperitoneally 30 minutes before radiation treatment in the doze 50 mg/kg and in 1 hour after the radiation treatment in the doze 75 mg/kg.

Thus, as a result of conducted work reasonably high antiray activity of chlorine C is established. At the same time it should be noted, that the compound is effective only at intraperitoneally injection before radiation treatment. In other variants of experience (after radiation treatment, before and after radiation treatment) it is not effective. So, it can be considered to be potential radioprotector.

In the subsequent work we are to evaluate radiodefensive abilities of chlorine C during injection per os and in conditions of radiation treatment in small dozes.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE CHANGES OF 5-AMINOLEVULINIC ACID-INDUCED PORPHYRINES ACCUMULATION IN THYROID GLAND CELLS UNDER IRRADIATION OF ANIMALS

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MAJOR SCIENTIFIC TOPIC NUMBER ...2.. (see page 7)

ABSTRACT (See instructions overleaf)

Changes of 5-aminolevulinic acid-induced porphyrines accumulation in thyroid gland cells under irradiation of animals

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The control of proliferate activity of cells under irradiation of organism gives a valuable information about process of cell populations reparation in different organs, danger of pathologic changes development as well as efficiency of radioprotective agents. We studied a changes of velocity of ALA-induced endogenous porphyrins accumulation in rat thyroid gland cells in different terms after acute (up to 3 Gy) and chronic (treatment up to 1400 kBq ^{131}I) irradiation of animals. The control experiments showed a correlation between indicated parameter and proliferate activity of cells. During the first week after acute irradiation it was registered an increase (in 1,2-3,0 times) of ALA-induced velocity of uro-, copro- and protoporphyrin accumulation in gland cells. The similar, but less pronounced changes were been detected during 1,5 month under chronic irradiation of animals. This effect strengthened under increasing of irradiation dose. With using of digital fluorescent microscopy in thyroid gland of irradiated rats after addition of ALA the appearance of cells with bright porphyrin fluorescence has been registered. A relation of observed changes with radiation-induced changes proliferate activity of thyroid gland cells as well as diagnostic value of used parameter are discussed.

GENOME STATE IN SOMATIC CELLS OF ATOMIC ENTERPRISE WORKERS AND THEIR CHILDREN, AND GRANDCHILDREN

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The clinical and experimental investigation evidence that instability of somatic cell genome plays the great role in forming the late consequences of exposure. There are comprehensive data about positive correlation and succession of damages induced in genome of somatic cells, and the consequent tumour development. It is emphasized the possibility to evaluate the genetic danger of exposure by level and character of chromosome changes into somatic cells (1). The purpose of this investigation was to evaluate the genome state in somatic cells of persons exposed to chronic radiation, and their offspring (generations I and II).

MATERIALS AND METHODS

The genome state of lymphocytes in peripheral blood was evaluated in workers of MAYAK PA, first atomic enterprise in Russia, who were exposed to chronic external γ-radiation or joint radiation (external and internal from plutonium-239 incorporated) at first years of building the MAYAK. Total doses of exposure varied from 0.5 to 4.0 Gy, and in some cases they went up to 6.0 Gy for 5-6 years of work. The amount of incorporated plutonium-239 ranged from 0 to 7.4 kBk. Period of 30 years went after end of contacting with exposure. Group of MAYAK workers contains 433 individuals, and control group consists of 150 persons. It was examined 179 children of exposed parents (the gonad dose of father - 1.78±0.11 Gy, of mother - 2.07±0.16 Gy). The age of children during period of chromosome investigation was in range of 30-40 years. It was examined 150 grandchildren (at age 2-6 years) of exposed professionals.

The standard methods of two-day culture of lymphocytes of peripheral blood and the obtaining of chromosome preparations were used. Preparations were stained by Romanovsky-Giesma method. The number of chromosomes in metaphase plate, and the aberrations of the chromatid and chromosome types were analyzed in 100 metaphases for each case.

RESULTS AND DISCUSSION

The frequency of aberrations of the chromosome type varies from 0 to 9 per 100 studied cells at late periods of chronic exposure. The mean values in different dose groups exceed the spontaneous level of aberrations at 4-8 times. The portion of aberration of stable type was 45.0-55.0% of total number of chromosome aberrations. The aberrations of instable type were mainly twin fragments and dicentrics with the accompanying fragments. The frequency of aberrations is higher when there is joint radiation influence, and is determined by correlation with the marrow dose of exposure to radionuclides (Figure). The higher frequency of chromatid aberration did not find out in comparison with control and pathological cell clones.

The frequencies of chromatid and chromosome aberrations in 179 children of exposed parents did not statistically differ from corresponding characters in control (children of town residents who had not occupational contacting with radiation) (Table 1). The portion of stable aberrations was higher in offspring exposed *in utero* (mean

dose of 0.17 Gy) and equal to almost up to 50% of total number of aberrations, than those in offspring not having antenatal exposure.

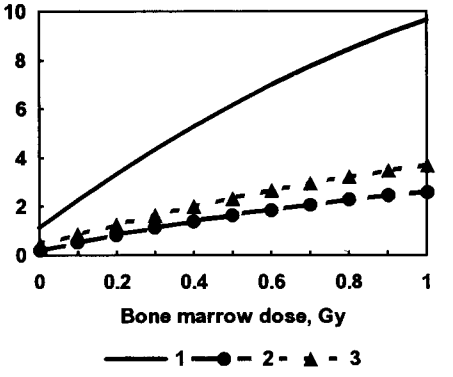

Figure 1. Chromosome aberration frequency (per 100 cells) by external γ-radiation and ^{239}Pu body burden (1 - Cromosome aberrations, 2 - Dicentrics, 3 - Stable aberrations)

Table

Chromosome aberrations (per 100 cells) in children of atomic workers

Groups	Number of patient	Cells scored	Chromatid aberrations	Chromosome aberrations		
				total	dicentrics	stable
Control	92	9200	0.4±0.08	0.37±0.07	0.11±0.04	0.16±0.04
Children of atomic workers	179	17900	0.53±0.06	0.37±0.06	0.11±0.03	0.13±0.08
Father	50	5000	0.52±0.12	0.30±0.08	0.08±0.04	0.10±0.04
Mother and father	37	3700	0.43±0.18	0.38±0.12	0.11±0.05	0.03±0.02
In utero irradiation	92	9200	0.57±0.08	0.40±0.10	0.13±0.04	0.19±0.06

The frequency of chromosome type aberration of examined 158 grandchild (generation II), which grandparents exposed to chronic radiation, corresponded to control (68 children) value (0.47±0.10 and 0.51±0.08 aberrations per 100 cell, respectively).

Thus, the instability of genome is keeping in professionals more than 30 years after chronical external and joint exposures in doses exceeded the maximum permissible dose. This cohort of workers is characterized the higher risk of malignant tumours, especially in those having plutonium-239 incorporation (2). Mutogenic effect of radionuclide was shown earlier (3, 4). In comparison with control, the heightened

instability of genome did not find out in first two generation (F1 and F2) of exposed parents by test of chromosome aberrations.

The cytogenetic effect of antenatal exposure is noteworthy. It is known that exposure *in utero* is the factor of higher risk of leukemia (4). It is necessary the future clinic-cytogenetic observation for seeking the possible prognosticating of exposure consequences by test of chromosome aberrations.

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Session 2: Thyroid effects

PERSPECTIVES OF THE USING LOW DOSES OF THYROID HORMONES FOR INCREASING ANTIOXIDANT AND REPAROGENIC POTENTIAL IN PERSONS WITH IRRADIATED THYROID GLAND (approach to the problem)

E.Antipenko, A Cheban, M.Pilinskaya, L.Owsyannikova, A.Antipenko

By the estimation of the postponed consequences of the irradiation by iodine mainly thyroid cancer and hypothyroidism usually had been evaluated. Meanwhile more actual problem there is the development of the symptoms which are typical for hypothyroidism, but not accompanied with decreasing the level of thyroid hormones (TH). In such patients the decreasing of the antioxidant potential and increasing of the level aberrant lymphocytes was marked. This phenomenon probably may be explained by the development of the functional deafness of irradiated cells to TH.

On the other hand the experimental data have shoved that TH increased the genome stability by inhibition of free-radical attack and stimulation repair from injures inflicted in the course of endogenous or induced mutagenesis. Thyroxine acts as antioxidant and reparogen within the physiological concentration range (up to 10^{-12} M), being able to stimulate the repair even after the self-dependent chromosome from injuries has been completed. Potential risks of the chronic overdosage at this situation is absent.

Thus the systematical investigation TH as antioxidant and reparogens and problems their practical treatment on irradiated patients is necessary.

MICROVASCULAR ALTERATIONS AND CYTOGENETIC CHANGES IN MEDICAL PERSONNEL OCCUPATIONALLY EXPOSED TO IONIZING RADIATION SOURCES

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INTRODUCTION

One of the most widespread applications of man-made radiation sources is their medical use. Consequently, numerous radiation-induced biological effects following both iatrogenic and occupational exposure, can be expected, including vascular and genotoxic ones. This study was aimed at detecting both microvascular impairments and changes in the peripheral blood lymphocytes' cell genome among medical personnel occupationally exposed to ionizing radiation sources. In addition, filmdosimetric data on occupational exposure will be reported.

SUBJECTS AND METHODS

The study comprised a total of 54 examinees, divided into two groups. Group 1 consisted of 24 employees of a surgical department of the University hospital Zagreb, performing operations in the X-ray zone. There were 13 men and 11 women in the group, mean age 36,7 years (age range 22-50 years), mean duration of occupational radiation exposure 12,9 years (range 1-29 years). Group 2 consisted of 30 control persons out of the same hospital, who had never worked with radiation sources. Eleven were men and 19 women, mean age 24,7 years (age range 21-38 years).

For each examinee, a detailed questionnaire was fulfilled, comprising data on personal, family and occupational background. By that means, data on counfounders known to affect both peripheral blood flow status and cytogenetic testing outcome, were collected. It was so revealed that 9 persons out of Group 1 and 13 persons out of Group 2 were smokers, none of them smoking more than 20 cigarettes a day. All the interviewed denied alcohol abuse, viral diseases, drug consumption and/or iatrogenic irradiation within one year prior to this study.

Peripheral blood flow status was established via serial application of widefield nailfold capillaroscopy, dermothermometry and digital photoplethysmography, as described already (1). Impairments detected by each method of approach were classified and interpreted according to own previously established criteria (1,2).

Changes in the peripheral blood lymphocytes' cell genome were detected by conventional structural chromosomal aberration analysis. Each examinee was taken a 5 ml venous blood sample. Immediately afterwards, a 48-hours lymphocyte cell culture was carried out according to the standard protocol (3). Coded slides were analyzed microscopically, scoring two hundred well spread first division metaphase cells per subject (4).

All the occupationally exposed examinees constantly wear KODAK personal monitoring film type 2 on their workplaces. Films are regularly read by the experts of the Laboratory for Radiation Protection and Dosimetry of our Institute. Within the framework of this study, radiation doses received in a year prior to it's start are reported. Doses lower than 0.03 mGy are always reported as 0 mGy, since they are bellow the sensitivity of KODAK type 2 personal dosemeter (5).

RESULTS

Overall results of the peripheral blood flow survey are presented in Table I. The data revealed a high incidence of microvascular impairments among surgical staff in comparison to

the controls. It was the method of capillaroscopy and dermothermometry that demonstrated the highest percentage of altered findings in both groups. Photoplethysmographically detectable microvascular changes were the fewest (none in controls). Statistical evaluation of the obtained results by means of t-test proved the increase of microvessel impairments in occupationally exposed subjects to be significant ($p<0.01$). There was also a marked difference in the type of microvascular alterations found in each group. Among the exposed subjects, mostly numerical and morphological changes occured, while in the control group functional changes of capillary blood vessels were predominant.

The results of cytogenetic testing are presented in Table II. Among the exposed examinees, not only significantly higher global frequency of chromosomal aberrations ($p<0.01$), but also a marked difference in aberration types occuring, was verified in comparison to the controls. This goes especially for dicentric and ring chromosomes, which were not found in any of the controls.

Filmdosimetric data on radiation doses received in a year prior to this study (total radiation doses;TRD) are reffered for each occupationally exposed examinee in Table III.

DISCUSSION AND CONCLUDING REMARKS

Ionizing radiation affects human health in a wide variety of ways, including direct endothelial cell damage and genotoxicity. It can be presumed that peripheral blood vessels' impairments underlie skin radiolesions (2). Capillaroscopy is the only noninvasive technique known so far, enabling direct monitoring of microcirculation ***in vivo.*** Despite high sensitivity, it's scope is limited due to the lack of pathognomonic findings (1). Therefore, peripheral blood flow survey should be completed by skin themometry. Findings of those two methods have shown very good correspondence (1). The use of photoplethysmography, however, is limited to the cases where affection of digital arteriolas is to be expected, since capillary alterations are not detectable by this method (1).

Although radiation-induced chromosomal aberrations were studied as early as in 1950's , it was the development of short-term peripheral blood lymphocyte cell culture technique (6) which enabled rapid progress of human radiation cytogenetics and biodosimetry. However, in cases of occupational exposure i.e. chronic exposure to fractionated low-dose radiation, one should be extremely cautious inhealth risk estimations. Namely, at low dose levels, dose response may not necessarily be linear; radiation effects may be immeasurable or undetectable; cellular repair processes and the presence of confounders can either decrease or increase the harmful radiation impact.

The principal aim of this study was to draw attention to some of the less investigated health hazards related to occupational radiation exposure. The results showed that even under regular workplace conditions, certain bioindicators can still reveale changes in the health status of the exposed medical personnel. Therefore, their constant comprehensive dosimetric monitoring and biomonitoring are necessary.

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Table I: Peripheral blood flow status - overall results.

No of subjects having	GRO UP 1		GRO UP 2	
	No	%	No	%
all findings normal	1	4.2	24	80.0
only cap. finding altered	10	41.7	4	13.3
cap. + thermo. finding altered	11	45.8	2	6.7
all findings (cap. + thermo. + photo.) altered	2	8.3	0	0

Legend: No = number
% = percentage
cap. = capillaroscopic
thermo. = thermometric
photo = photoplethysmographic

Table II: Cytogenetic testing outcome.

	TNo of cells analyzed	TAP	SSB	DSB	Ac	Dic	R
			TNo/(%)	TNo/(%)	TNo/(%)	TNo/(%)	TNo/(%)
Group 1	4800	2.29	31/(0.64)	32/(0.67)	41/(0.85)	4/(0.08)	2/(0.04)
Group 2	6000	0.90	26/(0.43)	16/(0.27)	12/(0.20)	0/(0)	0/(0)

Legend: TNo = total number
SSB = single-strain break
Ac = accentric
R = ring chromosome
TAP = total aberration percent
DSB = double-strain break
Dic = dicentric

Table III: Data on personal filmdosimetric monitoring.

Subject	1	2	3	4	5	6	7	8
TRD(mGy)/(NIF)	0/(0)	0/(0)	0.2/(1)	4.5/(8)	0/(0)	0.8/(3)	5.0/(7)	0/(0)

9	10	11	12	13	14	15	16
0/(0)	3.5/(5)	3.7/(6)	2.2/(5)	4.0/(6)	6.0/(9)	0/(0)	0.8/(4)

17	18	19	20	21	22	23	24
4.0/(6)	1.0/(3)	0/(0)	5.2/(8)	0/(0)	1.4/(4)	0/(0)	0.5/(1)

Legend: TRD = Total radiation dose received within 1 year prior to the survey
NIF = Number of irradiated films in the reffered period

Footnote: Doses reported as "0" are actually doses below 0.03 mGy i.e. below the sensitivity of "Kodak Type 2" personal dosemeter

A RECALCULATION OF THE AGE DEPENDENT DOSE-EFFECT-RELATIONSHIP OF THE LIFE SPAN STUDY OF HIROSHIMA AND NAGASAKI

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Abstract - The basis of the presented model is the multistage process of carcinogenesis as a biological effect. It provides simultaneously the age-dependent mortality of spontaneous and radiation induced solid tumors and dose-effect relationships at any age after exposure. The model has been used to describe the solid cancer mortality rates of the atomic bomb survivors of Hiroshima and Nagasaki. It has characteristics of both relative and absolute risk projections depending on the age of exposure.

INTRODUCTION

Major new insights into carcinogenesis have come from recent advances in cellular and molecular biology. The concept of oncogenes provides a simple explanation how agents as diverses as radiation, chemicals or retroviruses can induce tumors. Today it is known that a neoplasm usually arises from a single cell that has undergone a critical change. Despite its clonal origin, as the cells of a cancer grow and divide, progressive stages can be identified from preneoplasia to malignancy /1/.

To describe the age-dependent tumor incidence rate, we need to know the number of cells of an individual, cells which are able to divide and which are not yet damaged. M_0 is the number of these cells. In each of these cells are several DNA regions which are very sensitive to the induction of cancer. These are for example the proto-oncogenes, normal cellular genes, or tumor-suppressor genes. The mechanisms which convert proto-oncogenes into oncogenes, a process known as oncogene activation, are thought to be the critical genetic events in neoplastic transformation. An oncogene is a gene whose abnormal expression or altered gene product directly determines the production of the malignant phenotyp /2/. The generation of oncogenes from their non-transforming homologs, the proto-oncogenes, can occur in a variety of ways. Among other possibilities they can be generated by mutations within coding regions of oncogenes /3/. Much epidemiological and experimental evidence argues that malignant change is a process which results from multiple genetic alterations. Some of these steps involve oncogenes. The necessary mutations for the origin of cancer are assumed as independent from each other.

THE MODEL

The following differential equation describes the time dependent behaviour of the arising of the first step in the successive process of mutations.

$$\frac{dM_1(t,C,dD/dt)}{dt} = \left(B_0 M_0(t) - B_1 M_1(t)\right) P_u(C,dD/dt) \qquad (1)$$

M_1 (t, C, dD/dt) is the number of cells per individuum at time t which are in the first transformation step, for example: which have one point mutation on a critical, tumor relevant gene locus. The spontaneous cancer incidence rate is mainly dependent of the radicals produced by the natural cell metabolism and by chemical carcinogens. We describe these two damage probabilities with their concentration C. The spontaneous cancer incidence rate is also dependent on the dose rate dD/dt from natural background radiation. B_0 is the number of critical DNA bases (nucleotides) in critical codons of all tumor associated genes per cell. B_1 is the number of critical DNA bases (nucleotides) in critical codons of all tumor associated genes per cell after the first transformation. P_u (C, dD/dt) is the probability for mutations at usual conditions /4/. Equation (1) describes the first step necessary for a complete carcinogenesis. The number of steps necessary for the induction of cancer defines a system of coupled differential equations. Each transformation is represented by one differential equation. We assume that P_u (C, dD/dt) is constant over lifetime. The sink on the right side of equation (1) is due to the fact that cells which have reached the first step of damage are a source term for the second step. M_1 is several orders of magnitude smaller than M_0. Therefore this sink can be neglected and we can set M_0 (t) ≡ M_0 = const. Because of P_u = const. the differential equation (1) can be directly integrated:

$$M_1(t) = B_0 M_0 \; P_u \, t \qquad (2)$$

For M_i (t) we get:

$$M_i(t) = B_0 B_1 \ldots B_{i-1} M_0 P_u{}^i \frac{t^i}{1 \cdot 2 \cdot \ldots \cdot i} \qquad (3)$$

PROLIFERATION ABILITY OF BONE MARROW CELLS OF HUMAN IN LATE PERIODS AFTER CHRONIC RADIATION INFLUENCE

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The ionizing radiation influence on human causes damage of hemopoiesis, that is one of the most radio-sensitive systems. Long-term chronic exposure leads quickly to changing the peripheral blood content, and therefore cytopenia (leucopenia and thrombocytopenia) of different degrees of its expressions depending from the value of exposure doses is being observed(1). The picture of peripheral blood and bone marrow has been fully studied in period of exposure and in period after the radiation influence end, when the gradual recovery of blood indexes was occurring to level that was observed during primary examination (2). The initial thrombocyte level recovered by 5th year of observation, but the leukocyte level recovery occurred later, and approached to the initial indexes in exposed persons by only 10th year. Despite the long-term period after the exposure end (25-30 years), the persons exposed to chronic γ-radiation doses, which exceeded the maximum permissible dose more that at 10 times, had the leukopenia, and in some cases the hypoplastic state of hemopoietic.

There are different interpretations of the leucopenia genesis in the literature for persons exposed, and the number of researchers believes that leucopenia in these patients is the consequence of decreasing the granulocyte production by bone marrow (3, 4). The important significance in changing of production of these cells may be the damage of proliferation processes of granulopoietic elements in bone marrow. There are a few works in literature which indicating to the damage of proliferative ability of hemopoietic cells under chronic exposure. The decreasing of mitotic activity of bone marrow cells occurred in persons who worked during long time period having the contact with the ionizing radiation sources (3). Basing on the experimental data (5, 6), the increase of proliferative ability of bone marrow cells was observed during the processes of the chronic and fractionation exposures, however, for long time exposures (1.5-2 years) it was shown the significant decrease of the proliferation processes with lengthening of cell cycle time by 2-2.5 times (6).

It was conducted the study of morphology and proliferative ability of the bone marrow cells in 46 persons, who were exposed to occupational external γ-radiation, in 25-30 years after end of contact with the ionizing radiation sources. The total doses of external γ-radiation were 2.0 Gy for 12 persons, 2.1-4.0 Gy for 23 persons, and > 4 Gy for 11 persons. The group of the practically healthy persons (9 individuals), who had not contact with ionizing radiation sources, was taken as the control group according to the age-sex content of main group. The proliferative ability of the bone marrow elements was evaluated by mitotic index, and by the degree of incorporation of the specific labeled predecessor of DNA (tritium-thymidine) into nuclei of the bone marrow cell under conditions of the short-term cultivation of bone marrow cells. The bone marrow preparations were made in 0.5, 1, and 2 hours, then they were fixed and treated by autoradiography. Percentage of labeled elements was estimated in preparations for each type of the proliferating cells (the label index). The label intensity was determined by number of silver granules under the nuclei of labeled cells.

The number of erythrocytes and thrombocytes was on the level of norm in the blood of examined patients. For 20 persons, the leucopenia of the different degrees of

expression was found out, and the decrease of number of cells which having nucleus was indicated in punctates of bone marrow because of decreasing of all types of cells or only granulopoietic elements.

The evaluation of proliferative ability of bone marrow elements shown that there is the decreased number of mitoses of both the proliferating granulocytes and erythronormoblasts in persons with hypoplasia. The number of mitoses of proliferating bone marrow elements was on level of norm in patient with normal hemopoiesis.

The values of tritium-thymidine incorporation into nuclei of bone marrow cells differed also in persons with different content of leukocytes in blood. The increase of incorporation of labeled predecessor of DNA into young granulo- and erythro-poietic elements was found for persons with normal content of leukocytes. The significant decrease of index and label intensity of bone marrow cells was in patient with long and stable leucopenia and hypoplasia, this evidences that the proliferation rate of granulopoietic cells decreases when cell cycle and its separate phases have become more long at 1.5-2 times. Apparently, the rate reducing of proliferation of these elements influences greatly on the content of granulopoietic elements in the bone marrow and in production of mature granulocytes as well as its content in peripheral blood. In some cases with expressed hypoplasia of hemopoiesis it was found only single labeled granulopoetic elements. In comparison with granulopoietic elements, the damage of proliferative activity of erythronormoblasts do not significantly influence on the content of erythrocytes in blood, it may be because of the compensatory reactions in the erythropoiesis system: a) the possible change of differentiation direction of the stem pool cells mainly to erythropoietic-sensitive elements (3); b) the significant addition of erythrocyte production may be waited under changing the ratio of elements of early and late generations of erythronormoblasts, i.e., increasing the erythroblast content, pronormoblas and basophilic normoblasts (elements with more high proliferative potential in comparison with polychromatophilic normoblasts); and c) in some cases, it was found out activating of the erythropoiesis (together with the leuco-erythro-blast ratio), that causes the additional erythrocyte production.

Thus, the proliferative process rates in bone marrow elements are significantly lower in persons exposed to chronic γ-radiation and having the hypoplasia of bone marrow, and leucopenia.

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April 14-19,1996
Vienna, Austria

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PAPER TITLE ADAPTIVE RESPONSE FOR RADIATION-INDUCED CHROMOSOMAL ABERRATIONS ON PORTUGUESE URANIUM MINERS WORKERS

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ABSTRACT (See instructions overleaf)

Several studies show that small doses of ionising radiation induce an adaptive response to subsequent high doses. As we are doing cytogenetic studies of people working on Portuguese uranium mines for biomonitoring purposes, we use this opportunity to know if that people were more or less adapted to subsequent ionising radiation doses.

For these studies we address the following's questions:

First, can the dose received by miners, function like a prime dose for an adaptive response when they have a subsequent high exposition dose and this equal or different for low or high exposed individuals.

Second, what is the capacity for adaptive response of lymphocytes from miners when they are primed with a low dose of 5 cGy and exposed at a challenge dose of 300 cGy.

Two groups of miners workers were selected on the base of their historical monitoring doses, one as low exposed and another as high exposed and a control group.

Our observations show the ability of peripheral lymphocytes of the two groups of miners to display adaptive response after induction with a small dose of 5 cGy on G1 and a challenge dose of 300 cGy. The group low exposed present more cases with adaptive responses.

We saw also on some miners workers, that the frequency of chromosomal type aberrations after 5 cGy is less than background. If this is confirmed with more cases it means some kind of adaptation to small doses that we do not saw on the control group.

EVALUATION OF FLOW CYTOMETRIC RETICULOCYTE MATURITY INDEX (RMI) AS A DIAGNOSTIC AND PROGNOSTIC INDICATOR IN ACCIDENTAL TOTAL BODY IRRADIATION

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INTRODUCTION

In cases of accidental total body overexposures early diagnostic and prognostic methods are recquired. After high doses, bone marrow could be severely damaged. Between peripheral blood cells, reticulocytes could be considered as a useful indicator of erithropoietic activity to evaluate the extension of damage to haematopoietic system and to predict bone marrow recovery.[1]

After an accidental total body irradiation (TBI), their presence in peripheral blood, even in a very low quantity, indicates a remainder bone marrow activity. Their increase after a bone marrow aplasia, may be considered as a sign of functional recovery.

Flow cytometric analysis provide an objective measure of reticulocyte maturity, since fluorescence intensity is directly proportional to ribonucleic acid (RNA) content [2]. It is thus possible to derive a reticulocyte maturity index (RMI) taking into account the fraction of reticulocytes with high fluorescence intensity (HFR) which represents the subset at the earliest maturative stage .

With the purpose of testing the usefulness of flow citometric reticulocyte analysis with determination of a RMI as an early indicator of radioinduced marrow suppresion, this method was assayed in an animal model of TBI.

Considering that therapeutical irradiations could provide useful models of human accidental irradiations,similar studies were performed in fourteen patients undergoing TBI as a conditioning regime for bone marrow transplantantion in order to evaluate the clinical applicability of this method for monitoring bone marrow functional recovery.

MATERIALS AND METHODS

Animal studies: Wistar rats were irradiated with Co 60-gamma radiation, at doses of 2 Gy, 4 Gy and 6 Gy . Blood samples were taken from the tail vein on days 0, 1, 3, 7, 10, 17.and 30 after irradiation (a.i.). Similar sampling was carried out with non-irradiated animals.

*Clinical studies:*fourteen patients undergoing bone marrow transplantation (BMT) (6-54 years) were enrolled for this study. Pretransplant conditioning regimes consisted in chemotherapy (started on day -9) plus radiotherapy (between days -4 and -1). Bone marrow infusion was performed on day 0. TBI was administered at 3 Gy/day in twice daily fractioned doses, at a dose rate of 0.04 Gy/min. Blood samples were obtained daily during conditioning treatment and three times weekly until patient's discharge.

Flow cytometry of reticulocytes: all blood samples using EDTA as anticoagulant were treated in the same way. Whole blood (5uL) was stained for flow cytometric analysis with thiazol orange (TO) Retic Count® (Becton Dickinson) and the corresponding unstained sample was used as autofluorescent control. Both samples were analyzed on a Facstar Plus® cell sorter (Becton Dickinson).

The fluorescence of 50000 red cells was collected. The reticulocyte percentage and the mean fluorescence intensity were calculated using the Lysis I software program.

For RMI defining , data analysis was performed by two different protocols. For the first one, mean channel number of fluorescence was expressed as arbitrary units of fluorescence (UAF/RMI expression) [3]. For the second method, fluorescence intensity of reticulocytes was divided into regions of low (LFR), moderate (MFR) and high fluorescence (HFR) according to Davis [4] [5] and HFR %

was derived by dividing the number of reticulocytes in the high fluorescence area by the total number of reticulocytes.

Engraftment monitoring in BMT patients: a post-transplantation increase in absolute neutrophil count (ANC) to > 500/ml in two consecutive samples was employed to define myeloid engraftment. The second succesive increase of platelets count (PC) to > 25000/ml was considered as indication of trombopoietic recovery. Erithropoietic recovery was defined by three succesive rising counts in RMI following the nadir post-transplantation .

Statistical analysis: satistical analysis of data was performed using the Wilcoxon matched pairs signed rank sum test and one way Anova test.

RESULTS

Animal studies: Considering the whole groups, total reticulocyte percentage fell progressive to a nadir on day 3 a.i. (5.56 % from basal values , $p < 0.01$) A slow recovery was then observed There was a rise on day 17 a.i. This rise was more evident in groups irradiated with the higher doses. In most animals basal values were recovered on day 30 a.i. There was a significant reduction in AFU/RMI on day 3 a. i. (day 0 vs day 3 : 41.1 +/- 5.71 vs 30.83 +/- 1.77 $p < 0.01$) indicating a lower RNA content in reticulocyte population. Recovery was significantly earlier in the group irradiated with 2 Gy ($p < 0.05$). A rise was evident in all groups on day 17 a.i. (figure 1).

Clinical studies: in all patients HFR % decreased more rapidly than reticulocyte count and dropped to zero in most patients. After the aplasic period, the rise in HFR % preceded by several days the rise in total reticulocyte number. Engraftment was detected on day 16.5 +/- 3.2 post-transplantation (p.t.) by HFR % , on day 19.8 +/- 4.6 p.t. by ANC ($p< 0.05$) and on day 26.5 +/- 7.8 p.t. by PC ($p < 0.05$). Figure 2 shows temporal behaviour of total reticulocytes number and HFR % from the beginning of conditioning treatment until patients' discharge.

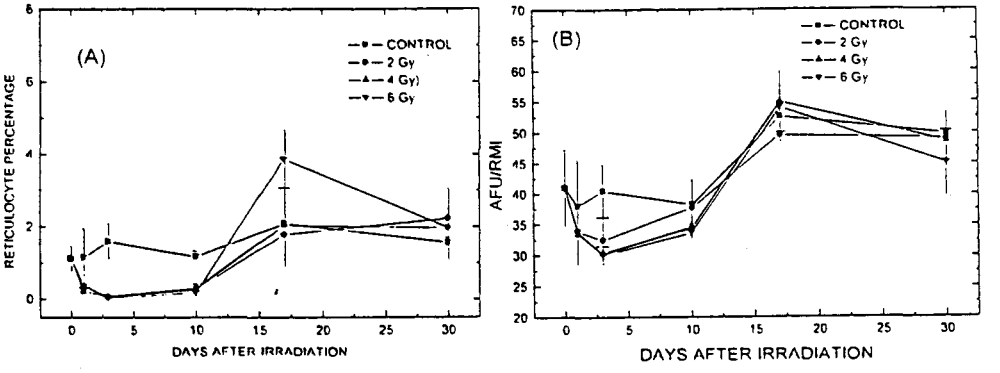

FIGURE 1 : Temporal behaviour of (A) reticulocyte percentage and (B) AFU/ RMI in total body-irradiated rats.

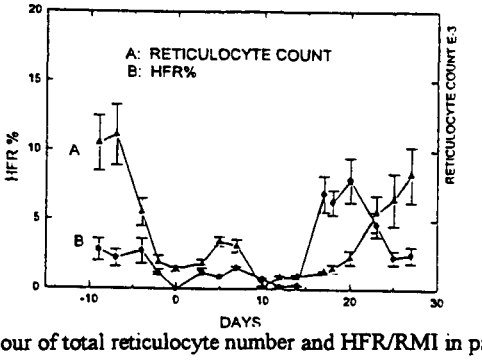

FIGURE 2 : Temporal behaviour of total reticulocyte number and HFR/RMI in patients undergoing bone marrow transplantatation Day 0: bone marrow infusion.

DISCUSSION

Reticulocyte count has been subject of numerous discussions regarding its clinical applicability and the new possibilities offered by flow cytometric methods. Besides the total number of these cells, the mean RNA content may play a role as a parameter of maturation stage , thus providing another useful tool for monitoring erythropoietic function.

Our data from animal studies indicate the sensitivity of this method for evaluating radioinduced damage to haematopoietic system, according to previous results of Tanke et al [6]. Early depletion in peripheral reticulocytes was evident in all irradiated animals but the results didn't reflect dependency to radiation doses . The percentage increased by the third week in all groups, including non-irradiated animals. A regenerative response from bone marrow to periodic blood loss could account for this late rise [7]. Higher values were reached in groups irradiated with higher doses. Thirty days a.i. basal values were recovered. Additional studies with other doses and dose-rates should be done to fully define dose-effect relationship. Considering AFU/RMI it has been possible to detect an earlier recovery in animals exposed to lower doses.

Our study with patients was primarily aimed on the early detection of bone marrow functionally recovery. Temporal relationship between erythroid, myeloid and megakaryocytic engraftment remains unanswered [3]. Although all peripheral blood cells ultimately arise from a totipotential haematopoietic stem cell, early regeneration of cell populations after BMT may be from commited progenitor cells. Hence, the rate of recovery of the three formed elements (red cells, white cells and platelets) could be independent of each other [1] Our results showed that bone marrow function was detected earlier by RMI than by either neutrophils or platelets Rise in HFR % preceded in about 3 days the rise in neutrophil counts This close temporal-relation is in good correlation with previous results of other similar studies [8]

Additional work will be required to standardize RMI and arrive at a common definition and calibration [2] Further experimental and clinical assays could provide more insight regarding the potential clinical utility of this method for monitoring accidentally overexposured patients.

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THERAPEUTIC POTENTIAL OF *EX VIVO* EXPANSION OF HAEMATOPOIETIC PRECURSORS FOR THE TREATMENT OF ACCIDENTAL IRRADIATION-INDUCED APLASIA.

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INTRODUCTION

Since the first radiation accidents which resulted into severe health effects into the workforce or the population, progress has been made in the fields of diagnosis, prognosis and treatment of accidentally overexposed victims. Since then, progress has also been made in the medical management of diseases such as aplasia. Because of the relative scarcity of radiation accidents, there is a need for complementary researches, in order to take profit of new techniques and medical approaches.

After whole body overexposure, the key issue is the therapeutic decision, i.e. the choice between bone marrow transplantation and other strategies. The indications of bone marrow transplantation cover only a short range of doses, provided the exposure is distributed uniformly within the body; a rare event in accidental settings. The results of the clinical trials for Granulocyte-Colony Stimulating Factor : G-CSF, Granulocyte/Macrophage-Colony Stimulating Factor : GM-CSF or Interleukin 3 : IL-3, *in vivo* and *in vitro* radiobiology experiments suggest that growth factor therapy could be of use after most accidental overexposures to evidence and to stimulate the remaining haematopoietic stem cells in order to shorten the duration of aplasia, although questions have been raised about growth factor infusion real clinical efficiency (1). The lessons learnt from the management of accidents evidenced that growth factor infusion has been mildly successful. This may be due to ill-defined administration protocols, to the rather narrow spectrum of action of the growth factors available for clinical use, but also to adverse side effects bound to the characteristics of these patients among them combined injuries and radiation induced inflammation (2). Therefore, great efforts should be devoted to researches in this field, exploring possibilities of alternative therapy such as haematopoietic stem cell and progenitor transfusion after their expansion.

Ex vivo expansion of haematopoietic precursor, stem cells and differentiated cells is a new approach of growth factor therapy, which may be of interest for the treatment of patients with accidental radiation-induced aplasia. These studies aim to expand the pool of progenitors and stem cells for transplantation or to expand differentiated cells (mainly granulocytes but also megakaryocytes) for transfusion (3).

This is made possible due to the development of techniques allowing the selection of a population of haematopoietic progenitors and stem cells from the blood (with stimulation by growth factors prior stem cell harvesting) or bone marrow using immature cell positive selection. The next step consisting in their culture with combination of growth factors or additional stroma cells is also under development. Autologous progenitor cells generated *ex vivo* has been recently used with some success for reconstitution of haematopoiesis after high-dose chemotherapy (4).

EXPERIMENTS

Our aim is to investigate the feasibility of expansion of the pool of progenitors and differentiated cells (mainly myeloid) for transfusion purposes.

The first step is the selection of a population of haematopoietic progenitors and stem cells from the blood (PBSC from multiple myeloma patients collected following Endoxan plus G-CSF- mobilisation) or bone marrow using CD 34+ cell positive selection (through immunoaffinity or immunomagnetic bead techniques 70-90% purity of the CD 34+ cell population are achieved).

Depending on the culture conditions, the selected human CD 34+ cell samples are then cultured at various cell densities, with recombinant growth factor combinations to promote expansion and differentiation (liquid cultures 10^3 and up to 50 10^3 cells/ml, G-CSF 10 to 1000ng/ml in conjunction with SCF, Interleukine-1, 3 and 6, 10 to 100 ng/ml) expansion of 88.3 +/- 32 fold could be obtained in one week of cultures for differentiating cells. Committed progenitors which have a longer survival *in vivo* and a high proliferation potential are also expanded 49+/-15 fold (assayed through secondary culture techniques). Expansion of 1625+/-615 fold could be obtained in 2 weeks of cultures (Fig 1,2 and 3). However simultaneous production of mature and progenitor cells was not observed.

High concentration of G-CSF and other growth factor are required to expand maturing cells whereas low G-CSF and high SCF are required to expand myeloid progenitors. Expansion of differentiating cells resulted in the loss of CD 34 expression amongst the cultured cells. The best growth factor combinations assayed allowed a good reproducibility of the experiments. For cultures up to 4 weeks, the harvested cells exhibit a fully-developed granulocyte phenotype.

Our results suggest that, in theory, it is possible to culture from the blood or bone marrow the cells able to proliferate and differentiate, in order to produce a sufficient quantity of cells to cover the transfusion needs of an accidentally irradiated victim through an aplasia episode (5).

The qualitative aspects of *ex vivo* expansion have to be studied in details. To be of therapeutic use the cells produced must retain normal mature cell functions. Our results suggest that the expanded cells have a close to normal chemotactic ability as measured through assay on PVP free polycarbonate filters 48 wells microchemotaxis chamber using fMLP as attractant (43+/-4 vs 68+/-5 for controls). The cells also retain a part of their capacity to generate superoxyde anions after two weeks of culture (measured through chemoluminescence tests). Cytological studies evidenced that *ex vivo* expansion resulted in a combination of cells of various degrees of differentiation in one to two weeks of cultures. For cultures up to 4 weeks, the harvested cells exhibit a fully-developed granulocyte phenotype (6).

DISCUSSION

Important research is necessary to adapt the *ex vivo* expansion of haematopoietic precursor stem cells and differentiated cells for the treatment of radiation-induced aplasia. Some of the growth factors (among which Stem Cell Factor : SCF, interleukine-1 and 6) have restricted use *in vivo* due to their toxic side effects although their effects on haematopoiesis could be useful. *Ex vivo* experiments could allow their use without adverse reaction. Furthermore, while proliferating, *ex vivo* expanded cells are not submitted to the effects of haematopoietic inhibitors (such as Tumor Necrosis Factor TNF alfa) which may be generated in the body following an irradiation. They are also independent from the clinical problems of the patient (such as burns and dehydration) which could delay haematopoietic recovery. It has been suggested after the Brazil accident (where a victim experienced a late hypoplasia) that, when internal exposure is involved, the use of growth factors *in vivo* would stimulate haematopoiesis-induced progenitors or stem cells to progress in the cell cycle, while the cells are irradiated. The combination of haematopoietic growth factors inducing mitosis and simultaneous prolonged radiation exposure might result in the depletion of the stem cell pool. *Ex vivo* expansion may allow the use of haematopoietic growth factors independently of these problems.

This approach could be of interest for the treatment of radiation induced aplasia if the cells necessary for *ex vivo* expansion could be found in the blood or in the marrow and harvested in sufficient quantity and that it is possible to culture them. Such cells could be available in the blood after various types of irradiation as suggested by results on haematopoietic progenitors in the peripheral blood after therapeutic irradiations. Data analysis from the past accidental exposure evidenced also that unhomogenous exposure is very frequent among victims of radiation injury (7), suggesting that in numerous accidents a bone marrow territory may still contain cells which could be used for expansion. Furthermore, preliminary results on *in vitro* irradiated human CD 34+ cells suggest that *ex vivo* expansion remains feasible for cells irradiated up to 2 gy without real loss of expanding ability.

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Figure 1 : Progenitor cell ex vivo expansion within 7 days

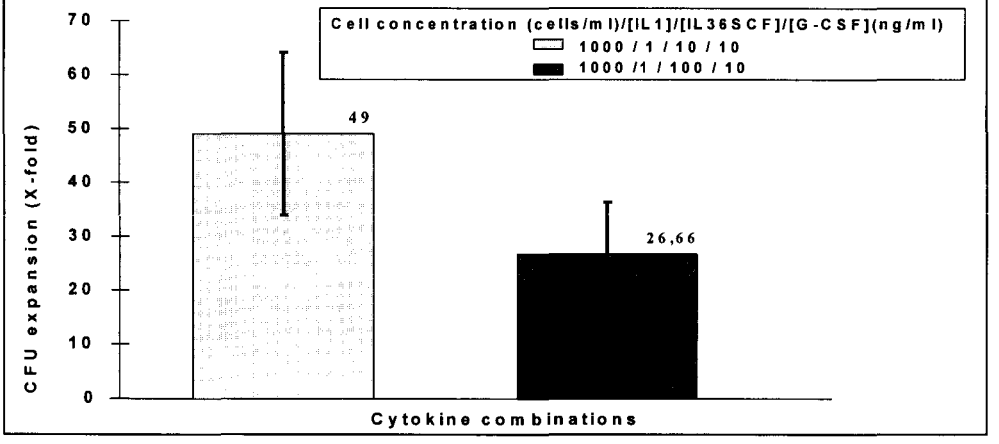


Figure 2 : Mature cell ex vivo expansion within 7 days

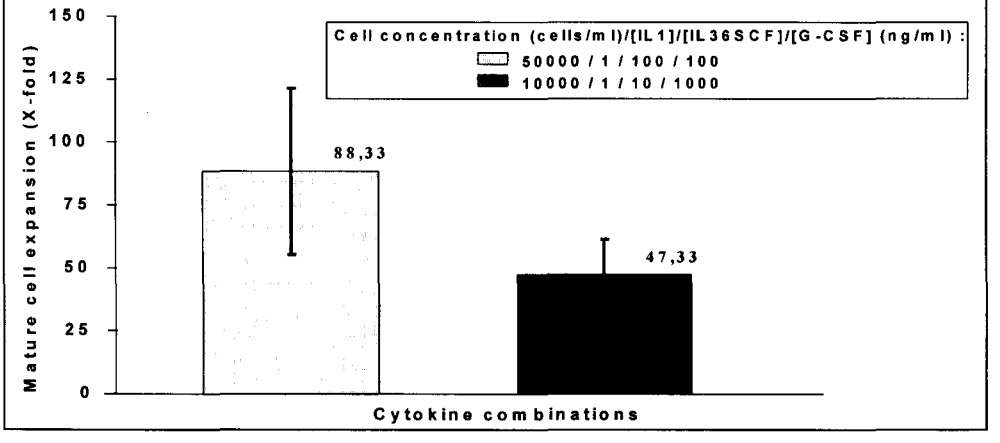
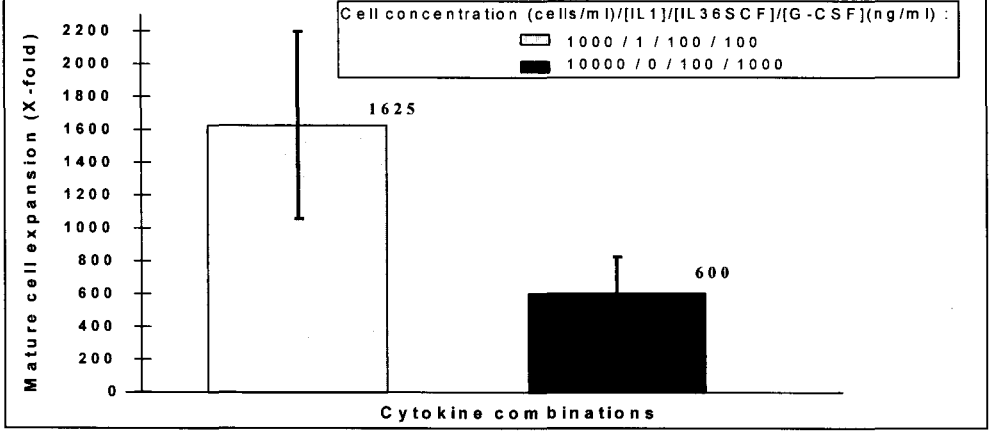


Figure 3 : Mature cell ex vivo expansion within 14 days



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PAPER TITLE CONDITIONAL PROTECTION COEFFICIENT (CPC) IS A NEW INDEX OF RADIOPROTECTORS EFFICIENCY

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

While searching and selecting radioprotectors to protect different groops of specialists under extremal conditions (such as a nuclear power plant accident) the irradiation parameters may be predicted in some cases with the help of a form of the injury distribution depending of the radiation dose.

With such information the preliminary estimation of a radioprotector efficiency is possible to fit these radiation conditions. For this purpose CPC should be used.

CPC is a probabilistic value equal to a share of the protected lives as append to a typic version of radiation injury distribution according to doses. Its value is estimated from the so-called formula of complete probability, and includes a share of biological objects irradiated with different doses as well as a protection probability while using the preparations studied over this dose range.

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PAPER TITLE INTEGRAL PROTECTION INDEX (IPI) AS A UNIVERSAL CRITERION OF COMPARATIVE ESTIMATE OF RADIOPROTECTOR EFFICIENCY

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ABSTRACT (See instructions overleaf)

For some reasons the current technique of the comparative estimation of antiradiation activity of radioprotectors belonging to different pharmacological groups not always permits to receive adequate and disfortion-free results.

That is why the authors suggest "the integral protection index" (IPI) which is essentially free from the shortcomings, inherent in current methods. Firstly, it belongs to integral criteria, that is characterizes preparation properties over all dose range of interest. Secondly, it may be used for evaluating the preparations with any tipe of action. These peculiarities are evidence of the advantage of such criterion over the rest ones.

IPI is a criterion of radioprotectors efficiency, and its biological essence is to determine a share of "protected" (as to a selected index of injury, as a rule, experimental animals mortality) biological objects with the help of one ore another preparation over the given radiation dose range.

PREVENTIVE TREATMENT OF COMBINED RADIATION INJURIES

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INTRODUCTION

The risk of sepsis development increases when thermal burns and other trauma occur in conbination with exposure to radiation. Only surgical correction of the life-threatening state recommends within 48 hours after irradiation. All other arrangements have to carry out when hemopoiesis recovery will complete (1). However exposed patients with combined injuries (CI) die during the first two or three weeks mainly due to sepsis. Therefore prophylaxis and preventive therapy of infectious complications are need early. Actual difficulties in choice of valid treatment procedure for acute radiation syndrome (ARS) exhibit additional aggravation under CI. The available facts prove decreasing early therapy efficiency for rather high dose exposure and wound trauma occurrence (2,3). The own results showed that bacterial polysaccharide pyrogenal, glycopin (synthetic analogue of muramil-dipeptide), thymus preparations (thymozin, thymotropin, thymogen), tuftsin, heterologic human and bovine immunoglobulines did not modify the low values of 30-day survival under CI (irradiation + thermal burn). Single injection of prodigiozan, zymozan and some other yeast polysaccharides in 1 hr after CI resulted at moderate increasing of survival (4,5). The main purpose of this study, which bases upon our understanding of CI pathogenesis (6), was search more effective means for preventive treatment of combined radiation injuries. Two groups of remedies were under study. The first group included so called "biological response modifiers" (BRM). These agents may increase host defences to infection, macrophage's activity and hemopoietic growth factor's secretion (7). The second group included antibiotics that should be directed against the potential gram-negative as well as gram-positive pathogens and simultaneously be useful for selective decontamination of gastrointestinal tract (8).

MATERIALS AND METHODS

Experiments performed using CBAxC57BL6 male mice and Wistar rats. Animals irradiated with a ^{60}Co gamma source at 0.45 Gy/min. The midline absorbed dose was 7 Gy for mice and 7.5 Gy for rats. Non-lethal per se full-sickness thermal burn 10% of the total body surface for mice or 15% - for rats inflicted immediately after irradiation by means of powerful lamp's light. Both animal models of CI characterised by sharp decrease of 30-day survival in compare with only irradiated mice and rats. The main increment of lethality observed during the second week after irradiation.

Human recombinant IL-1-β was kindly supplied by Dr S. Ketlinski (Institute of High-Purity Biopreparation, St.-Petersburg, Russia). IL-2 (Biotech, St.-Petersburg, Russia) was a generous gift from Dr J. Jurkevich. Extra-cellular yeast polysaccharides of Bullera alba (B-678), Sporobolomyces albo-rubescens (Sp-50) and Ronasan (sulphated mannan) prepared by Prof. N.Elinov et al. (Chemical and Pharmaceutical Institute, St.-Petersburg, Russia). Heat-killed Lactobacillus acidophilus have prepared in collaboration with Dr V.Pospelova et al. (Institute of Microbiology and Epidemiology, Moscow, Russia). Increase of survival during 30 days after CI and beneficial effects on blood system state used as indexes of BRM's efficiency.

The next broad-spectrum antibiotics were under study: ampicillin, doxiciclin, rifampicin, rifametoprim, sulacillin, ciprofloxacin and pefloxacin. We thank Prof. I.Fomina and Dr M.Vyadro, Russian National Research Institute of Antibiotics, who provided us with medicines for this experimental study. Increase of survival, absence of side effects to hematopoietic system and lymphoid organs, correction of intestine microbiocenosis, increase of host resistance to exogenous infection used as indexes of antibacterial therapy efficacy.

THERAPEUTIC PROPERTIES OF BIOLOGICAL RESPONSE MODIFIERS

Single i.p. injection of B-678, Sp-50 and Ronasan (20 mg/kg) in 1 hr after CI increased 30-day survival from 3% (untreated mice) accordingly to 23%, 20% and 34%. Intraperitoneal two or four injection of rIL-2 within the first or second days after CI (single dose was equal to 5000 U/mouse) provided statistically significant increase of survival up to 42-49% as compare with untreated mice. The best results obtained when mixture of heat-inactivated L.acidophilus used (base concentration 10^8 microbes per 1 ml growth media). Single s.c injection of this remedy in 15 min after CI (0.1 ml/mouse) provided survival increasing from 27% (untreated mice) to 80%. The own results are according to early published data (9) that single subcutaneous injection of heat-killed L.casei in 10 min after irradiation may increase survival.

The opposite data revealed when therapeutic properties of hrIL-1-β investigated in the murine model of CI. This multy-functional cytokine did not improve 30-days survival. Moreover, when given once in 4 hr after CI (100 micrograms/kg, s.c., dose recommended for ARS treatment) IL-1 caused a higher rate of mortality in early stage of CI. In particular, 28 from 40 "treated" mice died in 2-3 days instead of 100% survival untreated animals during the first 5 day after CI. Analogous results obtained when IL-1 dose reduced to 150 ng/mouse (40% of "treated" mice died at early phase of CI). Single s.c injection 150 ng IL-1 even followed 24 hr after CI accelerated lethal outcomes. It should be stressed that single administration of "high" dose IL-1 (100 micrograms/kg) to only irradiated mice did not modify the early phase of ARS in our experiments. Moreover, slightly therapeutic effect took place and survival rate in 30-day period increased up to 15 or 30%. Effects of repeated i.p. injections of smaller dose IL-1 (100 pg/mouse in 2, 4 and 6 days after irradiation and CI) studied too. Such therapeutic scheme slightly increased 30-day survival of irradiated mice. IL-1 repeatedly injected after CI did not improve survival rate during observation period. The influence of the most effective two BRMs (IL-2 and L.acidophilus) on blood system state studied in 8 days after CI. Results showed that number of bone marrow nucleated cells, white blood cells, neutrophyls end erythrocytes in peripheral blood did not differ in treated and non-treated groups. Statistically significant differences revealed only in platelet's number. However the most effective remedy (L.acidophilus) decreased blood platelet's number as compare to control group. Thus, revealed increase of 30-days survival did not accompany by corresponding correction of cytopenia under CI.

Increase survival rate of irradiated mice receiving BRMs may be connect with enhanced antimicrobial activity of relatively radioresistant macrophages, because protective effect has been seen before significant bone marrow regeneration could occur (10). Probably, BRM-induced increase of macrophages' activity has some "limits of possibility" to enhance antimicrobial host resistance under CI. Because in this situation microbial burden increases due to radioinduced translocation of microorganisms from gastroenteric tract and bacterial invasion from the contaminated burned wound. Macrophage's activation as affected by BRMs may prove to be insufficient for overcoming the infection under CI. An additional need appears in supportive therapy with antibiotics and replacement therapy with selective hemopoiesis regulatory cytokines.

THERAPEUTIC PROPERTIES OF ANTIBIOTICS

Selection of optimal antibiotics for preventive therapy made in rats and murine models of CI. According to obtained results beta-lactams, rifampicin, rifametoprim (rifampicin + trimetoprim) and ciprofloxacin did not modify the low value of survival as compare with non-treated animals. Moreover, ciprofloxacin's administration (the daily dose 20 mg/kg) aggravated bone marrow and thymus devastation, increased level of leukopenia and thrombocytopenia. New combined antibiotic sulacillin (ampicillin+β-lactamase inhibitor sulbactam) and pefloxacin has rendered strong prophylactic action during the first 8-10 days after CI but did not increase 30-day survival rate when given as single agent therapy. Results of the preventive therapy was the best when animals treated during 7-10 days simultaneously using sulacillin (the daily dose 500 mg/kg) and pefloxacin (the daily dose 50 mg/kg). Treatment course started from the first day after CI. Percentage of survival following such antimicrobial therapy increased from 7% (untreated group) to 53%. Selective decontamination effect occurred. There were not find side aggravating effects of sulacillin and pefloxacin to the radiosensitive haemopoietic system and lymphoid organ's state. Selected antibiotics increased host resistance to exogenously inoculated Ps.aeruginosa under CI.

Presented data indicate that some medicines recommended for ARS treatment provide rather insufficient effects under CI or completely fail to improve survival rate. Moreover, single injection of rIL-1-β within the first 4-24 hs after CI is very dangerous and increases death rate. Biological response modifiers such as IL-2 and heat-killed L.acidophilus as well as antibiotics sulacillin and pefloxacin may be recommend for CI preventive treatment.

This work supported in part by Commission of the European Communities according to "EC/CIS Agreement for International Collaboration on the consequences of the Chernobyl Accident" (Joint Study Programme JSP3).

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EXPERIENCE ON THE STUDIES OF MEDICAL-BIOLOGICAL EFFECTS OF RADIATION INCIDENTS IN THE URALS

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INTRODUCTION

In 1948 a facility for production of plutonium for atomic weapon (Mayak industrial complex) was put on line in the Southern Urals near the city of Chelyabinsk. In the early years the operation of the plant brought about several accidental situations. Of these incidents the most heavy ones, with respect to the impacts on the population of the nearby communities, were connected with large amounts of radioactive wastes resulting from plutonium separation. Inadequacy of the methods used for waste storage and lack of experience in radioactive waste management were the key factos which led to radioactive contamination of some territories in the Urals region and radiation exposures of the residents.

From 1949 through 1956 medium- and high-level wastes (about 3 million Ci) were discharged into the Techa-Iset-Tobol river-system. On September 29, 1957, a thermo-chemical blast occurred in one of the storage tanks containing 20 million Ci of high-level wastes. The radioactive plume with total activity of 2 million Ci passed over Chelyabinsk, Sverdlovsk and Tyumen Regions and formed the so-called East-Urals Radiation Trace (EURT). The third major radiation situation developed in the spring of 1967 due to a downwind transfer of radioactive silt deposits (total activity amounted to 600 Ci) from the dried up shoreline of the lake Karachay, an open depot of liquid radioactive wastes.

The population of the contaminated territory was exposed to a combined (internal and external) irradiation. The external exposure was due to an elevated gamma-background level on the Techa floodlands and on the territories of the villages, while internal exposure resulted from Sr-90 and Cs-137 body intakes with water and foodstuffs produced in the contaminated area (1, 2).

The critical organs in cases of population exposures were the skeleton and red bone marrow (RBM). The maximum values of the equivalent dose to RBM amounted to 4-5 Sv for the Techa riverside residents and 0.9 Sv for the EURT residents. The majority of the population in the Techa villages (74%) received doses below 0.5 Gy, 8% received doses to RBM of over 1.0 Gy and about 1% received doses to RBM in excess of 2.0 Gy (3).

ORGANIZATION OF MEDICAL FOLLOW-UP AND DOSIMETRIC STUDIES

Regular medical examinations and dosimetric measurements for the Techa riverside residents were started in the summer of 1951. They were conducted by the specialists of the Central Research Laboratory of the Mayak combine and Biophysics Institute, Health Ministry, USSR. Beginning from 1955 medical follow-up of the exposed population was carried out by the staff of the specialized dispensaries No 1 (Chelyabinsk city) and No 2 (Shadrinsk city). Radiation-hygienic and dosimetric studies were conducted beginning from 1958 by Chelyabinsk Branch of the Leningrad Research Institute for radiation hygiene and Agricultural Research Laboratory which were re-organized into Branch No 4 of Biophysics Institute, currently the Urals Research Center for Radiation Medicine (URCRM).

Medical examinations of the population after the accident of 1957 and incident of 1967 were started within the first weeks of the exposure by the staff of the Central Hospital of the Mayak combine and later were carried on by the specialists of the URCRM. Dosimetric studies of the

environment included radiometric and radiochemical analysis for contents of Sr-90, Cs-137 and Pu in water, bottom deposits, soil, as well as estimation of dose rates of gamma-radiation. Individual measurements conducted for exposed people included studies of autopsy material for Sr-90 and Pu, analysis biological discharges (urine, stools) for Sr-90, beta-metry of front teeth and frontal bone, analysis of Sr-90 content in the whole body using a whole body counter, measuring the absorbed dose in the tooth enamel using the EPR, etc.

REGISTRY AND MEDICAL-DOSIMETRIC DATA BASE

In 1967 the staff of the URCRM started to create a registry of exposed people and the medical-dosimetric data base to facilitate a medical follow-up of the exposed population. Currently, the computer-based data base contains individualized data on 90158 people exposed to radiation, their offspring, subjects exposed in utero and a comparison group.

The structure of the Data Base is composed of the following 5 registries: registry of the followed-up people, death registry, diagnosis registry, registry of family histories and dosimetry registry.

The sources of the information included in the medical-dosimetric data base are the results of medical examinations, data derived from the Civil Registrar's Office, Address Bureau, records of specialized medical institutions (oncology clinics, endocrinology dispensaries, etc).

EXPERIMENTAL STUDIES

For the purpose of predicting possible health effects for people exposed in the Urals region the URCRM staff conducted a large-scale experiments using chronic (including for life time) intakes of Sr-90 within a wide range of doses (including combinations with gamma-radiation) in the organism of laboratory animals (rats, mice). It became possible to determine the dynamics of Sr-90 accumulation in the skeleton and tissue dose formation in experimental animals and their offspring, effects of Sr-90 on the bone and hemopoietic tissue, immune system and reproductive function, dependences governing changes in life span, and optimum conditions for blastomogenic effect of Sr-90.

Among the long-term post-exposure effects of Sr-90 in animals osteosarcomogenic effect presented the highest interest. As a result of the investigations the minimum osteosarcomogenic dose was estimated (50 Gy) and the optimum Sr-90 exposure modes for sarcoma induction were defined:

- dose absorbed by bone tissue (70-90 Gy given by a single injection, 150-180 Gy for chronic irradiation)
- average dose rate (25-35 cGy/day for a single dose irradiation, 50-70 cGy/day for chronic exposure).

Extrapolation of the experimental findings to humans has shown that the daily Sr-90 concentrations which do not induce pathological conditions in cases of daily Sr-90 intakes throughout life in humans, account for 0.0185 kBq/kg, with dose rates to RBM being 0.06 cGy/day.

It has been established that under the conditions of a combined radiation exposure using osteosarcomogenic doses of Sr-90, gamma-irradiation does not exert a considerable influence on osteosarcoma induction. The effect of the combined exposure on the life span was estimated as additive at relatively high doses of Sr-90, and more additive at medium doses.

HEALTH EFFECTS OF POPULATION EXPOSURES

Medical follow-up of the population exposed in the Urals region allowed to observe the occurrence of deterministic and somatic-stochastic effects mainly in the residents of the Techa riverside area. The first medical examinations conducted 2 years after the discharges of radioactive wastes into the Techa started, gave an opportunity to detect the signs of chronic

radiation affection syndrome manifested by hemopoiesis and immunity inhibition, changes in the nervous system (asthenia, vegetative disorders, etc) and occurrence of ostealgic syndrome.
A proportion of the Techa and EURT populations developed post-exposure reactions of some of the body systems (leukopenia, granulocytopenia, immunodeficiency, asthenic conditions, vegetative disorders, microorganic affections of the nervous system, etc) at dose rates to RBM of 30-50 cSv/yr and higher (1, 2). Long-term radiation effects (cancer, leukemia) were only noted among the residents of the Techa riverside villages, those who received the highest doses (4).
Birth and fertility characteristics of the Techa residents (average equivalent dose to gonads for this population made up 16 cSv) witnessed to the normal status of their reproductive function. The incidence of spontaneous abortions and stillbirths correlated with the respective characteristics for unexposed women. The health status of the offspring of children and grandchildren of exposed people did not differ from the offspring of unexposed people with respect to certain parameters (mortality, morbidity, etc) (1).

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PROGNOSTICATING AND PHARMACOLOGICAL PROPHYLAXIS OF RADIOGENIC MALIGNANT TUMOURS

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Cancerogenic effect risks due to ionizing radiation, that impacted on large population groups because of Chernobyl and other accidents, cause the actuality of early diagnosis problems and of radiogenic tumour prevention. Since cancero-embryonic antigen and α-fetoprotein had been found, the tumour markers began to be frequently used by oncologists. However, attempt to use onco-markers, as test for earlier pre-clinic determination, have been unsuccessful. The secondary messengers of hormonal signal, cyclic nucleotides, that take the leading place in system of organism self-regulation, had attracted our attention. As known, the increase of cell division number and suppression of morphological and biochemical developments of differentiation are the fundamental characteristics of tumour growth and are proceeding together with participating of cyclic nucleotide system. The including of both nucleotides in neoplastic transformation and at the same time their constant presence in extracellular fluid (blood serum, urine) makes the perspective use of these compounds as indicators of tumour growth before the appearance of clinic signs of diseases. This coincides with the modern viewpoints on the developments of optimum programs for pre-clinic diagnosing of tumours, that needs to base on the change in homeostasis preceded the malignant tumour development (1).

The check of ability of nucleotide ratio test to give the information for prognosis of onco-diseases has been conducted in experiments on the *Wistar* rats, and allow to observe the cancerogenesis process from tumour growth induction through its outcomes during a cohort lifetime that was maximum short period - 650-700 days. Animals were exposed to γ-radiation at accumulated doses of 10 and 20 Gy, under which the malignant tumour frequency increased up to 65% (in control group up to 25%). The contents of cyclic AMP and cyclic GMP in urine were determined in each rat in 10 months after exposure (time of tumour outcome) and further in each 2 month up to natural death by using *Amersham* Kits. The postmortem diagnoses were determined by histologic methods. Results of retrospective analysis evidence that the increasing of cAMP/cGMP ratio was being observed in urine of animals, that had tumours, long before their deaths and independently from tumour localization. This period may be formally recognized as pre-clinic, and it includes different stages of tumour growth. The increase of the ratio was being due to both the reinforcement of cAMP excretion and decreasing of cGMP excretion. The data base on rats, that had the tumour, was processed statistically for obtaining the objective evaluation of possibility to use this test for pre-clinic tumour growth diagnoses. It was shown that test has the high diagnostic sensitivity (85%) and the high ability of prognosticating positive result (73%) as well as the satisfactory diagnostic effectiveness (68%). In special experiment, we have applied the prospective approach to data analysis. We assumed that the positive tumour diagnosis is recognized *a priori* during lifetime on the base of the registered increasing of cAMP/cGMP ratio. Diagnoses have been verified after animal death by histologic methods. The results of comparison between the biochemical prognosis and histologic diagnosis indicated that the coincidence was in 87% of cases. We think that the increase of cAMP/cGMP ratio, observed in urine of rats with tumour, evidence

there is the "accident regulation" start, which is used for preservation of homeostasis on new level and for control of differentiation in spite of tumour growth (2).

Basing on our conducted studies, we took out the patent for the method to find out the tumoural process in experimental animals (3). The similarity of molecular and cellular mechanisms of malignant transformation in animals and persons, and the same tissue response following the genotoxic factor influence have allowed us to begin the clinic test of this method. For this purpose, the group of high risk of malignant disease development have been selected in "Mayak" PA personnel, the first atomic complex in our country. Part of the group had been exposed to radiation at doses exceeded the maximum permissible dose at early stages of production organization, and now these individuals are at age of realization of malignant tumours.

At the same time we are conducting the studies of prevention of induced tumour diseases. The effectiveness of the medicinal preparation to prevent or to decrease the frequency of spontaneous and chemical cancerogens-induced tumours have been shown in the experimental and epidemiological studies (4,5). However, the scientific developments of the radiation cancerogenesis prophylaxis are not numerous.

We are developing the methodology to conduct the pharmacological prophylaxis of radiation cancerogenesis basing on the experimental models of chronic influence of different kinds of ionizing radiation. These models if taking into consideration the dose characteristics, were approached to conditions of occupational exposures and radionuclide influence on the contamination territories following the radiaiton accidents. The developing methodology allows for peculiarities of radiation cancerogenesis under the influence of incorporated α- and β-emitters (transuranium nuclides, tritium). These include the forming of tumour process under conditions of increased burden for antioxidant system and the corresponding increase of organism necessity of antioxidant substances. Further, the cancerogenic metabolite contents are increased because of damage of monooxygenase system P-450 of liver following the organism exposure and especially the incorporation of transuranium nuclides. In addition, the radiogenic tumour development goes frequently on the background of damage of the general and/or local immunity. The insufficiency of this system, as known, is the strong promoter of oncogenesis. Basing on the traits of radiation cancerogenesis, the medicinal preparations are selected, and their application regimen is determined (time period, pattern, and duration) for purpose of prophylaxis of tumour growth.

The approbation of the developing methodology was conducted in experiments with *Wistar* heterozygous rats and *CBA* inbred mice. The prophylactic effectiveness of medicinal preparations with different mechanisms of action (antioxidants, immunomodulators) was studied under different ways of one-time intake of plutonium-239 and under long-time (during 1/3 of lifetime) injection of tritium oxide. The preliminary findings in experiment with 391 rats indicated that the ration enrichment by β-carotin (2 mg per day) or by its complex with α-tocopherol (1 mg per day) did not modify the spontaneous cancerogenesis during all period of observation. At the same time, the using of β-carotin in 750 animals exposed to α-radiation of incorporated plutonium-239 after inhalation intake (plutonium-239 citrate, the lung absorbed doses of 240- 404 cGy) decreased the general frequency of malignant tumours in rat males at 1.5 times (48%; in control group - 72.3%) due to mainly the decreasing of lung cancer frequency. The frequency modification of osteosarcomas induced by α-exposure of bone tissue to incorporated plutonium-239 was also found only in rat males that had β-carotin and α-tocopherol in ration during all lifetime after inhalation, decreases were for standard ration - up to 30%, and for ration with vitamins - up to

5%. The β-carotin and its complex with α-tocopherol did not influence on plutonium-239-induced oncogenesis of rat females. The dependence of this preparations' modification effect from sex has been shown in other papers (6).

The distinct onco-prophylactic effect of β-carotin was also revealed on the basis of experimental model of even β-exposure of 250 *CBA* mice to incorporated tritium. The receiving of this preparation with feed (1 mg per mouse 3 times per week) during all period of radionuclide influence (6.5 months, the absorbed dose of 8.7 Gy) decreased the induced malignant tumour frequency at 1.5 times in comparison with exposure control (38.3% and 58.6%, respectively). Most demonstrative decrease was for lung cancer - 1.6% and 13.3%, respectively.

The immunomodulating preparations modified the radiogenic tumour frequencies when they were used at period of radiation damages in immune system (1350 *CBA* mice, and 800 *Wistar* rats). Myelopid (MP), produced at the Immunology Institute, Moscow, was being received by *CBA* mice exposed to β-radiation (the absorbed dose of 8.7 Gy) weekly during 3.5 months (0.5 mo. - the end of HTO intake, and 3 mo. - period after the intake end) in summary amount 700 μg per mouse ; and decreased the frequency of induced myeloleukemias at 3.6 times (up to 4.4% vs 16% in exposed control), and of lung cancer at 1.7 times (7.3% and 13.3%, respectively). However, the decreasings of the MP intake up to one-time per month and summary amount up to 300 μg per mouse during 3 months led to the decrease of MP effectiveness. The dependence of onco-prophylactic effect of MP from the pattern of intake and from received preparation amount has been also shown for α-exposure to incorporated plutonium-239 intake into blood flow. The decreasing of induced tumour frequency in organs of main deposition of radionuclide (liver, skeleton) has been only in case of using the first scheme of the MP receiving.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Acceptance	20127
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PAPER TITLE
The estimation of clinical consequences of occupational chronical exposure.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The estimation of consequences of chronical exposure was made in workers of atomic enterprise "Mayak" on the basis of studies for 40-45 years. The doses ranged from the permissible one for professionals to 10.0 Gy. The summary doses of gamma-exposure more than 2 Gy (more 1 Gy per year) caused the development of syndrome complex of the chronic radiation sickness. The early consequences were acute leukemia and bone marrow hypoplasia. At later period, the number of cases had the incomplete recovery in hematopoiesis, immune dysbalance, unstability of somatic cells' genome, and early atherosclerosis. The main cause of death was the malignant tumour, mainly pulmonary cancer in persons with plutonium-239. The multifactorial analysis enabled to reveal the significant influence of smoking on lung cancer outcome, stomach-gastritis outcome with secretory insufficiency. It was estimated the effects of internal exposure due to incorporated plutonium-239 (pneumosclerosis, liver and skeleton tumours, osteodysplasia, ostalgic syndrome, changes in lymphocytopoiesis, cytogenetic effects). It was found the reproduction function in families of person exposed, and the first generation health was evaluated. The consequences of expogure in utero were the high morbidity, immune dysbalance, unstability of genome. The study on 1577 grandchilren (at age 10-15) of exposed persons did not found deviation in physical development, biological maturation, morbidity and etc. in comparison with the control.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

ESTIMATION OF THE HEALTH EFFECTS IN PERSONS WHO PARTICIPATED IN THE LIQUIDATION OF THE CONSEQUENCES OF CHERNOBYL NPP ACCIDENT

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At present the most striking consequences in the health status are detected in persons who had suuffered from acute radiation syndrome and those who had worked for a long period in the 30-km zone of the Chernobyl NPP. Chronic inflammatory pathology, ulcerous and erosive changes of gastro-intestinal tract, circulation consequences and arterial hypertension, diseases of the nervous system, atypical anemias, lymphoadenopathias and fibrous changes of thyroid gland are characteristic.
Peculiarities of diseases are registered in the different organic pathology combinations, chronic progredient course, low effectiveness of usual treatment methods. It's supposed that genome instability induced by the complex of accident negative factors are the basis of these consequences as well as changes of cellular methabolism and autointoxication syndrome formation. Discirculatory, methabolic and immunity changes seemed to be of secondary origin.
Analysis of the existing pathology allowed to make the prognosis of the elevation of vascular and oncologic pathology frequency.

ESTIMATION OF CYTOGENETIC AFTER-EFFECTS OF THE CHERNOBYL EXPLOSION IN BELARUS

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INTRODUCTION

Investigation of mutation process in cells of human beings and animals from radiocontaminated areas of Belarus is a very important problem for understanding of Chernobyl accident consequences related with damages of genetic material in somatic and germ cells.

RESULTS AND DISCUSSION

Just after the Chernobyl accident we began cytogenetic studying of rodent, amphibian and children from radiocontaminated regions. Most of our investigations are carried out using a chromosome aberration analysis (metaphase and micronucleus methods) in different tissues: in mourine rodent - bone marrow cells, alveolar macrophages, intestinal epithelium; in frogs - bone marrow cells, intestinal epithelium; in children - peripheral blood lymphocytes. For slide preparation were used conventional methods (1-4). Berezinsky biosphere reserve, Minsk environs, Braslav District of Vitebsk Region were used as the control area. The data obtained in first years after the Chernobyl accident have shown a high level of cytogenetic damages in all objects examined: a) percentage of aberrant cells was much higher in cells of species from radiocontaminated areas than that in the control; b) aberration number in the damaged cells was higher in all objects from radiocontaminated areas. Multiple aberrations were found in cells of all the species studied; c) chromatide-type aberrations, chromosome type aberrations including markers of radiation (dicentrics and rings), were detected too.

A part of our investigations results in inhabitants of Gomel Region is presented in Table 1. Percentage of chromosome-type aberration is higher in peripheral blood lymphocytes of children than in bone marrow cells of animals because lymphocytes are out of cycle (Go) but bone marrow cells are proliferative, therefore during exposure they occur at different stages of the cell cycle. In view of this, aberrations of both chromosome and chromatide types arise naturally which corresponds to irradiation at stages G1 and G2. Increased level of cytogenetic damages in intestinal epithelium of frogs and rodent and in alveolar macrophages was found too.

For the last years a statistically significant reduction of the chromosome aberration frequency in bone marrow cells and in peripheral blood micronuclei of frogs from some areas was observed. In some areas we could not see aberration of chromosome type.

The results of 10-year examination of children from Gomel, Mogilev and Brest regions of Belarus either in dynamics or in statics have shown the necessity of evaluation a complete spectrum of genetic damages in peripheral blood lymphocytes (3-5).

Table 1. Level of chromosome aberrations in different objects from radiocontaminated regions of Belarus

Object of investigation, type of cells	Area	Density of contamination (kBq/m2)		Year	% aberrant cells in contaminated area / % aberrant cells in the control	Ratio chromosome and chromatide type of aberrations
		^{137}Cs	^{90}Sr			
Human (Children, peripheral blood lymphocytes)	v.Nudichi	680	118	1986	3.2	1.9 : 1
	v.Ilichi	815	127	1986	2.6	2.1 : 1
	v.Gluhovichi	490	67	1986	3.7	2.4 : 1
Bank vole (Cl.glareolus), bone marrow cells	v.Maisk	90	70	1986	4.5	1 : 2
				1991	4.5	1 : 10
	v.Babchin	1524	70	1986	3.3	1 : 25
				1991	4.0	1 : 7
	v.Lomachi	2331	284	1990-1993	4.4	1 : 2
Rana arvalis, bone marrow cells	v.Savichi	740	55	1986-1989	5.4	1 : 1.3
				1992	3.5	1 : 7,5
	v.Babchin	1110	77	1986-1989	4.6	1 : 5.5
				1990-1992	3.7	1 : 3
				1995	4.1	0 : 1
	v.Lomachi	2331	284	1986-1989	4.2	1 : 2.5
				1995	2.6	0 : 1

Biodosimetry carried out on the basis of our findings by using different standard curves (6-8) for evaluation of the absorbed dose has shown that the registered level of chromosome aberrations in children from the 30-km zone of Bragin District and the town of Bragin corresponds to the dose of 300-500 mSv.

A high level of mutant cells in HPRT locus ($1.4\text{-}2.9 \cdot 10^{-4}$ in comparison with the Minsk control group $3.5 \cdot 10^{-5}$ in 1991 and $3.0 \cdot 10^{-5}$ in 1993-1995) against the background of a relatively low level of chromosome aberrations as compared to 1986-1990 was revealed. However, quantitative values of chromosome aberrations exceeded appropriate frequences in the control groups and depended on the factors of the radioecological situation as well as on duration of residence in radiocontaminated region (9).

The role of cell death in forming the genetic effect of chronic irradiation of lymphocyte populations in peripheral blood of the examined children was revealed.

Complex interactions between mutation pressure and the selection effect of eliminating cells bearing chromosome aberrations and gene mutations take place in lymphocyte populations of children in radiocontaminated areas under chronic ionizing radiation. The cells with gene mutations compatible with their vital functions seem to be less subjected to the selection effect in minus-direction. Owing to this fact over the past years we observe a high level of cells with gene mutations against a background of a relatively low level of chromosome aberrations (in comparison with the level registered in the first years after the Chernobyl catastrophe).

Investigations demonstrate that chronic influence of the radiocontamination factors induces genome destabilization, increases genetic load at the level of somatic cell populations that results in reduction of their viability.

The increased radiosensitivity of a genetic apparatus of cultivated cells in children from contaminated areas was revealed (10). A chromosome aberration yield under additional X-irradiation of lymphocytes in children from radiocontaminated regions at the dose of 0.3 Gy is the same as the control children at the dose of 1.0 Gy. So far as there is relations between mutation emergence and carcinogenesis risk, the revealed children's populations with high genetic radiosensitivity should be classified as a risk group with high predisposition to tumoral diseases.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

SUM OF 10-YEARS MONITORING OF HEALTH STATE IN PATIENTS WHO SUFFERED FROM ACUTE RADIATION SYNDROME AS THE RESULT OF CHERNOBYL ACCIDENT.

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At present time 180 patients, to whom the diagnosis of acute radiation syndrome (ARS) as the result of Chernobyl accident was established in 1986, are living in Ukraine. 13 patients died in postaccidental period from different diseases.

In 40 % of patients there were determined pathological changes in haematopoiesis. It consists of transitory or steady leukopenia and leukocytosis. One patient after ARS III died from erythromyelodysplastic syndrome.

In erythrocytes were revealed changes of enzyme activity of antioxidant defence system, an activation of processes of lipid perosyde oxidation, weakening of mechanism of antiradical defence, increase of membranes destruction and activity of lizosomal enzymes (hydrolysis).

In 1/3 of patients the qualitative and quantitative changes of immunity were revealed.

In hormonal sphere there were observed an increase of basal plasma cortisol level and essential decrease of corticotropin concentration. Long hypotestosteronemia was accompanied by low hypophisis honadotropic activity and hyperprolactinemia. 5-6 years after radiation trauma tendency to increase of basal testosterone secretion was appeared. It was less in patients after ARS II and III degree of severity.

Chronic diseases of visceral organs characterise the somatic status of organism. The tendency to increase the digestive tract diseases, cardiovascular pathology, steady psychoneurological disorders, decrease mental and physical working capacity were revealed. The lost of ability to do their professional work was in 90% of patients.

The six cases of thyroid gland hypofunction were found. In others the deviation in hormones that determined thyroid gland function was not statistically reliable.

In half number of patients atherogenic changes in lipid metabolism were revealed. Their frequency and severity didn't depend from severity of irradiation.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

PRINCIPLES OF STAGE BY STAGE REHABILITATION IN PATIENTS WHO SUFFERED FROM ACUTE RADIATION SYNDROME. Bebeshko V.G., Kovalenko A.N., Khalavka I.G., Belyi D.A.

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The problem of rehabilitation of patients, who suffered from acute radiation syndrome (ARS), doesn't end at the end of acute period of this disease. After bone marrow syndrome nonspecific remote radiation pathology forms. It connects with destruction in low proliferative and nonproliferative cell systems and with specific type of their reparation that consists in compensation hypertrophy.

Inside this group of patients rehabilitation measures are based on some principles:
1) to weak of lipid peroxide oxidation; 2) to increase the activity of organism antioxidant system; 3) correction of vitamin and microelements balance; 4) to reduce the immunological insufficiency; 5) to stabilise the vegetative status; 6) to normalise the lipid metabolism (dislipidemia correction); 7) to treat and prevent chronic disease of different organs and systems; 8) to increase the physical and mental capacity; 9) to propagandise the health way of life and refuse from bad habits and professional harmfulness; 10) to have rational and balance nutrition; 11) to treat in sanatoriums; 12) to make succession in hospital, out-patient and sanatorium treatment.

For this purpose the individual combinations of antioxidants, vitamins, adaptogenes, polyvitanime-microelements complexes, immunomodulators, nootropics, adrenoblockators, sedatives, vasoactive drugs, hepatotropics and other drugs, physical training, reflex- and psychotherapy, massage and diet are used.

An introduction of elaborated rehabilitation system directed to reduce promote radiation pathology in patients after ARS has some difficulties. It is connected with deformation of mental, psychological and social status of these patients. So rehabilitation includes permanent explanatory and sanitary education.

IMMUNOBIOLOGY AND HEMATOLOGICAL ASPECTS OF HEALTH - 10 YEARS AFTER CHERNOBYL DISASTER

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INTRODUCTION

The Chernobyl Accident led to special situation when some people taking part in elimination of its onsequences of Chernobyl accident were irradiated in high doses. 145 men had suffered from acute radiation syndrome of different degree, and 105 of them lived in Ukraine. Millions ofUkrainians were also suffered from other accident consequnces as a clean-up workers, inhabitants of highly contaminated territories etc. Immune and hematologic disturbancies played the important role in early reactionsto irradiation and other factors [1 - 3]. They are veryimportant still in predicting late health effects 10 yearsafter the Chernobyl accident.

PATIENTS AND METHODS

Forty two thousand of patients who belonged variouscategories of irradiated population were studied during 10years after the accident. Chernobyl acute radiation syndromepatients, clean-up workers who were exposed to the dosesunder 1 Gy limit in 1986 and subsequent years, children andadults evacuated Prypjat, 30-km zone and contaminatedterritories population were the groups of first-lineinterest.

Immunologic studies included flow cytometric determinationof immunologic phenotype with Leu IMK , IMK Plus, Simulsetmonoclonal antibodies (MoAbs) kits (Becton Dickinson, CA, USA) in two-color assay, DNA cytometry and cell cycle analysis, activation studies (tetracycline incorporation, IL-2R, TrR expression, BRDU-, 3HTdR-uptake) in perturbed and non-perturbed cell populations, also serum Ig content, anti-RBC, anti-WBC and anti-platelet agglutinins and lysins. Radiation induced variant cells content was investigated inRBC GPA-expression assay from 1990, TCR-mutations assay from1992. Search of genetic basis of radiosensitivity was performed in HLA-A, B, C, Dr distribution (Dr.J.Minchenko).Patients evaluation was supported by IPHECA WHO project from1993. HbsAg and HbC prevalence studies in irradiated population were conducted with the help of Dept. ofEpidemiology, Hiroshima Univ.(prof.H.Yoshizawa). 'Comet'DNA assay was introduced in 1995. Hematological studiesincluded peripheral blood, bone marrow cells morphologic and cytochemical examination by means of light and electronmicroscopy, studies of microenvironment, progenitor cells and fibroblast cultivation, electron-paramagnetic resonance, infra-red spectrascopy etc. Dose dependent estimation was available with the help of dosimetric departments of Chernobyl NPP, Instituts of Experimental Radiology andEpidemiology of RCRM. Cytogenetic studies of acute radiationsyndrome patients in 1986 were performed in Medical Radiology Centre in Obninsk. Clinical investigation and laboratory data analysis were performed with the help of standartized criteria.

RESULTS AND DISCUSSION

A three stage process of the immune system recovery was seen in persons who suffered from external acute irradiation.Dynamic investigations of immune state revealed apostradiation defficiency. The cellular or/and humoralimmunity disturbances included surface phenotype changes,especially in mitogen-associated populations of CD3+, CD4+ and NK- cells (CD57+,CD11+16+) . It was worth to mention the mozaic of injury and reparation of surface membranes of immunocompetent cells. B-cells exhibited the short-time decrease of C3+ and pan-B-antigen bearing cells. Membrane changes also evoked the difficulties in locus A,B,C HLA-typing. A study of T-cell receptor stable alterations was performed among the population of contaminated territories and personnel of the Chernobyl station in 1990 -1992. A two-fold (0.4 - 0.6 per cent of the PMNC) increase of aberrant cells with decreased TCR expression was found in investigated contingents. It was accompanied by the 2 -3fold increase of GPA aberrant RBC. The duration of first stage was from 6 to 24 mnths, the absorbed dose and theduration of irradiation, the presence of somatic andpsychosomatic

pathology, age at the moment of irradiation have influenced greatly.At the second stage was the restitution of the radiation injury the increase of subpopulation of CD3+DR+ lymphocytes was seen ,accompanied by stable tendencies of finding in the peripheral blood of lymphocytes with phenotype CD4+8+ ànd CD1+, which normally were characteristic only for intrathymic stages of differentiation. The increase of B-lymphocyte count, which had pan-B-markers as CD21, C3, late differentiation markers such as surface IgG and early antigens as CD10. HLA-typing revealed the prevalence of HLA A1, 28; B 5,38; B 6, 17; B17, 18; B8,22; B 8, 27; B 27, 35; A1, B16; A1, B27; A2, B38; A10, B38; A 28, B 8 in patients who had suffered from acute radiation sickness. These data are very important for the estimation of the radiosensitivity in individuals exposed to the lesser doses and for professional selection. HLA-A1,9; B8,12; B12,13; B15,35 antigens were associated with thyroid pathology both in contaminated and clear regions of Ukraine but no differences depended on radiation. Five-six years after the irradiation a heterogenity of types of immunologic response of acute radiation sickness reconvalescents was detected. A group of patients demonstrated normalisation of CD3+, CD4+, CD11+ cell count and serum IgG and IgA content, while the others revealed immunologic defficiency of the mixed type.Subset cell cycle analysis with PJ showed the decreased response to Con A in 18 hour cultures as well as 3H-Tdr uptake in 72 hourscultures. Dose-dependent changes of enkephaline receptor on PMNC and sensibilisation to brain antigens accompanied by changes of TrR, RIL-2, CD10, CD23 activation antigens expression in healthy irradiated persons as well. Late radiation subset and functional effects could be explained by neuro-humoral regulatory changes.Late effects included compensation of radiation injury in 32% of patients. High CD3+19-, CD4+8-, CD16+56+ cells counts were accompanied by low proportion of CD8+4- cells. Non- specific activation of T-link was revealed with high contens of CD3+ and CD4+ cells, exhibiting HLADR antigens and high IL-2 and Transferrine receptors expression. Non-specific activation, especially of B-cells could be the basis of immunopatologic processes in the future. Compensatory changes were associated with the low mutation rates in T-cell receptor complex.

Dysregulatory changes were detected in 37% of patients. Individualisation of immune disorders, changed response to neuro-humoral stimuli and wavy variations of immune parameters were accompanied by the low variant CD3-4+ cells incidence - less than 5 cells/10000. Depression of cellular immunity developed as the decrease of compensation abilities especially in T-cells.

During the investigations of the mechanisms of radiation-induced pathology of haemopoietic system and role of haemopoietic cells and its microenviroment in creation of the distant irradiation consequences among reconvalescents of ARS different qualitative modifications of peripheral blood and bone marrow cells were found. Hypersegmentation, fragmentation of nucleas, the toxical granulation, fringe and basophilia of the lymphocyte's cytoplasma, the vacuolisation of cytoplasma and nuclea were found. The increase of bone marrow cellularity about 10,0 - 13,0 x 10^9 /l in 8 patients of examined group was determined. For the other 14 people the number of myelocaryocytes was normal and the hypercellularity was found in 7 patients. The number of megacaryocytes fluctuated , as a rule , on the level of physiological standart, but the increase of the number of megacaryocytes with less size of nuclear and its shape modification was found. The number of cells with active production of platelets was decreased.

The correlation between the hematopoietic lineages was normal in the most cases in spite of the number of myelocaryocytes have decreased. Four of the patients had the narrowing of erythroid lineage from 5,5 to 10,5 %. For only one patient with myelodysplastic syndrom the irritation of erythroid lineage about 42% was observed. Three of the patients had lymphocytosis in the bone marrow about 33-34%, two of them had eosinophylia about 5%.

The membrane permeability of blood erythrocytes was increased in the investigating group according to the data of osmotic and mechanical resistanse of erythrocytes. The results of studying of surface membran adsorbtion ability and its volume showed the considerable modifications of these indices. The thorning and spherical shapes of erythrocytes were found in the peripheral blood of 40 % observed patients.
Bone marrow paramagnetic centres studies showed that intensity of index of bone marrow paramanetic centres differented from one in blood in persons with steadfast deviations of peripheral blood. These results testify to significant modifications in function of important metalloproteides - transferrine and ceruloplasmine, which carry out iron transport in bone marrow, in each specific case.

The restitution of the homeostatic characteristics of hemopoetic and immune systems occured at the deviated level of regulation especially in patients with high absorbed doses; individual radiosensitivity could be revealed in the low dose limit as well as in radiation pathology. Comprehencive investigation of various exposed groups led to the opinion that the control of cell differentiation could be the key point in late effects of irradiation. Still the quantity of questions is overwhelming.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

HORMONAL HOMEOSTASIS IN LIQUIDATORS OF CHERNOBYL ACCIDENT WHO HAVE DOSES OF IRRADIATION LESS THEN 1 GY Kovalenko A.N

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In liquidators of 1986 year basal levels of some hormones were investigated by radioimmunological method 2-6 years after they left 30 km zone around Chernobyl power plant. There was revealed an increase of hypophisis function that takes part in reactions of non-specific adaptation (corticotropine, somatotropine) but not others (follitropine, lutropine, prolactine, vasopressine). Spontaneous secretion of peripheral glands was increased in common (cortisol, insulin, C-peptide, angiotensine-II, estradiol). Glucohone, renine, aldosterone secretion was not change but testosterone was reduced.

The reaction of hormonal functions on radiation trauma has prolonged character because changes were observed during some years. A part of indexes returned to optimal levels or had a tendency to normalisation.

A concentration of cyclic nucleotides in plasma was essentially high and their dynamic shows an increase of total pules during years of observation.

According to conclusion that polyfactorial changes of hormonal status connected with consequences of radiation exposure are systemic and non-specific it may be characterised as postradiation endocrinopathy.

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FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE Autoimmune responses to the brain antigens in patients with postchernobyl cerebrasthenic syndrome

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The autoantibody content of neurospecific proteins (NSP): MBP, S-100, NSE was studied in the blood serum from 70 patients with cerebrasthenic syndrome who were exposed to low dose of radiation after Chernobyl disaster in 1986 - 1987. All these patients worked on liquidating of consequences of Chernobyl catastrophe. Their syndrome is characterized by asthenia, fatigue or chronic tiredness, headaches, dizziness, poor concentration, and attention, poor memory, irritability, mood liability, expressiveness, exhaustion of physical and mental activity, high blood pressure, vegetative and vascular symptoms, feelings of hopelessness, worthlessness, lack of libido. They had a high sensitivity to loud sounds, light and temperature. The overall symtoms of these syndrome were so alike that we called it as postchernobyl cerebrasthenic syndrome.

The autoantibody levels to NSP were determined after Voller et al.,1979, the NSP being isolated from the bull brain (Berezhnoy, 1986). Increased autoantibody level to NSH was found among our patients more friequently (30- 50%) than in healthy subjects (14-20%).Despite diagnosed disease no antibodies were identified in 20% of patients. The patients revealed autoimmune responses to various brain structures-myelin-MBP, glia -S-100 and neuron - NSE proteins thereby testifying to the involvement of both glia and neurons in the brain pathology. The highest level of autoantibodies was found to MBP.

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PAPER TITLE THE METABOLIC EFFECTS OF INFLUENCE OF IONIZATING RADIATION AT PATIENTS WITH POSTRADIATION ENCEPHALOPATHY

AUTHOR(S) NAME(S) I.V.Stepanenko,A.R.Vinnitsky,N.V.Guzhovskaya,G.M.Yakhnenko,T.S.Bondar

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MAJOR SCIENTIFIC TOPIC NUMBER 2 (see page 7)

ABSTRACT (See instructions overleaf)

The revealing of clinico-biochemical and haematological infringements at patients as a result of effect of small doses ionizing radiation can promote to understanding the gears of formation the postradiation encephalopathy. Was investigated the 300 patients, damaged at liquidation of Chernobyl accident. Were studied the parameters of protein metabolism, acid- main condition, waterionic exchange, mediomolecular peptides, adaptation parameters (index Garkavi).The clinical and laboratory parameters are processed mathematically, on basis that the mathematical models of gravity condition of patients and gravity of metabolic infringements are created.The models were compared among themselves. The significant infringements of metabolic homeostasis in form the change of parity of protein fractions, activity of sulfhydryl groups, free and connected water, increase of hamatocrit, medium molecules, activity of transaminases, hypoosmoliarity are found out. Was detected the heterogenesis and undulation of biochemical deviations in dynamics of disease and intensification in time (in researches 1991 - 1994 years), with drawing into the increasing number of vital significant metabolic links, that is connected with various degree of exertion of adaptation and compensational reactions. Was established the correlation between the level of erythrocyties and haemoglobin, reduction pO2 and increasing pCO2. That all correlated with level of reduction of active blood reaction, changing of unprotein sulfhydryl groups and transaminases and testified about availability in patients the phenomena of endogenous intoxication and about stress of compenssational organism reactions. Was established statistically authentic correlation of clinical and biochemical parameters at majorities of patients.

Thus, revealed infringements on party of metabolitical homeostatis testified about formation at damaged steady pathological confition and can promote to development the postradiation encephalopathy.

FEATURES OF INFRINGEMENT OF RESPIRATORY BLOOD FUNCTION AT PERSONS, DAMAGED AFTER CHERNOBYL ACCIDENT.

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The study of gears of infringement of biochemical parameters, connected with maintenance of main function of respiratory system- delivery of to metabolic tissues and conclusions forming in constitution carbonic-acid at ills with postradiation encephalopathy is interesting and little-learning task in connection with complexity of metabolic changes at effects of ionizing radiation, availability of neurochemical conversions as a result of such effect, infringement of regulatory gears of constitution.

Actuality of given task is caused by necessity of dinamical check condition and execution duly correcting measures to persons, damaged at liquidation of consequences of Chernobyl failure.

The infringements physiological processes in organism are especially dangerous for the brain tissue, sensitive to effect of ionizing radiation. The influence of radiation, hypocsia entail the infringement of processes of synthesis of structural components of plasmatic membranes, resulting in changing in structure of membrane layers, prescription properties of membranes, processes inside and inter-cellular transport [1].

With the purpose of study of influence of infringements some biochemical parameters on maintenance respiratory the processing was conducted of function of blood at ills with postradiation encephalopathy the biochemical analyses 110 patients, damaged at liquidation of Chernobyl accident. The change of biochemical parameters at va - rious expressing of infringements of pressure of oxygen and carbon dioxide in veinus blood was evaluated. The received data on changes, potassium and sodium in plasma of blood, acid of environment- pH, sulphgidril groups, average molecules and etc. are adduced in tab. 1, 2.

Табл. 1 SOME BIOCHEMICAL PARAMETERS AT VARIOUS EXPRESSION OF INFRINGEMENTS OF OXYGEN PRESSURE IN VEINUS BLOOD

Parameters	pO2 < 35	pO2 35-45	pO2 > 45
Potassium, plasma	4.09 + 0.05	4.35 + 0.136	4.25 + 0.156
Sodium, plasma	142.6 + 0.59	160.9+18.03	143.4 + 0.94
pH	7.3 + 0.007	7.33 + 0.011	7.31 + 0.009
Middle molecules	0.35 + 0.01	0.31+ 0.023	0.31 + 0.04
AlAt	0.47 + 0.013	0.44 + 0.034	0.48 + 0.07
AcAt	0.41 + 0.016	0.36 + 0.034	0.44 + 0.06
Sulfgid. groups			
- general	321.85+13.28	344.09+29.61	448.98+22.36
-without-protein	86.82 + 3.72	84.66 + 8.2	89.54 +2.36
- protein	214.54+12.08	268.9 +34.76	359.45+21.02

Табл. 2 SOME BIOCHEMICAL PARAMETERS OF VARIOUS EXPRESSION OF CARBON DIOXIDE PRESSURE IN VENOUS BLOOD

Parameters	pCO2 < 35	pCO2 35-45	pCO2 > 45
Potassium, plasma	4.06 +0.069	4.24 + 0.09	4.14 + 0.078
Sodium, plasma	141.6 + 0.79	153.1 + 9.01	142.9 +0.65
pH	7.3 + 0.075	7.34 +0.007	7.29 +0.005
Middle molecules	0.39+ 0.015	0.31 + 0.012	0.32 + 0.016
AlAt	0.54+ 0.036	0.43 + 0.04	0.45 + 0.03
AcAt	0.44+ 0.031	0.38 +0.024	0.39 + 0.02
Sulfgid. groups			
- general	358.26+17.66	319.26 +24.9	318.54+21.94
-without-protein	94.64+ 7.59	81.29 +3.76	86.75+ 5.69
- protein	263.77+18.02	238.98+22.05	231.13+21.64

Is received, that at oxygen deficit the processes endogene intoxication, acidosis and deficits of potassium are aggravated.

The low pressure of carbonic acid in blood is possible to be regarded as the certificate of unsufficient activity of metabobo-

lic processes with accumulation in blood underoxidizing products.

The marked normalization of parameter of concentration of hydrogen ions (pH) for significances of pressure of oxygen and carbon dioxide in venous blood within the limits of norm speaks about importance of given parameter for valuation of processes of transport of carbon dioxide and in organism.

For postradiation encephalopathy the tendency to decrease the parameter of acid environment- pH was marked. The parameters, informative for infringement the correlative communications with changes of acid of environment, are adduced in tab. 3.

Tab. 3 CHANGE SOME BIOCHEMICAL PARAMETERS AT INFRINGEMENTS of PARAMETER OF ACID OF ENVIRONMENT - pH

Parameters	pH < 7.35	pH >=7.35
Protein gener.	65.002 +0.69	67.78 + 0.986
Middle molec.	0.391 + 0.02	0.32 + 0.011
AmAT	0.54 + 0.03	0.442 + 0.024
AcAT	0.45 + 0.03	0.39 + 0.016

The increase of average molecules at decrease of parameter pH speaks about importance of this parameter for judgement about expression of processes endogenous intoxycation.

Connection with the above-mentioned constructs the mathematical model of dynamics the pressure of oxygen in various tissues of organisms and the regimes of regulation in dependence on kind external and internal indignations are investigated.

The conducted mathematical simulation permits to receive the additional criteria of character of clinical course of postradiation encephalopathy in dynamics of treatment.

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PAPER TITLE
CHANGES OF ERYTHROCYTIC SYSTEM IN IRON DEFICIENCY ANEMIA (IDA) IN CHILDREN ON THE BACKGROUND OF COMBINED EFFECT OF RADIATION AND OTHER UNFAVORABLE FACTORS IN THE REPUBLIC OF BELARUS

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

CHANGES OF ERYTHROCYTIC SYSTEM IN IRON DEFICIENCY ANEMIA (IDA) IN CHILDREN ON THE BACKGROUND OF COMBINED EFFECT OF RADIATION AND OTHER UNFAVORABLE FACTORS IN THE REPUBLIC OF BELARUS.

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The morphological characteristics and physico-chemical properties of erythrocytes, erythrokinetics, activity of enzyme protection antioxidant (superoxide dismutase, catalase, glutathione reductase in 668 children from the region of the Republic of Belarus with different level of contamination with radionuclides, lead, mercury and nitrates have been examined.

It was shown that the children with the 3-rd degree of IDA develop the impairment of erythrocyte morphology, a decrease of their life duration at the expense of intensified hemolysis, change of erythrocyte membrane stability to some physico-chemical factors and decrease of the enzyme activity of antioxidant system. The direct dependence on the heavy metal and nitrate concentration in blood and urine, degree of the impairment of structural-functional indices of erythrocytes and a marked anemic syndrome in children has been determined. It has been established that in IDA genesis a combined effect of unfavorable ecological factors plays a considerable role and chronic effect of low doses of radiation.

MORPHOLOGICAL PECULIARITIES OF CHRONIC GASTRITIS IN CHILDREN AND ADOLESCENTS OF BELARUS UNDER CONDITIONS OF LOW DOSE RADIATION-CHEMICAL EXPOSURE

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Following the accident at the Chernobyl nuclear power station, the unfavourable ecological situation due to chemical contamination of the environment was aggravated by the radiation effect. At present, radionuclides of caesium and lead, as well as the nitrates, are the most widely distributed ecotoxicants.

Under conditions of the negative ecological effect as the result of per os penetration of the majority of xenobiotics, the gastro-intestinal tract is at particular risk. During 1990-1994, 442 children and adolescents aged 10-17 residing at the ecologically different regions of the Republic have been examined. The program of examination included morphological investigation of bioptates of the stomach mucous membrane, determination of the radiocaesium specific activity in the body and the evaluation of lead content in blood and nitrates content in urine.

Depending on the level of radionuclides and xenobiotics accumulated in the body, all the children were divided into two groups. The main group (n=289) was represented by the schoolchildren residing at the contaminated with radionuclides territories of the Gomel and Brest Regions. Average annual effective dose equivalent in these children exceeded 3-5 times relatively permissible level (1 mSv). In 2/3 of the examined children, due to the Cs-137 accumulation in the body, the contribution from the internal exposure to the total dose formation made 63-96%. Besides, an excess of the permissible level of lead in blood was marked in 53.3% of children from this group (average index in the group- (0.117 ± 0.008 mg/l) and an excess of nitrates in urine - in 24.4% of children (71.8 ± 3.5 mg/l).

The control group (n=153) was represented by the schoolchildren residing under similar socio-economical conditions of the Vitebsk Region at the territory with natural background radiation. Nitrates loads in these children were identical with those of the main group (average index - 72.0 ± 5.2 mg/l) with the permissible lead indices in blood (0.053±0.009 mg/l).

It has been ascertained that the occurence of chronic gastritis is high in the first and second group with no statistically significant differences between them and equals, accordingly, to 70.9 ± 2.7% and 71.2 ± 3.7%. The majority of chronic gastritis are helicobacter (HP)-assOciated (Table 1).

The occurence of HP infection with the unchanged stomach mucous membrane was considerably higher in schoolchildren residing under conditions of ionizing exposure.

Table 1

Distribution of the examined children (%) with respect to the revealed HP-infection in the stomach mucous membrane

Analysed index	Main group	Control group
Chronic gastritis associated with HP	88.8 ± 2.2	78.0 ± 4.0

Table 1 (continuation)

Analysed index	Main group	Control group
HP presence on the stomach mucous membrane in healthy children	44.0 ± 5.4 *	18.2 ± 5.8 *

Note: * - differences are statistically significant ($p < 0.001$).

According to literature reports, statistically significant relationship between the HP infection of the stomach mucous membrane and the morbidity and mortality from stomach cancer has been ascertained [1,2,3], the possibility of procancerogenous effect from the HP infection being increased when combined with the effect of toxic factors of the environment.

It is known that chronic gastritis caused by HP infection is developing, as a rule, until atrophyc changes in the stomach mucous membrane occur [3]. The conducted by us analysis revealed statistically significant differences ($p < 0.001$) in the frequency of atrophy of the stomach mucous membrane in schoolchildren from the first and second groups (Table 2).Intestinal metaplasia of the mucous membrane epithelium and a combination of metaplasia with the atrophyc changes on the background of chronic gastritis were also observed more frequently in children residing at the contaminated with radionuclides territories.

Table 2

Occurence (%) of atrophy and intestinal metaplasia in stomach mucous membrane in children and adolescents with chronic gastritis

Analysed index	Main group	Control group
Atrophy	16.1 ± 2.6 *	2.7 ± 1.6 *
Intestinal metaplasia	5.4 ± 1.6	2.7 ± 1.6
Atrophy + intest. metaplasia	2.9 ± 1.2	0.9 ± 0.9

Note: * - differences are statistically significant ($p<0.001$).

It's important to note that the frequency of the mucous membrane atrophy was growing with the increase of a number of toxicants in the body.In schoolchildren residing under conditions of high radiation levels, the occurence of atrophyc gastritis made 4.1% with the permissible concentrations of lead and nitrates in the body. In children of the same group with the elevated concentration of nitrates in urine (more than 100 mg/l), the occurence of the stomach mucous membrane atrophy was increasing and reached 9.4%. With the elevated lead concenrations in blood (more than 0.10 mg/l),it made 11.7%. With the elevated level of three toxicants in the bodsy (caesium-137, lead, nitrates), the frequency of atrophyc gastritis made 36.8%.

Thus, in a considerable number of children and adolescents residing at the contaminated with radionuclides territories of Belarus, an increased content of caesium radionuclides in the body was revealed, as well as increased concenrations of lead and nitrates which present a potential health hazard.

From our point of view, these factors contribute to the appearence, already in the childhood, of chronic stomach pathology, to its rapid growth with the development of atrophy and intestinal,metaplasia of the mucous membrane. Taking into account high occurence of the HP stomach infection not only in patients with chronic gastritis, but also in healthy children, we face the situation which is not typical for children's age and prognostically dangerous, since it suggests high pre-cancerous potential for chronic gastritis.

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New IAEA recommendations on the calibration of radiation protection monitoring instruments

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Introduction

In 1971, the IAEA provided the Technical Report Series No. 133 (1) as a guide for those establishing or operating calibration facilities for radiation monitoring instruments. The sources of radiation and associated apparatus and calibration techniques presented in the report are examples of what established calibration laboratories deemed adequate. Because of the multitude of applications for radiation monitoring instrumentation, e.g. in medicine, radiography, agriculture, it is impossible in one report to describe the complete calibration of all instruments. However, the principles described serve as a basis for calibrating radiation protection instruments.

The calibration report had to be updated for various reasons:

- Considerable progress has been made in standardizing reference radiation fields and calibration procedures by the International Organization for Standardization, ISO.
- The International Electrotechnical Comission, IEC, has produced many standards on the performance specifications and type testing of radiation protection monitoring instruments.
- New quantities and units were introduced by the International Commission on Radiation Units and Measurements, ICRU, in Report No. 39, 43, 47, 51 (2, 3, 4, 5).
- The International Commission on Radiological Protection, ICRP, has recommended a revised system of dose limitation, including specification of primary limiting quantities for radiation protection purposes (6).
- A new method has been recommended by ISO for the expression of uncertainty in measurement (7).
- A network of secondary calibration laboratories has been set up by WHO/IAEA in many countries. Although these laboratories were primarily concerned with therapy standards they are increasingly becoming more involved with calibrating radiation protection instruments.

The contents of the individual chapters of the report (1) have been considerably revised and extended. After explaining the purpose of calibration, this paper describes some general concepts applied in the revised report which is expected to be published in 1996.

Purpose of calibration

The primary objectives of a calibration are defined as follows:

- To ensure that an instrument is working properly and hence will be suitable for its intended monitoring purpose.

- To determine, under a controlled set of standard conditions, the indication of an instrument as a function of the value of the quantity intended to be measured. This shall be done over the complete range of indication of the instrument.
- To adjust the instrument calibration, if possible, so that the overall measurement accuracy of the instrument is optimized.

The revised report describes a comprehensive range of calibration equipment and techniques. However, the scope of tasks performed by any one particular facility will depend upon the types of instruments that must be calibrated, as well as upon the conditions under which the instruments are likely to be used. The facilities may range from those which perform routine calibration or checks using simple assemblies to highly sophisticated laboratories where detailed energy response characteristics can be determined. The more sophisticated of these facilities will have a range of reference instruments and reference sources which, in general, will be compared to national primary standards which are themselves probably subject to international intercomparison.

General concepts

Calibrations of dosimeters are only considered in terms of the operational quantities for area monitoring ambient dose equivalent $H^*(10)$ and directional dose equivalent $H'(0.07, \Omega)$, and for individual monitoring personal dose equivalent $H_p(10)$ and $H_p(0.07)$. The quantities $H'(3, \Omega)$ and $H_p(3)$ are used very rarely and are often conservatively estimated by $H'(0.07, \Omega)$ and $H_p(0.07)$. Therefore they are not dealt with in the revised report.

One has to be aware of the relationship between reference radiation fields, physical quantities that characterize the dosimetric properties of the reference radiation fields, and quantities used for calibrations and type tests. Reference fields recommended by ISO are established in the calibration laboratory. Physical quantities like fluence for neutron radiation and air kerma for photon radiation characterize the reference radiation field and are measured with reference instruments. Quantities related to calibrations and type tests are derived from the physical quantities by appropriate conversion coefficients. Tables of conversion coefficients for ISO reference radiations of normal incidence are given in the revised report.

Personal dosemeters should, in principle, be irradiated on standardized phantoms. The water slab phantom, water pillar phantom and PMMA rod phantom of ISO have been selected for calibrations and type tests. When these phantoms are used no corrections shall be applied to correct for any differences in backscatter relative to ICRU tissue.

Routine calibrations of personal dosimeters may be done, sometimes more simply, free in air or on a PMMA phantom, and even with a type of radiation other than that which the instrument is intended to measure. Such simplifications can be justified provided the calibration procedure is checked during the type test so that the difference in the responses of the dosimeter under both irradiation conditions is the same for each dosimeter of the same type. Calibration on a phantom should be done if the dosemeter is very sensitive to the radiation backscattered from the phantom, such as the neutron albedo dosimeter, for example.

A distinction is made between four methods of calibration

- calibration by successive irradiations of a reference instrument and the dosimeter without using any monitor detector

- calibration by successive irradiations of a reference instrument and the dosimeter with the aid of a monitor detector
- calibration by simultaneous irradiations of a reference instrument and the dosimeter
- calibration in a known radiation field (without any reference instrument)

The second method is illustrated in figure 1.

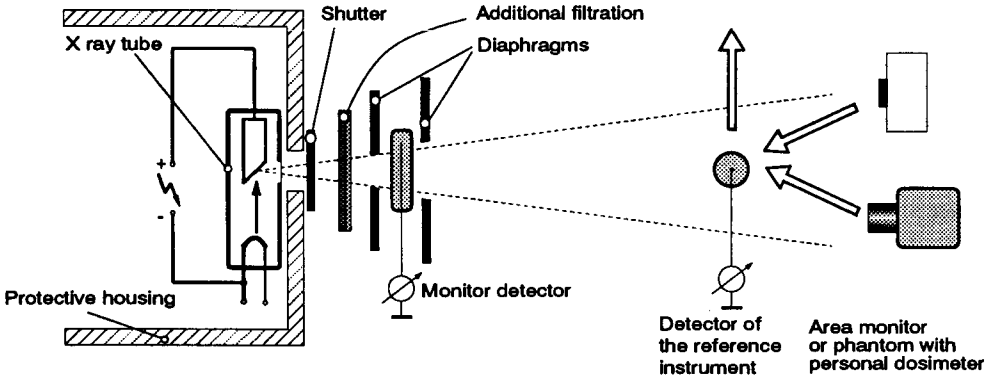

***Figure 1:** Calibration set-up for reference filtered X radiations consisting of an X ray tube with protective housing, beam limiting and shielding diaphragms, shutter, filter, monitor detector, detector of the reference instrument, and area monitor or phantom with personal dosimeter to be calibrated.*

Detailed numerical examples are given including the estimate of the uncertainty of the calibration factor and how to state this uncertainty in calibration certificates.

Concluding remarks

In assesssing whether a particular radiation monitoring instrument is adequate for its intended use and before it is used for the first time, it is important to have access to reliable type test data on that instrument. Often the instrument manufacturer does not possess facilities for the complete type-testing, and even sometimes cannot calibrate the instrument over the complete dose equivalent range with a reference radiation. There is a tendency for new users of radiation monitoring instrumentation to overestimate the facilities of the manufacturers. Each individual instrument should be calibrated before its first use and then recalibrated periodically. There are examples where inadequate calibration procedures have in the past caused large errors in some dose estimates.

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THE RUSSIAN STATE SYSTEM FOR QUALITY ASSURANCE OF RADIATION MEASUREMENTS

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The State system for ensuring quality of radiation measurements (hereinafter referred to as "System") has been establishing in Russia throughout a long period of time. When arranging the System, any measurements of physical quantities and parameters, which characterize sources and fields of ionizing radiations, were considered as the radiation measurements (RM).

The System has some distinguishing features caused by the factors which are basically characteristic for this field of measurements made in Russia.

The first factor resides in the fact that in Russia, the RM are currently central problems because of a great number of installations where the ionizing radiations exhibit activity. Among them there are radiochemical enterprises, atomic power plants, research reactors, Navy and Civil fleet with atomic engines, radionuclide diagnosis, radiation therapy etc.

The second factor is a great variety of the RM types, stemming from a large number of physical quantities and parameters: specific activity - with numerous radionuclides being measured, the flux and flux density of the charged particles and neutrons - within a wide range of energy, power of the absorbed and equivalent doses - within a wide power range for different types of radiations etc., as well as from a number of problems, the solution of which require such measurements as: technological (at the enterprises and plants), diagnostic (in medicine and manufacture), radioecological (at monitoring laboratories) and so on.

The third factor consists in the necessity to conduct quite often the RM for economic subjects such as atomic power plants or ecology organizations which have the same interests and should come to some responsible conclusions related to the fate both of manufacture and of population.

To satisfy the above requirements, the System should be arranged in such a manner that it could be used as an objective basis for solution of any problems including the disputable ones which refer to the RM.

Thus, from the very beginning, the System has been created with the aim to provide the metrological assurance of all economy branches of Russia in the field of the RM, using as a basis the centralized principle which means:

- reproduction of measurement units which are the main ones for a given field of measurements with the help of state standards;
- the legalized hierarchy chart for calibration of measuring means with an approved list of secondary and working standards and definite intercalibration intervals;
- availability of specifications regulating methods and accuracy limits of measurements for the whole park of the working measuring means being in usage.

At present the System is based on 11 state primary and special standards which provide to reproduce the main measurement units for the activity of radionuclides in sources, gases, aerosols, flux and flux density of the charged particles and neutrons, power of the absorbed dose of photon, neutron and beta-radiation as well as some other measurement units.

The size of the units reproduced by the standards is transferred to the working measuring means according to the hierarchy calibration charts including more than 50 working standards operating in the form of suitably designed dosimetry and radiometry set-ups installed at the leading enterprises or at the main metrological laboratories. As the certified radiation sources, they use radionuclide sources, x-ray apparatus, reactor channels, output accelerator beams. All measuring means including working standards, are tested every 3-5 years, the same interval being used for metrological certification of the measurement methods.

The System described, which to some extent appears to be bulky, has demonstrated its efficacy in carrying out the works on liquidation of consequences of the accident at the Chernobyl Atomic Power Plant.

First of all, it was found to be very important that all radiometers and dosimeters being used at the accident site, had been calibrated in the established units and that the measurements were performed according to the specified procedures. This enabled to assess the accident scale correctly, to bring out the major seats of origin of the damage and to forecast the development of radiation conditions. Thus, it was possible to organize and monitor the radiation dose of individuals promptly and on a large scale.

At the same time with the liquidation of the direct accident consequences, a demand arose for a radiation inspection of population, food, soil, water. This problem was promptly solved at a high level of reliability by means of manufacturing the standard specified measures of activity serving as full-scale models (imitators) of the tested samples taken at the contaminated locality to calibrate scales of radiometers and spectrometers, as well as whole body counters at the laboratories of sanitary and epidemiology services including those situated in the regions of the radiation contamination. At the same time, to certify the radiometer set-ups and to measure the radionuclide activity in the samples, some methods and corresponding procedures were developed. Thanks to the System, they were brought up to their direct executors.

Thus, thanks to the existence of the System, it was possible to solve the problem of carrying out reliable radiation measurements under the conditions of liquidation of the large-scale accident consequences. This in itself is a good evidence that the choice of a centralized system for metrological assurance within this field of measurements in Russia has been chosen correctly.

In 1981 at the IMEKO international symposium in Leningrad we underlined that the most rational method to provide the metrological assurance of devices measuring the specific activity of low activity media involves a wide use of standard solutions of radionuclides, whose application provides a high accuracy of instrument calibration over the whole range of specific activity values.

Numerous international comparisons of national standards of the radionuclide activity with the help of radionuclide solutions eliminate the possibility of rough systematic errors for widely used and well studied radionuclides.

The question of mutual acceptance of certification results of these solutions in different countries refers rather to the category of legislative than to technical problems. Hence the application of the SSR for the metrological assurance of measuring devices solves the problems of accuracy and traceability of the specific activity measurements.

For example, in Russia more than 60 certified standard solutions of radionuclides are manufactured in large quantities. The specific activity of main solutions is 10^5 Bq/g (for separate radionuclides 10^2-10^5 Bq/g); that is they are convenient both for measurement with the help of standard set-ups and for practical use.

The uncertainty of the most radionuclide solutions manufactured commercially is usual $\pm$ 2% at a confidence level P=0.99.

For environmental monitoring part of these solutions is used directly, the other part, used for the manufacture of imitators of soils and food is diluted by a factor of 100-1000 using precision methods of dilution as a result of which an additional uncertainty of 0.1-0.5% is introduced.

If at all stages the rules to avoid sorption are followed strictly (special vessels glassware, disposable pycnometers, nonradioactive carriers, etc are used) when applying a step-by-step dilution (by two-three stages), then it is a relatively simple matter to manufacture standard measures (simulators) with the uncertainty of 3-5% (P=0.99) at the level corresponding to the natural activity.

To provide precise laboratory measurement methods simulators with radionuclides existing in real samples are manufactured from solutions: ^{137}Cs, ^{134}Cs, ^{90}Sr, ^{90}Y or with other long-lived radionuclides, with close energies of particles. In addition ^{40}K from natural chemical agents is used. Volume standard measures of radioactivity based on natural disperse materials of various densities (quartz sand, resins, grain cultures, sawdust etc) containing one or several radionuclides are of considerable current use. For a short period of time low activity water solutions are suitable for use as well.

The volume measures of radionuclide activity imitate well real samples, they can be contained in any vessels, cups and dishes. Some of them are manufactured as a set with a nonradioactive carrier to measure background accurately. The importance of express methods for a bulk inspection of foodstuffs should be noted. We take the word "an express method" to mean radiation monitoring without taking a sample for an analysis, that is, we measure activity directly in commercial containers, boxes, cans, barrels etc. Devices with scintillation gamma-detectors are used to conduct such inspection. It is necessary to separate products known to be "noncontaminated " from products

whose radiation contamination is significantly below permissible limits, and to detect "contaminated" products to be tested accurately. The metrological assurance of such express methods of measurements involves manufacture of special volume measures of specific activity large in size (e.g. cans or boxes). In this case the usual volume measures of radioactivity is mixed with nonradioactive carrier in two or three steps. The uncertainty of such a measure is within 5%.

The above mentioned examples show the convenience, effectiveness, reliability, versatility of the metrological assurance of measuring means based on measures manufactured on the base of certified standard solutions of radionuclides.

In connection with the accident at the Chernobyl Atomic Power Plant the problem of the metrological assurance of measurements of the radionuclide activity of the personnel operating and maintaining nuclear installations and of the population living in regions exposed to radioactive contamination has become urgent. To check the content of radionuclides in human body direct measurements of γ-radiating radionuclides activity are carried out. The most authentic data on the radionuclide content in the human body are obtained using body radiation counters that have been previously calibrated against standard special-purpose sources. Semiconductor detectors based on Ge and scintillation detectors based on NaI(Tl) are employed as gamma-ray detectors in most commonly used counters. Before usage these counters must be certified taking into account age anthropometric distinctions of a person. To this end special phantoms imitating an adult, teenager and child have been developed and are used. Built-up phantoms consisting of standard special-purpose volume radionuclide sources are used; wherein 35 standard sources imitate an adult's body and 6 standard sources -that of a child.

The system developed by now for the metrological assurance of radionuclide activity measurements in the human body includes:

- development and investigation of phantom sources imitating an adult, teenager and child;
- certification of these sources as standard measures of gamma-ray radionuclide activity;
- development of procedures for counter certification;
- metrological certification of these spectrometers on a periodic scheduled basis.

As a rule counters are certified against the activity values of ^{137}Cs gamma-ray radiation. Counters certified as standard activity measures of ^{60}Co, ^{241}Am etc are used as well.

The activity of radionuclides found in the human body is measured by the relative method, comparing the pulse count rate of gamma-rays from the person with the pulse count rate of gamma-rays of the same energy from the phantom which serves as the standard activity measure of the radionuclide present in the human body. The radionuclide activity values obtained in such a way are used to calculate the absorbed (equivalent) dose a person takes due to the internal irradiation.

In addition to the above-listed programs the Government of Russia is currently funding a number of others within which a system for ecological monitoring of the country is being created including radiation monitoring. A network of testing laboratories is being created. These laboratories are accredited for the right to carry out radiation measurements for mapping territories, the hydrosphere, the atmosphere, for certification of building materials, food, wood etc. This allows not only to define localities contaminated with radionuclides of the technical origin but to forecast correlations between health of the people living in this locality and characteristics of the locality, given in the certificate.

Once the System has been created, it is being continuously developed and improved. One of the most urgent problems which it faces now is the creation of a network of laboratories for monitoring the radiation. These laboratories will be accredited for the right to carry out the radiation measurements according to the rules regulated by ISO/IEC standards for the testing laboratories included in the system of radionuclide product certification.

The System is being continuously developed and improved. Radiation measurements performed in the certified testing laboratories in compliance with the ISO/IEC standards will make it possible to harmonize the Russian System with the international one.

NEW IAEA COMPENDIUM OF NEUTRON DOSE CONVERSION COEFFICIENTS, SPECTRA AND DETECTOR RESPONSES FOR RADIATION PROTECTION.

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INTRODUCTION

The recommendations made by the International Commission on Radiological Protection (ICRP) in their Publication 60[(1)] mean some major changes in concepts of radiation quantities. The protection quantity effective dose equivalent (DE), H_E, has been replaced by the effective dose, E, radiation weighting factors, w_R, have been introduced for protection quantities and the $Q(L)$ relation used with operational quantities has been modified. These recommendations have also been discussed by the International Commission on Radiation Units and Measurements (ICRU) in their Report 51[(2)]. A joint ICRP-ICRU task group has collected the fluence to dose conversion coefficients for the new quantities.

These recommendations brought a renewed focus on the importance of neutron spectra in occupational radiation protection. Indeed, in recent years there was considerable progress made in developing new and improving existing spectrometers and dosemeters and in the determination of spectra at work places[(3)].

The availability of this data has prompted the International Atomic Energy Agency (IAEA) to prepare a supplement to TRS 318[(4)]. In addition to the presentation of the new data, the supplement will include spectrum weighted values of the protection and operational quantities for the original spectra. As with the prior publication the data will be available on diskette. A feature of the new work is software that will allow the user to perform spectral weighting with other data that are not available in either publication.

The aim of this paper is to give a sneak preview. In the following, the formalism used in the new publication is summarized. Some examples are given to illustrate the data contained, i.e. conversion coefficients, spectra and responses. Finally, the use of the supplement is discussed.

FORMALISM

The primary physical quantity for the assessment of derived radiation protection quantities for neutrons is the neutron fluence, Φ, and its distribution density in neutron energy, E, i.e. the neutron spectrum, $\Phi_E(E)$.

The instruments used are described by their neutron fluence response, $R_\Phi(E)$, and the reading of an instrument exposed to a neutron spectrum, M, is computed as the energy integral over the product of $\Phi_E(E)$ and $R_\Phi(E)$. Possible influences due to dead time, fading or background are not discussed.

The quantity to be determined is a specific dose equivalent (DE), H, which is computed as the energy integral over the product of $\Phi_E(E)$ and the fluence-to-DE conversion factors, $h_\Phi(E)$.

In the context of this compendium only ratios of fluence weighted quantities are of interest and it is sufficient to restrict the discussion to *normalized* fluences and to assume that the fluence is

uniform throughout space. The neutron fluence is in practice also distributed in the angle and the fluence response function of an instrument and the DE quantity considered may depend on the angular distribution of the neutron fluence. However, for the sake of clarity, this compendium is restricted to the consideration of broad parallel beams and few idealized irradiation geometries. Measured or computed neutron spectra are in most cases originally obtained as group fluences, φ_i, in energy bins, (E_i, E_{i+1}). A fixed energy grid is used in the compendium. Any bin is described by its lower and upper boundaries. Sixty energy bins are used from E_1 = 1 meV to E_{61} = 0.5 GeV. It is understood that the fluences are normalized: the value of $\Phi = \varphi_1 + ... + \varphi_{60}$ is by definition 1 cm^{-2}. The mean energy of a bin is defined as the geometric mean of its lower and upper boundary, i.e. $\overline{E}_i = \sqrt{E_i \cdot E_{i+1}}$.

The fluence response function of instruments and the fluence-to-DE conversion coefficients are in general continuous functions of neutron energy. In order to avoid any ambiguity the integrals needed for the calculation of spectrum averaged quantities are replaced by sums and as discrete representation it is defined:

$$M = \sum_{i=1}^{i=60} \varphi_i \, r_i \qquad \text{and} \qquad H = \sum_{i=1}^{i=60} \varphi_i \cdot h_i \qquad (1)$$

where $r_i = R_\varphi(\overline{E}_i)$ and $h_i = h_\varphi(\overline{E}_i)$. Depending on the width of an energy bin and the energy dependence of the fluence response function or the conversion factor this procedure may lead to slight deviations as compared to the results obtained from correct integration. In general, these deviations are irrelevant within the scope of this compendium and within its main purpose to study ratios of spectrum averaged quantities. However, as far as the calibration spectra are concerned, it is felt to be perturbing, if the values provided differ from the recommend ones. Therefore a table with recommended mean fluence-to-dose equivalent conversion factors is given in the supplement.

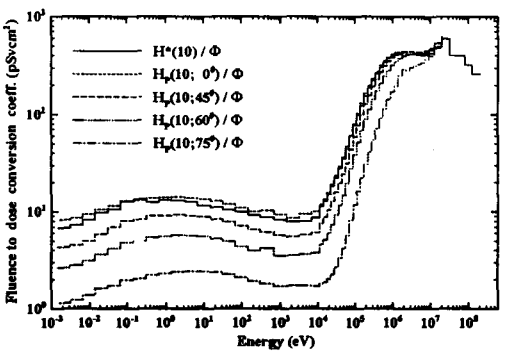

Figure 1. *Neutron fluence to dose equivalent conversion coefficients*[5] *as a function of incident neutron energy.*

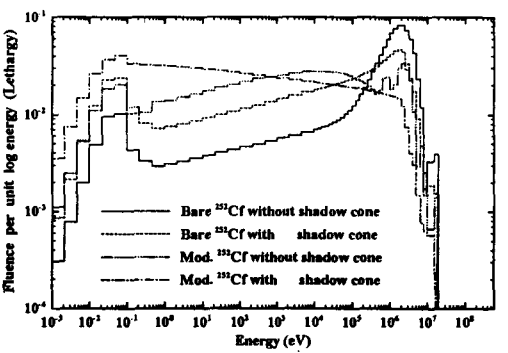

Figure 2. *Calibration spectra in a bunker room*[6] *as a function of incident neutron energy.*

QUANTITIES, SPECTRA AND RESPONSE FUNCTIONS

The *operational* quantities for area monitoring, i.e. the ambient DE[1,2], $H^*(10)$, and for personal monitoring, i.e., the personal DE[2], $H_p(10,\alpha)$, are shown in Figure 1. The *protection* quantity effective dose[1], E, -not shown here- will be given for the standard irradiation geometries, i.e. AP, PA, LAT, ROT and ISO.

Examples of new calibration spectra determined by Bonner spheres spectrometry and calculations[6] are shown in Figure 2. It is interesting to know, that a sequential irradiation in these spectra allows to realize virtual spectra with interesting dosimetric properties.

Responses of some new survey instruments[7,8] and etched track detectors[9-11] are shown in Figures 3 and 4, respectively. The supplement will also feature new response functions for Bonner spheres and Bubble detectors.

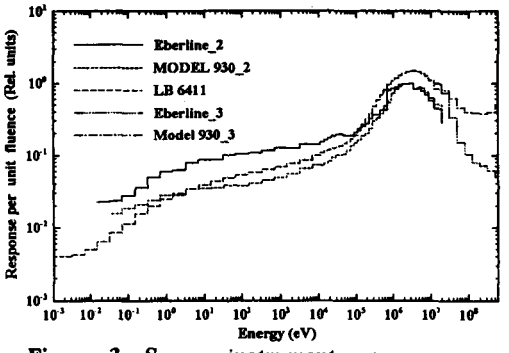

Figure 3. *Survey instrument responses as a function of incident neutron energy. Eberline and Model 930 cf. ref. 7 and LB 6411 cf. ref. 8.*

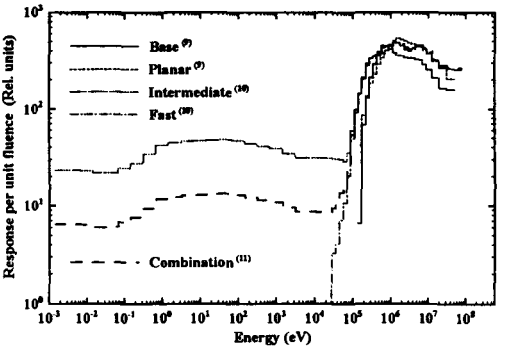

Figure 4. *Etched track dosemeter responses*[9,10,11] *as a function of incident neutron energy.*

THE USE OF THE CATALOGUE

Printed tables and the data and the software that will be supplied on diskette will assist in practical neutron radiation protection in several ways.

- The data on fluence conversion coefficients to the new protection and operational quantities to be included provide a complete set of data for routine neutron radiation protection.
- Many new spectra usable for calibration and found in work places provide valuable assistance in deciding where and how to calibrate instruments and to judge the dosimetric properties of actual work places of interest.
- The influence of the change of DE quantities for actual work places and for particular instruments can be studied. This is of great importance for the decision whether existing instrumentation is still adequate with the new DE quantities.
- Finally, valuable support is given in selecting new appropriate instrumentation if necessary or provide correction factors if the old instrumentation must be used.

In summary, the supplement will assist in assuring continued high quality in neutron radiation protection practice.

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POSTER: RADIATION PROTECTION OPTIMIZATION AT DIFFERENT WORKING POSTS

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Increasingly sophisticated elecronic dosemeters have been introduced into the market over the past few years. The use of electronic dosemeters is becoming more and more widespread as a result of demands being made for dosimetry management on both the national and international level.

The dosicard system involving the use of credit-card sized dosemeters and a data acquisition and processing environment is well suited to satisfy demands in this field. It should be noted, that in addition to assuring the traditional functions of dose and dose rate equivalent measurements, these dosemeters also provide an autonomous management of cumulated doses (on a daily, weekly, three-monthly or annual basis), which can be consultated in real time by the wearer.

This system also allows the restitution of the previous hundred measurements (through variable increments) together with time, date and duration information for events involving the exceeding of predetermined thresholds (defined in terms of dose and dose rate equivalents). These characteristics have incited studies on the use of such dosemeters to determine the characteristics of working posts. Experimental studies have been conducted over a period of several months in a radioelement production laboratory and in a hospital service specialized in radiochemistry and medical imagery.

The main characteristics of the Dosicard system are recalled. The results of these two series of studies are then examined. A demonstration is then provided on the way in which this sort of approach influences the behaviour of the wearer relative to radioprotection specialists.

PERSONNEL MONITORING AT JINR

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1. INTRODUCTION

For more than 40 years the Division of Radiation Protection (DRP) has provided personnel monitoring (PM) at the Joint Institute for Nuclear Research (JINR). The JINR have a set of basic nuclear fasilities, such as: 10 GeV proton and light nuclei Synchrophasotron and superconducting accelerator of relativistic nuclei "NUCLOTRON", the U-400, U-400M, U-200 heavy ion cyclotrons, the 680 MeV proton phasotron, the fast neutron pulsed reactors IBR-2 and IBR-30. These facilities are a powerful sources of ionization radiation filds with wide energy range and complex composition. The DRP carries out a sistematic personnel monitoring service of gamma, beta, X-ray and neutron exposures at the JINR. The number of persons monitored has not changed considerably for the last years and at present it amounts to about 2100 persons, including visiting scientists and outside contractors(120). The PM for external exposure is still based on film badge IFK-2,3 with TLD cards for gamma dosimetry, the double film ORWO RD-3,4 for X-ray and beta and nuclear emulsion MK-20 for neutron. The film badge IFK-2,3 contains the different filters combination and open window.

2. METHOD AND MEANS

For gamma dosimetry are used cards containing two natural LiF chips (G1) of identical dimensions (3x3x1mm) mounted between Teflon foils. A hope code is used for identication of the dosimeter cards. One TL-detector is placed behind a filter of 850 mg/cm^2 Pb and another one behind open window. The dose equivalent randges are 0,1 - 1000 Sv. For photon the energy randges 100 keV - 3 MeV the error is not more than 40%, at low energy the error is increasis. The system consists HARSHAW 2271, containing an automated reader and autoranging picoammeter. The absorbed dose beta radiation above 0,12 MeV are indicated by means of film by comparing the optical densities behind the three filters and open window. The absorbed dose randge are 0,2 mGy - 0,15 Gy. The TL-detectors of $CaSO_4$ thickness 0,1mm are used in finger ring dosimeters ters for beta and X-ray. The neutron dosimeter are based on nuclear emulsion MK-20 thickness 20 μm supported on a 140 μm cellulose triacetate base. The MK-20 are packed into the correction packet to correcte the energy dependence. The correction packet with MK-20 are placed behind a TLD card in film badge - IFKn-method /1/. Remresponse of the MK-20 are determined for energe randges from thermal neutron to 20 MeV /2/. The dose equivalent for high-energy hadron (above 20 MeV) are determined by the star prodaction in the nuclear emulsion (three and more rays). Most of neutron dose mesurements (without high-energy hadrons) made by film are within the error randge 50%. For high-energy stray filds IFKn-method is overestimates the dose equivalent by a factor of two. In these cases are used corretion method for reading of film dosimeters /1/.

Latent image fading of track density from high-energy hadrons is not more than 30% and from Cf-252 source neutrons is not more than 10% for three months. A data bank realises storing, updating, retrieval, statistics processing and output of the personal and dose information for over 2100 persons, controlled quarterly for external radiation /3/. The basic compontnts of the data bank are a data base (personal and dose files as current as archival) and a package of application programs for the work with the data base in the dialog mode by means of menus. The data bank was organized on the basis PC AT-386 with assistance of the relational DBMS dBASEIII PLUS and the Clipper compiler for the interpretive dBASEIII programming language, working under the control MS DOS, version 6.10.

3. CALIBRATION PROCEDURES

The gamma dosimeters are calibrated free in air of Co-60 radiation. The calibration irradiation are in terms of exposure. The exposure values are converted to dose equivalent with conversion 0,97 cSv/R. The area of the open window of the film badge with ORWO is calibrated in additional to dose equivalent of beta-rays of Sr-90+Y-90. In routine the neutron dosimeters are calibrated free in are of Cf-252 (for dosimeters are used in Laboratories of Neutron Physics and of Nuclear Reactions) and Pu-Be (for anothes laboratories). For calibration are used the fluens-to-dose equivalent conversion factor ($3{,}3{*}10^{-10}$ Sv$*cm^2$ for Cf-252 and $3{,}6{*}10^{-10} Sv * \mathrm{cm}^2$ for Pu-Be), factor on-phantom and free -in-air calibration and factor normal and rotatory irradiation for calibration /4/.

4. PERSONNEL DOSES

About 2100 radiation workers are subject to individual monitoring of external radiation in the JINR. This number includes 1500 persons monitored for neutron exposures. About 60 persons wear additionally film ORWO. Dosimeters are issued normally for a period of 3 months. Each monitored person has a personal card, where the values of all doses recieved and recorded. The individual dose values besides being stored in an electronic data bank also recorded in the personal radiation pasport of each radiation worker. The results of dose measurement for whole body are summed each year. The mean annual dose equivalent for person has not changed considerably for last 15 years and amounts about 2 mSv. Annual reference dose for radiation worers is 50 mSv in Russia. The mean annual dose ($\bar{H}$) and collective dose (H_Σ) is shown in fig.1. As an illustration fig.2 show the annual dose distribution for 1994. About 40% persons have annual dose below the detection limit (about 1 mSv). The exposures registered are very low, for exsample, in 1994 about 99,8% less than 15 mSv and 0,2% between 15 and 50 mSv per year. Maximum value was 20 mSv.

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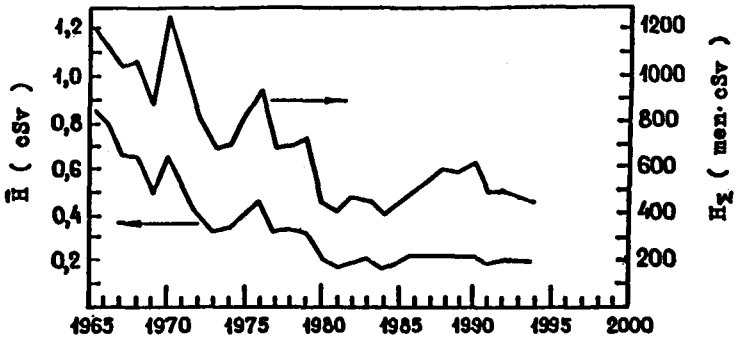

FIG.1 THE COLLECTIVE (H_Σ) AND EVERAGE ($\bar{H}$) DOSE EQUIVALENT

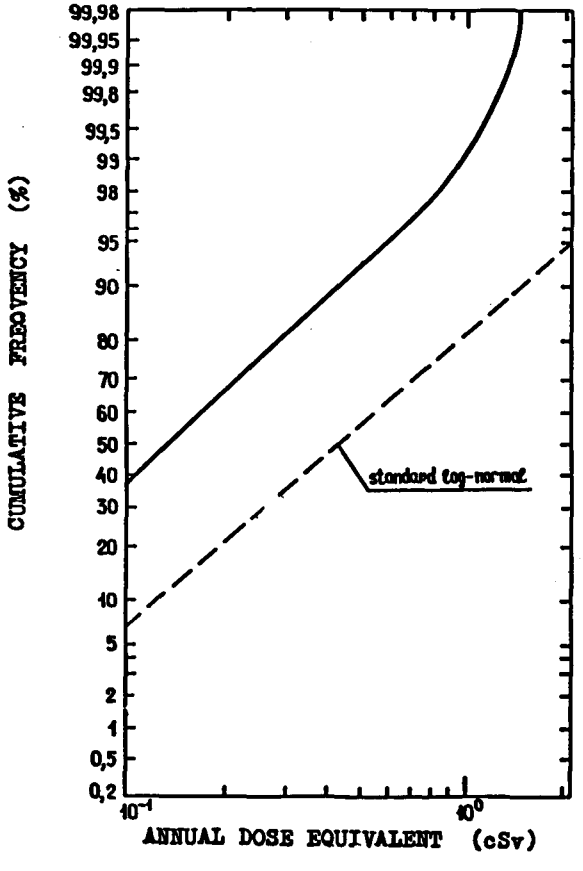

FIG.2 DOSE DISTRIBUITION FOR 1994 FOR PERSONNEL JINR

THE CURRENT STATUS OF NATIONAL RADIATION METROLOGY LABORATORY IN TAIWAN

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INTRODUCTION

The Institute of Nuclear Energy Research (INER) has been entrusted by the National Bureau of Standards (NBS) , Taiwan to establish the national radiation metrology laboratory (1-4) since 1991. Three major radiation fields such as photons and betas, neutrons and radioactivity measurement, have been studied by the INER. The established standards and techniques are applied in personnel dosimetry proficiency test in addition to instrument calibration. Whenever possible INER also actively takes part in international intercomparison programs to ensure the uniformity of the established standards and techniques with the international community.

PHOTONS(1)AND BETAS(2)

High energy photons are produced by radionuclides ^{60}Co and ^{137}Cs with average mono-energies 1.25 MeV and 0.662MeV, respectively. Since there have no radionuclide emitting photon energy below 300 keV with suitable half-life, X-ray machines are used to generate photons with energies from 10keV to 300 keV. Self-made graphite ionization chambers are used to measure the exposure from high energy photons. For low energy photons free-air ionization chambers are used. Transfer ionization chambers are used for intercomparison with Electrotechnical Laboratory (ETL, Japan) and National Institute of Standards and Technology (NIST, USA). The results show about 1-2% deviation. For beta fields, three sets of radionuclides $^{90}Sr+^{90}Y$, ^{204}Tl, and ^{147}Pm are used to create beta irradiation fields which are determined directly by an extrapolation ionization chamber. The results of intercomparison with PTB (Germany) show about 3% deviation.

NEUTRONS(3)

A room with dimension 5×6×9m is used for neutron irradiation. The four walls and the ceiling are made of aluminum in order to minimize neutron backscattering. Neutron sources can be hanged about 5m above the ground which is made of concrete. The neutron scattering effect is studied thoroughly. Radionuclide neutron sources such as ^{252}Cf situated at the center of a D_2O sphere with radius of 0.15m is used to simulate the neutron field outside the containment of a nuclear power plant. Neutron detectors are calibrated with bare and D_2O moderated ^{252}Cf respectively. The neutron emission rate of a radionuclide neutron source is determined directly by a manganese sulfate bath system.

Mono-energetic neutron fields produced by accelerating protons with a 7MV van de Graaff accelerator hitting lithium target are measured by various suitable techniques such as associated particles, proton recoil telescope and proportional counter.

Neutron spectrums are determined by bubble detectors, activation foils, Bonner spheres and NE-213 detector whenever is appropriated. Various computational techniques are applied to calculate the neutron spectrum of a D_2O moderated ^{252}Cf. The results are in good agreement with that of experiment.

RADIOACTIVITY MEASUREMENT(4)

The activity of radionuclide is determined directly by conventional $4\pi\beta-\gamma$ coincidence method which is good for simply decaying radionuclides such as ^{60}Co. For complex decaying radionuclide such as ^{133}Ba, $4\pi\beta-\gamma$ coincidence method coupled with two dimensional extrapolation method is used. Radionuclides distributed by ETL are used for intercomparison. Good agreement shows about 1% deviation but, in some cases, 7% deviation can be achieved which need some more careful studies. A system of $4\pi\gamma$ ionization chamber is used as the secondary standard to determine the activity of a radionuclide because of its long term stability and easy calibration. Other secondary standards for radioactivity measurement are high purity germanium system and $2\pi\alpha-\beta$ system.

PERSONNEL DOSIMETRY PROFICIENCY TEST(5)

INER is also entrusted by the Chinese National Laboratory Accreditation (CNLA) as the testing laboratory to carry out the proficiency test of personnel dosimetry service. Six personnel dosimetry service laboratories participated the performance tests in 1991 and 1993. In these two tests only gamma and X-ray were carried out. All of the laboratories passed the tests. Recently a full span of the test including beta and neutron dosimetry in addition to photon dosimetry was carried out. About nine personnel dosimetry service laboratories took part in the test.

SUMMARY

INER has paid much effort to establish the primary radiation standard in this country since 1991. The radiation fields include photons, betas and neutrons. In addition to instrument calibration, these fields are also applied to the personnel dosimetry proficiency test.

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SEVEN YEARS OF INDIVIDUAL MONITORING SERVICE IN CUBA

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ABSTRACT

The Center for Radiation Protection and Hygiene (CPHR) has been carrying out from 1987 the individual monitoring of workers occupationally exposed to the ionizing radiations in the Republic of Cuba, excepting those that employ X rays in diagnostic radiology. In this paper the results of the individual monitoring services during the period 1987-1993 are discussed. For all occupational practices the current system of dose limitation established in the country with 50 mSv as limit of annual dose is satisfied. The distribution in all occupational practices other than Nuclear Medicine and Gammatherapy is characterized by having more than 95% of the controlled personnel with an effective dose lower than 5 mSv. In the case of Nuclear Medicine and Gammatherapy, more than 80% of the workers were below that dose value. For the practices evaluated in this paper the possibility of assuming the system of dose limitation recommended by ICRP is evident. The evaluations carried out for the introduction of operational quantities Hp(0.07) and Hp(10) in dose assessment procedure are presented. The expressions obtained during the characterization of the film badge dosimeter, in terms of operational quantities guarantees a deviation of response of the dosimeter with depending upon energies, lower than 20 %.

INTRODUCTION

The Center for Radiation Protection and Hygiene (CPHR) has been carrying out from 1987 the individual monitoring of workers occupationally exposed to the ionizing radiations in the Republic of Cuba, excepting those that employ X rays in diagnostic radiology. The services offered refer to the control of external exposure by employment of film badge dosimeters for whole-body irradiation and TLD rings for the control of the dose in hands. "In vivo" and "in vitro" internal contamination monitoring for different radionuclides (^{131}I, ^{125}I, ^{32}P, ^{3}H, ^{14}C, ^{35}S) is also included.

MATERIALS AND METHODS

The individual monitoring of external exposure was carried out by using film badge dosimeters. The characteristics of the commercial "whole-body" dosimeters utilized, such as the procedures of calibration and methods for dose assessment are presented in [1]. Until 1991 the radiation monitoring film used in the service was the ORWO RD3-4 made in German Democratic Republic. That model was substituted by the AGFA-GEVAERT film "Personal Monitoring". The Service included individuals occupationally exposed to external radiation , who worked in following practices : Nuclear Medicine, Gammatherapy, Roentgentherapy, Research with ray X, Customs Control, Unsealed Sources, High Activity Sources, Geological Prospection, Industrial Radiography with ray X and gamma radiation. The monitoring period was for one month until 1990. Starting from 1991 the monitoring period was extended to three months.

In 1990 the individual monitoring of exposure in hands with TLD (LiF) given by the IAEA was established. That Service was limited to 50 users. For selection of workers to be controlled was kept in mind the dose received for the whole body in practices Nuclear Medicine and Unsealed Sources.

For the internal contamination monitoring, the effective dose calculation was carried out by applying the metabolic models, recommended by the ICRP in their publications [2]. The procedures of calibration and methods for dose assessment are presented in [1].

As a consequence of the recommendations of the ICRP [3], the response of the film badge dosimeter was characterized per unit of personal dose equivalent at both depths of 10 mm and 0.07 mm, (Hp(10) and Hp(0.07) respectively) [4]. Two reference methods for determination of the dose were applied, one in which dosimeters for calibration, irradiated with ^{137}Cs are used, and another in which two sets of dosimeters were employed for

calibration, one irradiated with a ^{137}Cs source and the other one with a ^{241}Am source. As a procedure for dose assessment the linear combination of the dose calculated for the different sectors of the dosimeter are analyzed.

RESULTS AND DISCUSSION

In the Film Dosimetry Service, values of mean annual effective dose per occupational practice (Table 1) evidence that practices with largest mean doses for external irradiation are those of Nuclear Medicine and Gammatherapy. The total mean dose in these years oscillates from 1.2 to 2.6 mSv (Table 2). Only 42 workers had exceeded the 15 mSv (three tenths of permissible limits of annual dose). Practices of Nuclear Medicine (17) and Gammatherapy (16) had the major number of workers involved.

Table 1: Mean annual doses for practices. Period 1987-1993. Film dosimetry service.

Occupational practices	Mean Annual Effective Doses [mSv]						
	1987	1988	1989	1990	1991	1992	1993
Nuclear Medicine	3.21	3.16	4.33	3.89	2.44	3.25	2.71
Gammatherapy	3.19	3.60	3.59	4.06	2.39	1.94	1.65
Roentgentherapy	2.49	1.28	1.42	0.72	2.55	1.34	1.15
Research with rays X	0.74	0.74	0.73	0.76	0.57	0.54	0.43
Customs Control	1.29	0.80	0.72	0.80	0.30	0.19	0.36
Unsealed Sources	2.51	2.44	2.58	2.71	1.07	1.21	0.84
High Activity Sources	2.40	2.41	2.56	2.41	1.00	0.82	0.57
Geological .Prospection	2.40	2.40	2.50	2.41	1.25	1.03	0.33
Industrial Radiography with rays X	0.82	0.73	0.98	0.75	0.26	0.36	0.48
Industrial Radiography with gamma radiation	2.80	2.60	2.80	2.81	1.14	0.75	0.67

Table 2: Results of the individual monitoring with film badge dosimeters

Year	Number of controlled workers	Mean effective dose [mSv]	Workers with doses above 15 mSv	Maximum Dose [mSv]
1987	1263	2.47	2	16.8
1988	1475	2.38	2	16.1
1989	1714	2.48	9	24.2
1990	1542	2.55	11	42.7
1991	1459	1.28	4	44.2
1992	1079	1.53	8	34.8
1993	1002	1.16	6	22.8

The variation of annual mean effective dose for different occupational practices, shows that in no case it exceeds the 5 mSv. The percentage of workers which received doses lower than that value, is in all practices over 95%. Practices of Nuclear Medicine and Gammatherapy where the accumulative frequency is over 80% are exceptions. No worker had exceeded the permissible dose limit of 50 mSv established for one year.

In the Individual Monitoring Service of exposure in hands, the maximum number of workers controlled during a year was 34. No worker had exceeded the annual limit of dose equivalent for a specific organ. The higher annual doses registered were: during 1990- 151 mSv, in the practice of Unsealed Sources in 1991- 86 mSv and in 1992- 96 mSv both in Nuclear Medicine.

In the Internal Contamination Monitoring Service the higher number of detected incorporations was found for radionuclides ^{131}I, ^{125}I and ^{32}P. In the case of Nuclear Medicine for the ^{131}I, values of committed dose equivalent oscillated between 2.50 - 28.76 mSv, the last being the largest committed dose equivalent detected throughout the control periods. The values of the mean committed effective dose for each occupational practice did not reach the level of 5 mSv.

During the last three years the lowest values of annual effective dose was registered for all practices. This situation could reflect the effectiveness of the radiation protection measures implemented in each practice and of the complete supervision system in Cuba.

As a result of the characterization of the film badge dosimeter in terms of operational quantities it was verified that the lowest deviation of the response of the dosimeter with the energy occurs when the 5 sectors of film badge dosimeter observed are combined [4]. Expressions that guarantee a deviation of the dosimeter response with depending upon energies lower than 20 % were gotten. Both methods of reference dose showed a comparable effect in the smoothing of the energetic dependence. It is observed that only for energies near 20 keV differentiated calculations of Hp(0.07) and Hp(10) make sense.

In order to evaluate the practical implications that the reduction of the dose limit to value lower than 50 mSv could have, the following analysis was performed. The frequency of cases above 20 mSv oscillate between 2 - 3 workers per year. This number represents a 0.2% of the total of the personnel controlled and it could be possible to reduce it with a more specific control of the practice where that is manifested [1]. Taking into account these aspects, it could be affirmed that is possible, in a practical situation, to reduce the current dose limit without great investments in order to guarantee compliance with the new dose limitation system, at least in those practices controlled by the CPHR.

CONCLUSIONS

For all occupational practices the current system of dose limitation established in the country with 50 mSv as limit of annual dose is satisfied. The workers receive annual mean effective dose that does not exceed three tenths of this limit. The distribution in all occupational practices except Nuclear Medicine and Gammatherapy is characterized by having more than 95% of the controlled personnel with an effective dose lower than 5 mSv. In the case of Nuclear Medicine and Gammatherapy, more than 80% of the workers were below that dose value.

The low value of mean effective dose evidences that the radiological protection requirements in our country guarantee the avoidance of significant doses for external and internal exposure. For the practices evaluated in this paper, the possibility of assuming the system of dose limitation recommended by ICRP is evident.

The expressions obtained during the characterization of the film badge dosimeter, in terms of operational quantities guarantees a deviation of response of the dosimeter with depending upon energies, lower than 20 %. Only for energies near 20 keV differentiated calculations of Hp(0.07) and Hp(10) make sense.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Development of a Testing and Calibration Service for Protection Level Test Instrumentation

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MAJOR SCIENTIFIC TOPIC NUMBER 3 (see page 7)

ABSTRACT (See instructions overleaf)

The RRPPS provides a service for calibration of external x-ray, beta and gamma dose and dose-rate meters and alpha, beta and gamma contamination monitors. An upgrading of the RRPPS testing and calibration facility is described. In this exercise latest UK HSE, ISO & NAMAS guidelines have been considered. Results are presented of tests of the ^{137}Cs irradiation jig for calibration of gamma dose-rate meters. Spectral measurements and Monte Carlo simulations have been performed at the defined calibration points; the applicability of the standard air kerma to ambient dose equivalent conversion factor is examined.

Calibration conditions are not intended to reflect field conditions. However in order to also fulfil the RRPPS role as an RPA, quantitative estimates of the differences between calibration and field conditions have been made for a limited number of cases. Typical working surfaces were uniformly contaminated with known activities of ^{99m}Tc, ^{125}I and ^{32}P and measurements were made using commonly encountered instruments.

MONITORING SURVEY INSTRUMENTS BEHAVIOUR IN STANDARD LOW ENERGY X-RAYS

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INTRODUCTION

The Calibration Laboratory of São Paulo offers calibration services for monitoring survey meters with gamma, beta, alpha and X radiation. For the calibration of these instruments with X-rays, at Radioprotection level, seven standard low energy X-rays fields were established. Five of them are used at the National Physical Laboratory (NPL) (1), with energies between 16 and 38 keV, and two are recommended by the International Standard Organization (ISO) (2), with energies of 33 and 48 keV.

Different survey meters such as ionization chambers and Geiger-Müller detectors were studied on relation to their energy dependence. These instruments are used for area monitoring of gamma radiation, but some of them detect low energy X-rays too. The energy spectra in area monitoring are usually large because of the scattered radiation. Therefore it is important to know the behaviour of these instruments in a wide energy range.

MATERIALS AND METHODS

The low energy X-rays calibration system consist of a Rigaku-Denki Generator, model Geigerflex, with a Philips tube model PW/2184/00 (Tungsten target and Beryllium window). The voltage and the current can be varied between 20 and 60 kV and 2 and 80 mA, respectively. Table 1 shows the characteristics of the low energy X-rays qualities established at the Calibration Laboratory. The measurements were taken at 200 cm distance from the target, in a field of 25 cm diameter. Six ionization chambers and four Geiger-Müller detectors of different models and manufacturers were tested (Table 2).

Table 1-Characteristics of low energy X-rays qualities

Voltage kV	Additional Filtration mmAl	mmCu	Effective Energy keV	1st. HVL mmAl	mmCu
20	0.92		16	0.35	
25	1.70		20	0.66	
30	2.70		24	1.02	
40	4.92		31	1.95	
50	1.12	0.23	38	3.27	
40 (ISO)		0.21	33		0.086
60 (ISO)		0.57	48		0.232

Table 2-Characteristics of monitoring survey instruments

Ionization Chambers	Model	Geiger-Müller Detectors	Model
Bicron	RSO-5™	Eberline	HP 270
Nardeux	Babyline 31	Eberline	HP 290
Nardeux	Babyline 81	Nortron	NMR-1000
Victoreen	450	Victoreen	Minimonitor II
Victoreen	470A		
Victoreen	471		

RESULTS

Several monitoring survey instruments were tested, but the results of only the most representative of them are shown in this work. In the Figures 1 and 2 the results obtained with some ionization chambers can be observed; they presents an energy dependence between 7 and 33 % in the 16-48 keV range. The most suitable of these radiation detectors is the Victoreen 470A, although all of them can be used in this energy range.

In the case of the Geiger-Müller detectors, they presents a very high energy dependence, as can be seen in the Figures 3 and 4.

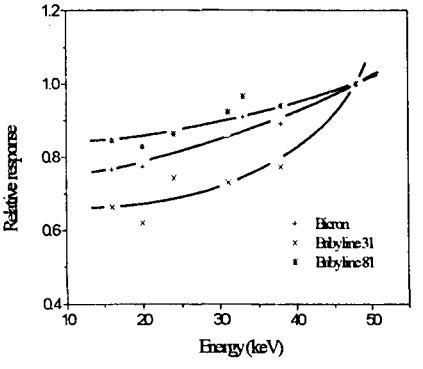

Figura 1 - Energy dependence of ionization chambers

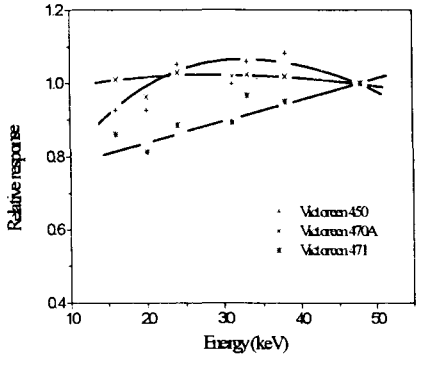

Figura 2 - Energy dependence of ionization chambers

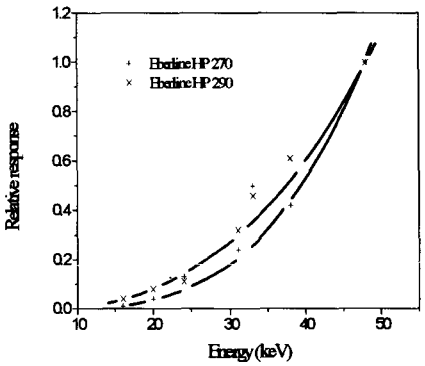

Figura 3 - Energy dependence of Geiger-Müller detectors

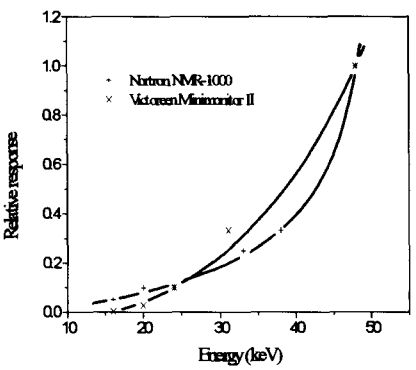

Figura 4 - Energy dependence of Geiger-Müller detectors

CONCLUSIONS

All tested instruments, ionization chambers and Geiger-Müller detectors, can be used for low energy X-rays detection, but the results show the importance of knowing their energy dependence in order to allow the application of correction factors on the measured values of exposure rates.

ACKNOWLEGMENTS

The authors acknowledge the technical assistance of Mr. Marcos Xavier and the partial financial support of Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq).

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE ESTABLISHMENT OF STANDARD X-RAYS USING A TANDEM SYSTEM OF IONIZATION CHAMBERS

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

A sequential Tandem system of plane parallel ionization chambers developed at IPEN was used to determine the radiation qualities of an X-rays system (300 kV) for instruments calibration purposes at radiotherapy level. The Tandem arrangement is formed by chambers with different energy dependences. The results were compared with those obtained through the conventional method of HVL determinations using thimble chambers and absorbers with different thicknesses.

Work partially supported by CNPq - Conselho Nacional de Desenvolvimento Científico e Tecnológico, Brazil

COMPARISONS INHOMOGENEITY OF PHOTON FIELDS AND BACKSCATTER FACTORS FOR PMMA AND WATER SLAB PHANTOMS

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Ivana Avramović and Olivera Čuknić

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ABSTRACT

The dose rate distributions of photon fields for calibrating personal dosimeters are measured free-in-air and in front of the PMMA and water phantoms. The backscatter factors and relative inhomogeneity of photon fields at both of phantoms are compared. From radiation protection aspect, for the hospital staff at Rö diagnostics, we have discussed the preferences and defects on the occasion of use for everyone of phantoms separately. It turned out that in to give the objective opinion for occupational health, we have to consider to the positioning of the personnel dosimeter during their calibration in photon fields X ray apparatus of the Standard Dosimetry Laboratory.

INTRODUCTION

In the many investigations, it were appearing that the staff at Ro diagnostics and radiotherapeutics division have been greatly depending of coordinates extent in ionization irradiation fields. On another side, the individual dosimeters have calibrated on the conventional phantoms and in defined reference radiation fields. Instantaneously, the ICRU recommendations have referred one to the phantoms, like a sphere or a slab with tissuequivalent matter or simply Plexiglas (PMMA,polymethyl methacrylate) (1,2). The slab phantom is better for perceive the extent inhomogeneity fields. In as much it's envisaging angular and energy distributions of the photon flux in a longitudinal direction parallel to primary radiation beam, a buildup factor is significant. In a lateral direction, perpendicular to primary beam, a backscatter factor is representative (4,5). In both cases, it's adopting that the photon backscatter is minor from room's wall. Finally, should be establish influence of anisotropy angular distribution of emitted photons from the target X ray tube and contribution of the photon backscatter at the total field profile on the front phantom, on account of the primary radial beam. At the same time, should be see and the good reason for the water phantom utilization.

EXPERIMENTAL MEASUREMENTS

Firstly, all measurement results have concerned at exposure dose rate. The measurements are performed by X ray apparatus with conditions for maximal exposure dose rate at all mean energy photon spectra (6). We have used ionization chambers made by PTW-FREIBURG, volume 0,2 cm^3 , 1 cm^3 and 30 cm^3, uncertainty in measurement ± 2%. Constant potential X ray system MG 320 Philips with metal-ceramic tube MCN321, can generate direct voltage in the range from 30 kV to 320 kV. Other technical data are: anode angle 22^0 , inherent filtration 2,2 mmBe and standard focus 4mm x 4mm. It is existing possibility that conventional mean energy X ray spectra are reproduced by the beam passes through selective filters disc. The beam quality have been accord with the requirements of international standards (3,6). Dimensions water and PMMA (density 1200 kg/cm^3) phantoms were 20 cm x 20 cm x 15 cm. On the occasion of those measurements exposure dose rate at focus-phantom distance 1,5 m , beam diameter was 25 cm. However, measurement results at an edge of the phantom exist insecure for real conclusions because the border effects change backscatter factor for different mediums.

RESULTS AND DISCUSSION

Maximum lateral distribution free-in-air, perpendicular to the beam axis and parallel to the tube axis (PLD), with increasing voltage (60kV-300kV) shifts to the "anode side" (fig.1). For fixed voltage of 300 kV , (fig.2), an alteration lateral profile for different filtration are demonstrated /A30(4 mmAl + 2,2 mmBe, A0(2,2 mmBe)/. The most unfavorable field profile exist for very heavy filtration and high tube

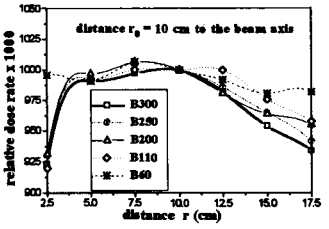

figure 1. PLD free-in-air for different tube voltages

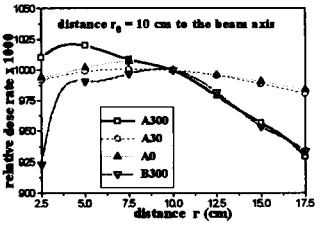

figure 2. PLD free-in-air for different filtrations at 300 kV

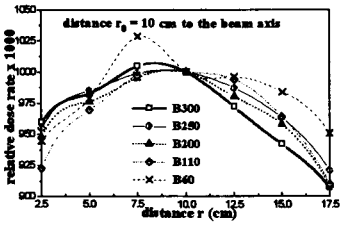

figure 3. PLD in front water phantom for different tube voltages

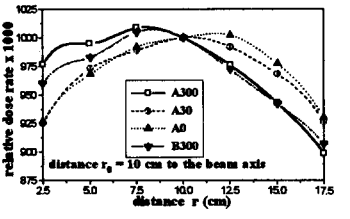

figure 4. PLD in front water phantom for different filtrations at 300 kV

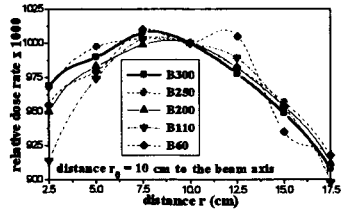

figure 5. PLD in front PMMA phantom for different tube voltages

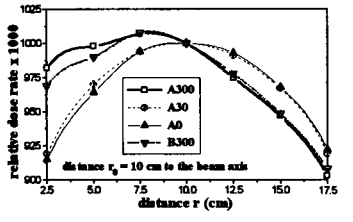

figure 6. PLD in front PMMA phantom for different filtrations at 300 kV

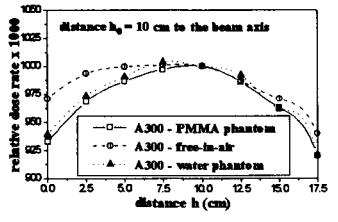

figure 7. NLD in front phantoms and free-in air for HFHV conditions

voltage (HFHV conditions as A300 and B300). Lateral distribution in the direction perpendicular to the tube axis (NLD) is smooth, but asymmetrical to the center field because there are the photon scatter from the bench measuring (fig.7).

Table 1. Radiation qualities of calibration fields and backscatter factors for PMMA and water slab phantom

Code/high voltage (kV)	Additional (#) filtration (mm)				Mean Energy (keV)	Backscatter factor for phantom	
	Al	Cu	Sn	Pb		PMMA	WATER
Narrow spectra							
A60	4.0	0.6	-	-	45	1.32	1.28
A100	4.0	5.0	-	-	82	1.38	1.37
A150	4.0	-	2.5	-	115	1.27	1.24
A200	4.0	2.0	3.0	1.0	155	1.18	1.17
A250	4.0	-	2.0	3.0	210	1.14	1.13
A300	4.0	-	3.0	5.0	250	1.13	1.12
Broad spectra							
B60	4.0	0.3	-	-	43	1.31	1.27
B80	4.0	0.5	-	-	55	1.33	1.29
B110	4.0	2.0	-	-	78	1.40	1.38
B150	4.0	-	1.0	-	105	1.32	1.30
B200	4.0	-	2.0	-	135	1.23	1.22
B250	4.0	-	4.0	-	170	1.18	1.16
B300	4.0	-	6.5	-	200	1.14	1.13

(#) Inherent filtration 2,2 mm Be.

In front of the water and PMMA phantom, on account of divergence beam, values backscatter factor are smaller than for "ideal" parallel beam. It's very significant fact since exposure dose rate profiles are created by local distribution of the photon backscatter. Maximum backscatter yield is for 82 keV (narrow spectra) and for 78 keV (broad spectra) , Table 1. It's finding out the most steep slope at the field profile for B110 (fig. 3 and 5). The symmetry PLD is loosed at extreme irradiation HFHV conditions on account of photon anisotropy emission by X ray apparatus (fig. 4 and 6). Then profile is spreaded evenly to the "anode side", because local diminution photon backscatter is compensated by yield of angular anisotropy emission. There are recommendations (2,3) that permissible inhomogeneity is less than 5%. For all energies in Table 1, at the front slab water phantom those express request have accomplished from 5 to 12,5 cm, until for slab PMMA phantom from 3,75 to 13,75 cm. As for NLD, the symmetry is peculiar and influence of photon scatter from bench measuring is slight. Inhomogeneity within 5% is tolerated from 2,5 to 15 cm above bench (fig.7).

CONCLUSIONS

Prior to employment modern X ray apparatus in the Standard Dosimetry Laboratory should be acquainted the profiles of photon fields for calibrating. At the base well known dimensions of the individual dosimeters (TLD, RPL, film and pMOS solid state) it can project and execute the conventional calibration. One fact is worth putting forward, that the region of high homogeneity (greater than 95%) at front PMMA is greater than for front water phantom.

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STANDARDIZATION OF AN IRRADIATION FIELD USING ^{60}Co AND ^{137}Cs SOURCES (AT RADIOISOTOPE RESERCH CENTER OF OSAKA UNIVERSITY)

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INTRODUCTION

On the occasion of installing the γ-ray irradiation system for animal experiments at the Radioisotope Research Center of Osaka University both lead collimators and shield screen have been supplemented to the system in order to satisfy the Japanese legal regulation that the dose equivalent rate outside the controlled area should be less than 300 μSv/w because the experimental room has not been so designed as to install such an apparatus.

The original use of the system is to study the internal β-ray exposure of a small animal on a dosage of tritium water, which will be eliminated from a body with a biological half-life. Accordingly, the dose rate of internal exposure due to β-rays will change with time, and hence such a situation could be simulated with an external exposure due to γ-rays by changing the dose rate spatially, that is, the distance between the γ-ray source and a sample.

It is, however, anticipated that improvement of the system would bring increase in the scattered γ-rays at an irradiation point and hence it becomes the purpose of the present paper to obtain precise exposures including scattered γ-rays at each irradiation point for animal experiments and also to find an optimum point for standard calibration where no scattered γ-rays are observed. For that purpose the effect of them will be evaluated with both calculations due to the Monte Carlo code for neutron photon transport (MCNP) and experiments due to the ionization chamber calibrated at the National Bureau of Standard in Japan.

EXPERIMENTAL APPARATUS

The γ-ray irradiation system is composed of fixed γ-ray source part and an exposure deck movable along the rail. The latter is so controllable remotely with a computer that the dose rate could continuously be changed from 0.0668 mSv/h to 10.7 mSv/h by adjusting the distance between the source and the measuring point. In the source part are contained three kinds of radioisotopes, ^{137}Cs's of 111GBq and 11.1GBq, and ^{60}Co of 3.7GBq (nominal activity).

The irradiation system has been improved to clear the legal regulation for the radiation safety in the following two items. One is that the exposure deck has been mounted with a lead shield screen (1200w × 1200h × 30t) covered with iron plates (20t) at the back edge side as shown in Fig. 1. The other is that a lead ring collimator system has been supplemented to the source part. It is composed of three kinds of square collimators whose angular apertures are 6.03°, 9.27° and 20.02°, respectively. They can

selected automatically by rotating the ring collimator system as shown in Fig. 2 in response to the distance between the source and the deck, which enables the irradiation field to be confined within the added screen.

The collimator system can smoothly rotate round the RI sources owing to the compact structure though it weighs to be about 280 kg, which assures high stability and reliability in operation.

CALCULATED AND EXPERIMENTAL RESULTS AND DISCUSSIONS

The Monte Carlo code for neutron and photon transport (MCNP) has been used to evaluate the effect of the scattered γ-rays on the energy spectrum of the irradiation field. In Fig. 3 are shown the calculated results of normalized energy spectra including the scattered γ-rays for three different aperture sizes of the collimators. In the calculation a point source of ^{137}Cs has been set on the axis of the cylindrical irradiation part at a height of 1.2 m from the floor, and three right pyramid holes have been excavated through the lead cylinder perpendicularly to the axis as collimators, whose outlet areas are $3\times5cm^2$, $6\times8cm^2$ and $9\times11cm^2$, respectively.

The calculation has been carried out assuming that a detector with a known energy resolution is set at a point of 1m from the source. It is obvious from the figure that adopting a collimator of the smaller aperture would cause decrease in the effect of scattered γ-rays, that is, the total energy peak becomes more remarkable.

In Fig. 4 are shown the calculated results of dependence of the absorbed dose rate on the distance from the source in order to clarify the effect of the added shield screen on the spectra which would be a little modified by scattered γ-rays. In the calculation the absorbed dose rates of 10 different points located between two reference points, distances of 0.5mand 2.2m from the source point, have been obtained with a collimator of $9\times11cm^2$ and with the shield screen at 2.2m from the source.

In the same figure the absorbed doses measured with an ionization chamber are also shown together with the $1/r^2$-dependence curve. The chamber has been calibrated at the Electrotechnical Laboratory(National Bureau of Standard in Japan). It can be seen from the figure that the effect of scattered γ-rays on the absorbed dose is remarkable in the neighborhood of both collimator and screen. The difference between calculated and measured results is mainly attributed to the fact that the collimator size adopted in the calculation was different from the real collimator one. It is, however, found that there exist several points near the middle region where both numerical and experimental results obey the inverse square law. In other words the absorbed dose rates observed there include no scattered component of the irradiated γ-rays.

CONCLUSION

A well-controlled γ-ray field has been established with the improved commercial γ-ray irradiator system. It is ascertained that in some limited region the observed dose rate coincides with the calculated results and also satisfies the inverse square law, which means that the γ-ray field could be a standard one for calibration of both survey meters and all kinds of dosimeters prevailing in Osaka University.

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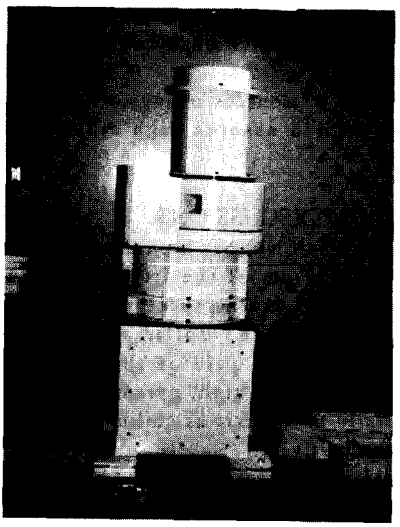

Fig. 1 An irradiator of an irradiation system

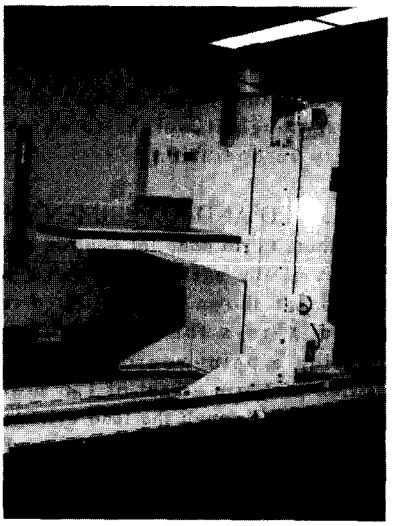

Fig. 2 A mobile exposure deck of an irradiation system

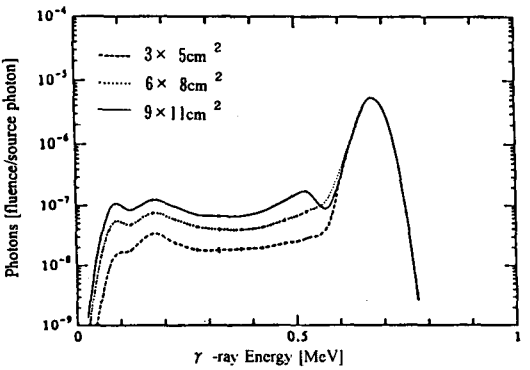

Fig. 3 Caluculated energy spectra from ^{137}Cs source passing through the collimator with exit dimensions of 3×5, 6×8 and 9×11cm ^{2}

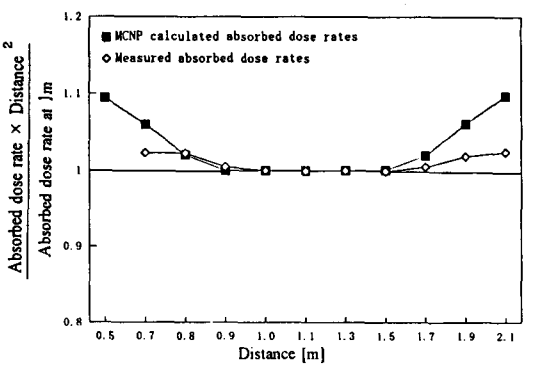

Fig. 4 Dependence of normalized absorbed dose rates on distance

ESTIMATION OF SCATTERED RADIATION SPECTRAL DISTRIBUTION OF EXPOSURE IN GAMMA CALIBRATION FIELDS.

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ABSTRACT

A study was made on the energy spectrum of gamma radiation field used for calibration of radiation survey instruments. Energy spectra were measured with a NaI(Tl) detector and compared with those by Monte Carlo calculations. A good agreement was obtained between the measurement and the calculation. It was found that the use of collimator gives a different energy specification.

INTRODUCTION

A high-quality calibration of photon survey instruments requires information of energy spectrum in the calibration field, because the response of the instruments depends significantly on photon energy. Thus, it is important to estimate the mixture of scattered radiations in the calibration field.

Origins of the scattered radiations are the collimator, the room wall, the room floor, the table of instrument for calibration and the surrounding air. The scattering feature depends on the distance between source and instrument, the size of irradiation room, the type of gamma-ray source and the collimator. Then, the energy spectrum of radiations should be estimated at each calibration point in the field.

In the present work, gamma-ray spectrometry was carried out in some different calibration fields. Monte Carlo calculations were performed using the EGS4 code[1] to verify the experimental results.

METHODS

(1) Gamma-ray spectrometry

The Japan Atomic Energy Research Institute possesses two gamma-ray calibration rooms: a small room and a large one. The small room measures 6.5m x 5m x 4m high and the large one 6.5m x 12m x 6m high. Two calibration sources of ^{137}Cs and ^{60}Co are used extensively in each room. Gamma-ray spectrometry was performed with a 3 inch-diameter spherical NaI (Tl) scintillation detector. The distance between the source and the detector ranged from 1.5m to 2.0m in the small room and from 1.5m to 3.0m in the large one. The source was contained in an irradiation apparatus with a lead collimator and placed on an open field at a height of 1.2m above the floor. A lead shadow shield was placed between the source and the detector to absorb the primary gamma radiation. Four types of irradiation were selected considering the combination of the collimator and the shadow shield as shown in Table 1 ((a), (b), (c) and (d)).

(2) Separation of pulse height spectrum and unfolding

Scattered radiations were divided into three components by their origins. The first is radiations scattered by the collimator (collimator-scattered radiation), the second by the room wall, floor and the table (room-scattered radiation), and the last by air (air-scattered radiation). Table 1 gives these components on the pulse height spectrum for each type of irradiation.

The pulse height spectrum of the room-scattered radiation was obtained by subtracting (c) from (d) and that from the primary radiation by subtracting (d) from (b). Subtracting the primary radiation and (c) from (a) gives the pulse height spectrum from the collimator-scattered radiation. The measured and

divided pulse height spectra were converted to energy spectra by unfolding procedures. The energy distribution of exposure was calculated by multiplying the energy spectra by the corresponding conversion coefficients.

(3) Monte Carlo calculation

Monte Carlo calculations of energy distribution were carried out to verify the experimental results using the EGS4 code[1]. The calculation conditions were modeled as being close to the experimental conditions.

RESULTS AND DISCUSSION

Figure 1 shows the spectral distributions of exposure for the two irradiation types of (a) and (b) in the small room with the ^{137}Cs source. A significant difference can be observed in the energy spectrum between the two irradiation types in the low energy region below 300keV. The result by the Monte Carlo simulation is also depicted for the irradiation type of (a). A good agreement is seen between the measurement and the calculation for energies below 600keV.

Figure 2 shows the collimator-scattered and the room-scattered components in the spectral distributions of exposure in the small room with the ^{137}Cs source. The photon energy of the collimator-scattered radiation is distributed near 662keV and that of the room-scattered is distributed over the energy range from 50keV to 350keV. The former component consists of radiations scattered at the collimator with small angles, and the latter component consists of radiations scattered at the room wall, floor and the table of instrument with large angles. The energy spectrum of each component closely relates with the scattering angle. The two scattered components resulted in a difference in the spectrum between two irradiation types of (a) and (b) in Fig.1.

Table 1 gives the ratio of exposure by scattered radiation to that by primary radiation in the small room for the two irradiation types of (a) and (b) with the two calibration sources. In the results of the ^{137}Cs source, the effect of scattered radiation was larger in (b) than in (a), whereas the inverse results are seen for the ^{60}Co source. It should therefore be noted that the use of collimator can give different characteristics in the effect of scattering between the ^{137}Cs source and the ^{60}Co source.

It is recommended that radiation survey instruments should be calibrated in terms of the ambient dose equivalent, $H^*(10)$, for strongly penetrating radiation[2]. $H^*(10)$ is usually calculated with multiplying the exposure by the conversion coefficient corresponding to the energy of the primary radiation. A more accurate $H^*(10)$ can be calculated with the spectral distribution of exposure in Fig.1. $H^*(10)$ values obtained from Fig. 1 are larger by 1.9% for the irradiation type of (a) and by 3.0% for the irradiation type of (b) than the value conventionally calculated.

CONCLUSION

The feature of energy spectra in the gamma-ray calibration fields was clarified by the measurement and the Monte Carlo simulation. The scattering feature depends significantly on the use of collimator. $H^*(10)$ values obtained in consideration of the energy spectrum are a little larger than the value conventionally calculated.

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Table 1 The combination of collimator and shadow shield for each irradiation type and the gamma -ray components on the pulse height spectrum

Irradiation type	Collimator	Shadow shield	Gamma-ray components on the pulse height spectrum
(a)	used	not used	primary, collimator-scattered and air-scattered radiations
(b)	not used	not used	primary, room-scattered and air-scattered radiations
(c)	used	used	air-scattered radiation
(d)	not used	used	room-scattered and air-scattered radiations

Table 2 The ratio of exposure by the scattered radiation to that by the primary radiation for the two irradiation type in the small room, where the source-to-detector distance is 2.0m.

Nuclide of source	Irradiation type	The ratio of exposure by the scattered radiation to by the primary radiation			
		collimator	room	air	total
^{137}Cs	(a)	0.064	--------	0.021	0.085
	(b)	--------	0.112	0.021	0.133
^{60}Co	(a)	0.094	--------	0.016	0.110
	(b)	--------	0.066	0.016	0.082

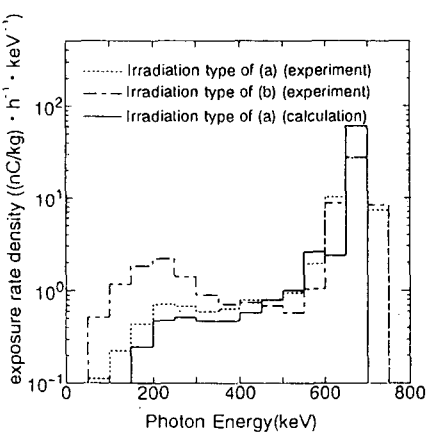

Figure 1 The spectral distribution of exposure for the two irradiation types of (a) and (b) in the small room with ^{137}Cs source. The source-to-detector distance is 2.0m.

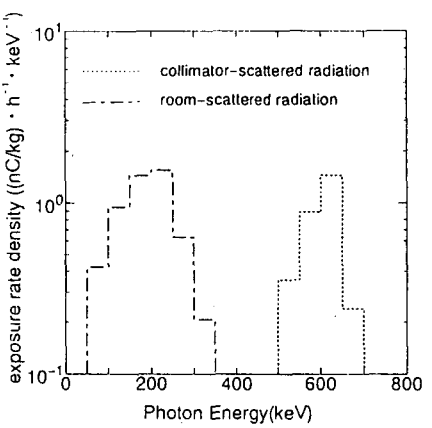

Figure 2 The spectral distribution of exposure of scattered radiation in the small room with ^{137}Cs source. The source-to-detector distance is 2.0m.

GAMMA RADIATION STREAMING THROUGH TRIPLE BENT LABYRINTH

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INTRODUCTION

Radiation streaming is one of the difficult problem in the radiation shielding. For photons, several works on the attenuation of photon dose through the streaming pass exists and the discussions were done by mainly considering geometrical effects. As a recent movement, connecting to the radioactive waste handling facilities and the reprocessing plants, the precise handling of gamma streaming has been required(1-3). In this paper, as a purpose of providing a benchmark data, the experiment and calculation for photon streaming of three bent labyrinth was performed.

EXPERIMENT

The experiment was done in the Cobalt-60 Irradiation Facility Room of the University of Tokyo(4). The irradiation room and labyrinth structure are shown in Fig.1. Gamma dose were measured by ionization chambers, which were used as a monitor and an absolute detector. Thermoluminescent dose meters (TLD) were also used as a dosimeters for maping of spatial gamma dose distribution. For several typical positions, a photon spectrometer of NaI(Tl) scintillator was used to get gamma spectra by doing spectral unfolding.

CALCULATION

The calculational method was a Monte Carlo method and MCNP4a code was applied. The photon cross section was Hubbell's one(5). The variation reduction techniques were used based on the discussion of scattering importance and the determinations of scattering region and angular selection. The used calculational geometry is shown in Fig.2. The detailed structure of concrete and iron lining of room wall etc. were included in the calculation.

RESULTS AND DISCUSSIONS

Some of the measured and calculated results are shown in Fig.3 and 4. The comparison between experiment and calculation was done. On the spatial dose attenuation along the axis of labyrinth, a completely good result of C/E was obtained in Fig.3, in spite of seven decades decrease of gamma dose. From the detailed discussions of dose distribution in the labyrinth, the followings were found. The first scattering wall has an important roll of streaming estimation, which is, for instance, a kind of shining source to the labyrinth.

For photon spectra, typical results are shown in Fig.4. In this figure, Co-60 high gamma was suddenly degraded in energy, and Compton and multiple scattered gamma made a soft spectrum. This procedure was excellently reproduced by the calculation. The uniform spectral field was fabricated after second bent. This means the conversion factors of the measured TLD to radiation dose should be differently used, depending on the position, for the reasons of sudden spectral change.

CONCLUSIONS

Co-60 photon streaming experiment through triple bent labyrinth was done and dose attenuation along the central axis and gamma spectral change were obtained. The analysis of the Monte Carlo calculation was followed and the important scattering region and angular section was clarified. This good estimation of C/E contributed to the establishment of the photon streaming analysis.

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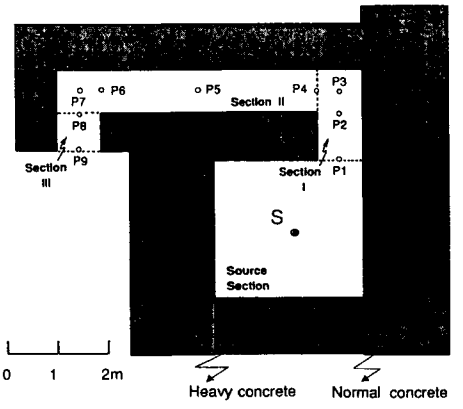

Fig.1. Structure of labyrinth in cobalt-60 irradiation facility. (S: Cobalt-60 Source; P1,....P9: Measuring point identification number).

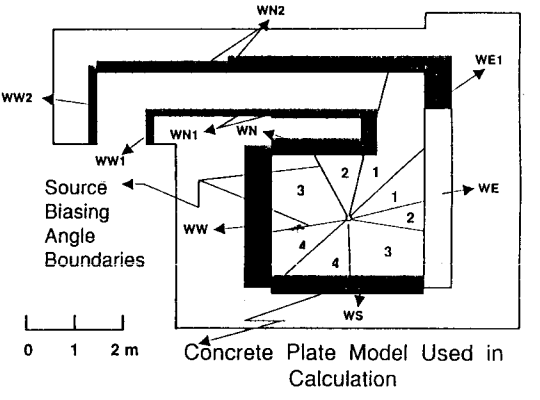

Fig.2 Calculational models for gamma radiation streaming through the labyrinth.

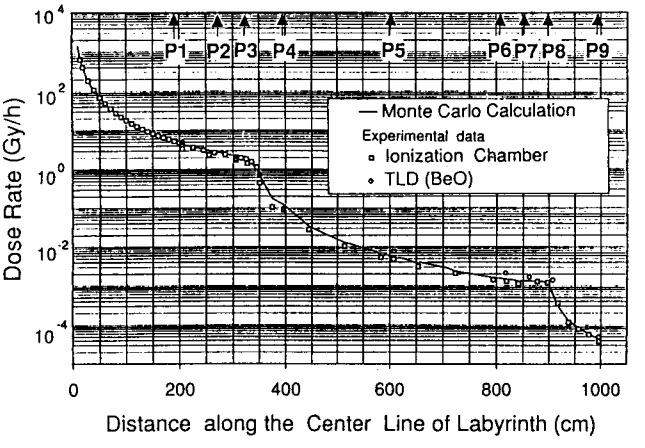

Fig.3. Dose rate spatial attenuation from cobalt-60 source center in the irradiation room.

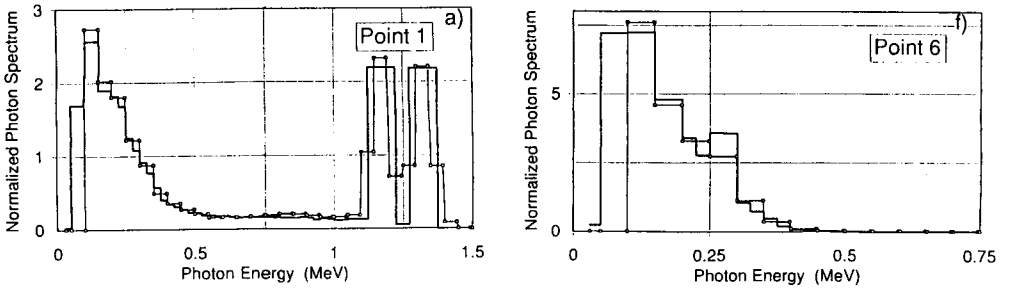

Fig.4. Comparison of measured and calculated photon spectra along the labyrinth.

ON DETERMINATION OF THE OPTIMAL SAMPLE SIZE IN RADIATION CONTROL

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INTRODUCTION

Among various statistical models used for fitting occupational and environmental radiation data the lognormal distribution is the most common one. Its application became especially wide after this model has been adopted by International Commission on Radiological Protection (1, 2). Natural and occupational gamma- radon- and ultraviolet exposure, rate of global radioactive fallout, radiation contamination of the atmosphere are very often distributed lognormally (3,4). Typical problems arising when using the lognormal distribution model for purposes of radiation control are: how to estimate the average level of radiation with a given accuracy (the problem of estimation of unknown true mean of the distribution) and how to determine the minimum number of measurements by which the calculated average contamination differs from the true mean value no more than a given value of error (the problem of the number of sampling which is enough for estimating the unknown true mean with a given accuracy). The method universally practised of estimating the mean value consist of finding the arithmetic mean of the set of N measured values. Although the arithmetic mean as a statistical estimator is unbiased, it not always provides minimum variance of the estimation. Naturally arithmetic mean often differs from the true mean if the statistical distribution differs from Gaussian normal distribution. Therefore, it is necessary to know the distribution of sample mean from a lognormal population, but unfortunately this distribution is not known. All that can be said is that for a large number of samples it has an asymptotically normal distribution behaviour (5). Early, an attempt was made to obtain this distribution using some simplifications (6, 7), but uncertainty of the results caused by these simplifications, is too difficult to assess. In the present paper the sampling distribution of the lognormal population was found without any simplification.

GENERAL CONSIDERATIONS

As shown by Finney (8) the best estimation x_{EST} of the true mean value $\bar{x}$ (unbiased and given minimum variance of the differences from the true mean) is given for lognormal distribution by the formula:

$$x_{EST} = \exp(\mu)\,\psi_N(S^2) \qquad (1)$$

where μ and S^2 are the sample mean and variance, respectively, and $\psi_N(S^2)$ is a correction function depending on the sample variance S^2 and on the sampling number N. In (9) was shown that the variable L, defined as the ratio of sample mean calculated using expression (1), to its true mean value

$$L = x_{EST} / \bar{x} \qquad (2)$$

can be considered as a function of two random independent variables ξ and U, where the first is distributed normally and the second has a χ^2 distribution. Taking this into account we can express the distribution of L by multiplication of the distributions of ξ and U. Therefore the probability of fulfilling the condition $L \leq L'$ can be expressed by the integral of this product in that region of ξ and U, where this condition is fulfilled:

$$P(L \le L') \sim \iint_{L \le L'} \varphi(\xi)\, \chi^2(U)\, d\xi\, dU \qquad (3)$$

CALCULATION OF THE DISTRIBUTION

In the paper (9) the distribution of L was calculated by method of statistical testing (Monte Carlo method). But this technique is not an appropriate method if it necessary to get the results with a high precision. That is why in this investigation analytical and numerical methods were used. It gave the possibility to improves considerably the accuracy of calculations.

We have determined the distribution of L by numerical calculation of the integral (3) using Gauss' method with 64 nodes and their corresponding weight coefficients. For each combination of the sample size N (varied from 2 to 200) and sample standard deviation S (varied from 0.1 to 2.0) values for the cumulative probability, corresponding to different relative deviation D, between the true mean value and its best estimation, given by

$$D = (x_{EST} - \bar{x}) / \bar{x} \qquad (4)$$

were calculated. The D values for which calculations were made varied from 0.1 to 1.0 with a step equal to 0.1. The probability F_- and F_+ for negative and positive deviations of x_{EST} from the true mean $\bar{x}$ was calculated separately. Thus, the obtained data give a possibility to determine the probability of underestimation as well as overestimation by using x_{EST} as estimator of the true mean $\bar{x}$. Also their sum $F = F_- + F_+$ were calculated.

RESULTS AND THEIR APPLICATION

The distribution of the sampling mean were calculated and then used for the determination of the number of the sample size N that is enough for the estimation of the unknown true mean with a given accuracy. A part of the results obtained are presented in Table 1. The required number of sampling N are given in the Table as a function of the empirical variance S^2 and the confidence level P. For estimation of N it is necessary first to calculate μ and S^2. Then for a given value of error D and for a desirable confidence level P in Table 1 the number of N can be found.

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Table 1. The number of sample size N that is enough for the estimation of the mean with a given accuracy

Empirical standard deviation, S	Confidence level, P	Relative deviation, D				
		0.1	0.2	0.3	0.4	0.5
0.1	0.68	3	2	2	2	2
	0.90	5	3	3	3	3
	0.95	8	4	4	4	3
0.2	0.68	6	3	2	2	2
	0.90	14	5	4	3	3
	0.95	20	8	5	4	4
0.3	0.68	12	4	3	3	2
	0.90	28	10	5	4	4
	0.95	40	12	8	5	5
0.4	0.68	20	6	4	3	3
	0.90	50	14	8	5	4
	0.95	70	20	10	8	6
0.5	0.68	30	10	5	4	3
	0.90	80	22	12	8	5
	0.95	120	30	16	10	8
0.6	0.68	44	12	6	4	3
	0.90	120	32	16	10	8
	0.95	170	44	22	14	10
0.7	0.68	65	18	8	5	4
	0.90	170	44	20	12	10
	0.95	> 200	65	28	18	12
0.8	0.68	90	22	12	8	5
	0.90	> 200	60	28	16	12
	0.95	> 200	90	38	22	16
0.9	0.68	120	30	14	8	6
	0.90	> 200	80	36	22	14
	0.95	> 200	120	50	30	20
1.0	0.68	120	38	18	10	8
	0.90	> 200	110	46	26	18
	0.95	> 200	150	65	38	24

NEW DOSIMETRIC INSTRUMENTS BASED ON A RECOMBINATION CHAMBER

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INTRODUCTION

Three prototypes of new dosimetric devices were recently developed for different purposes of radiation protection in mixed radiation fields. The REM-2 high pressure tissue equivalent ionization chamber serves as the detector in all the systems described. The operation of the chamber under conditions of initial recombination of ions enables evaluation of radiation quality in radiation fields of unknown composition.

RECOMBINATION CHAMBER

The REM-2 chamber contains 25 tissue-equivalent electrodes, which form 12 parallel sections (1). The electrodes are 12 cm in diameter and 3 mm thick. The distance between electrodes is 7 mm. Total mass is equal to 6.5 kg. The effective wall thickness of the chamber is equivalent to about 1.8 cm of tissue and the gas cavity volume is of about 2000 cm^3. The chamber is usually filled, up to about 1 MPa, with a gas mixture consisting of methane and 5% of nitrogen. The central rod connects 12 collecting electrodes. The polarizing electrodes are connected alternately to one of two side rods and thus form two sets of electrodes. The chamber can be used in different operating modes. Two of them are important for the devices considered:

1) The *saturation* mode, when the voltage ensuring near saturation conditions (1200 V) is applied to both sets of the polarizing electrodes. This mode is used for determination of the ambient absorbed dose, $D^*(10)$ of mixed radiation. The sensitivity of the chamber in this mode is equal to 100 $fA/\mu Gy \cdot h^{-1}$.

2) The *differential* mode, when the saturation voltage is applied to one set of the polarizing electrodes and the recombination voltage of opposite polarity is applied to the another one. This mode ensures direct proportionality of the charge collected on the measuring electrodes to the ambient dose equivalent of mixed radiation of any composition. The sensitivity of the chamber operated in this mode equals to 2 $fA/\mu Sv \cdot h^{-1}$.

MICROCOMPUTER CONTROLLED DOSE EQUIVALENT RATE METER

It was shown earlier (1-3) that the ionization chamber of REM-2 type may serve as a good detector of $H^*(10)$, with weak dependence on neutron energy (see Fig. 1).

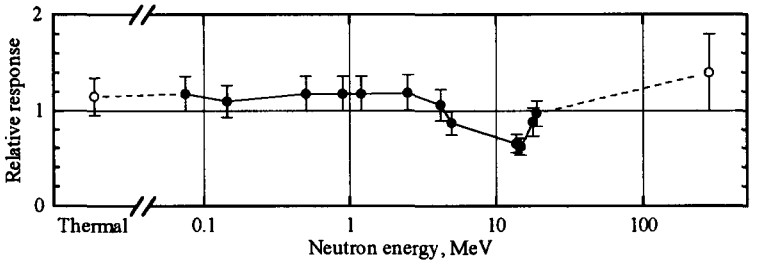

Figure 1. Relative response of the REM-2 recombination chamber to H*(10) versus neutron energy (1)

Now a laboratory set-up was designed (1), which can be considered as a prototype of a microcomputer controlled dose equivalent rate meter, based on a recombination principle and intended for routine use in radiation protection. The device consists of: a recombination chamber REM-2, a monitoring chamber, and an electronic circuit connected to a personal computer and controlled by special software for the measurement control, data collecting and on line processing.

The basic circuit of the device is shown in Fig. 2. Additionally to the main measuring channel with the REM-2 chamber it contains also the monitoring channel with the second chamber M. The monitoring is necessary if the radiation field is unsteady i.e. when the mean dose rate changes more than 1% during the time of measurements, which is of order of minutes.

The dose equivalent is proportional to the amount of recombination, so it can be determined by comparison the ionization currents collected at two different collecting voltages applied to the chamber. In our device the sequential application of the voltages to the chamber is controlled by the computer. The high-stability voltage unit U_{CTRL} is used as the voltage source. Its output voltage is exactly 200 times higher than the control voltage given by the analog output of the DAC computer card.

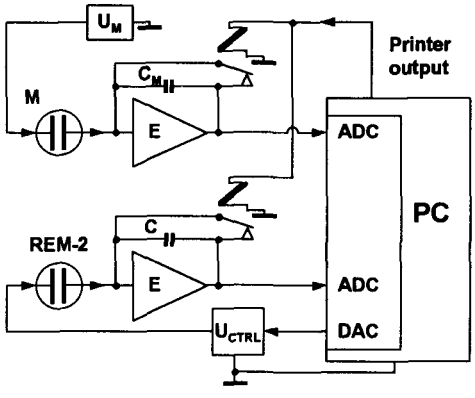

Figure 2. Principle circuit of the ambient dose equivalent meter (1).
REM-2 - recombination chamber, U_{CTRL} - externally controlled high voltage supply unit, E - electrometer, PC - personal computer with ADC/DAC card, M - monitor chamber, U_M - steady voltage unit.

Ionization current is measured by the electrometer E operated in an integrating mode. Ions collected on the electrodes of the chamber charge the integrating capacitor C during an appropriate time interval. The output voltages are measured using the analog input of the computer card (ADC). The 5 V signal from the printer output of the computer is used for discharging of the capacitor C and for shortening of the electrometer at the time of the voltage change-over.

The basic measuring range of the dose rate is from $\dot{D}^*(10) = 25$ μGy/h up to 250 μGy/h, and the whole range - from 1 μGy/h to 0.25 Gy/h, which corresponds to the values of $\dot{H}^*(10)$ from 1 μSv/h up to about 1 Sv/h. The accuracy of measured values of $\dot{D}^*(10)$ and of $\dot{H}^*(10)$ depends primarily on energy characteristic of recombination chamber and on dose rate. Usually, it is better than 10% for $\dot{D}^*(10)$ and better than 20% for $\dot{H}^*(10)$, in the basic range of dose rates. Accuracy of $\dot{H}^*(10)$ in the whole range of dose rates and energy covered by the device is within a factor of two.

SYSTEM FOR IN-FLIGHT AND LOW-LEVEL DOSIMETRY

The minimum detection level, stated in the technical manual of the REM-2 chamber is 10 μGy/h for the saturation mode and 500 μSv/h for the differential mode. These values do not satisfy the requirements of in-flight dosimetry. The problem can be solved by use of the self-contained measuring system (4), shortly described below (see Fig. 3).

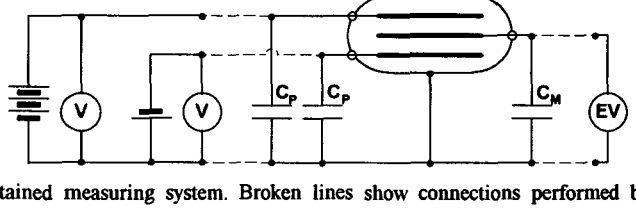

Figure 3. Self-contained measuring system. Broken lines show connections performed before and after irradiation.

The system consists of recombination chamber with three capacitors. The distortion charge is compensated by use of a special procedure for charging the supplying capacitors and for reading the voltage on the measuring capacitors. The signal measured by the system is proportional to $H^*(10)$.

Before the flight the capacitors C_P should be charged to the voltage U_{P0} at known temperature T_0. It was shown that the charge collected on the measuring capacity C_M does not depend on course of changes of the polarizing voltage and temperature and is proportional to the absorbed dose if the temperature of the chamber at the time of read out of the voltage on the measuring capacity is equal to T_0 and the voltage between the chamber electrodes is equal to U_{P0}. Therefore, after the flight the irradiated measuring system should be placed in laboratory conditions at the temperature T_0 for sufficiently long time. Then the polarizing capacity should be connected to a source of voltage, exactly equal to the initial voltage U_{P0}.

Provisional results indicate that our system gives possibility to measure the $H^*(10)$ of order of some μSv with uncertainty ca. 25% in any field of penetration radiation, with an integration time up to some days. Investigations of the performance characteristics of the system are under progress.

ANALOG METER OF QUALITY FACTOR

In this device (5), the modified recombination chamber, denoted REM-2Q, serves as a detector. The modification lies in the change of the interelectrode spacing such that two parts of the differential recombination chamber are not equal: the ratio of their volumes equals to about 0.9. The device enables direct determination of the radiation quality factor (QF) in mixed radiation fields.

The operational principle of the QF-meter is shown in Fig. 4.

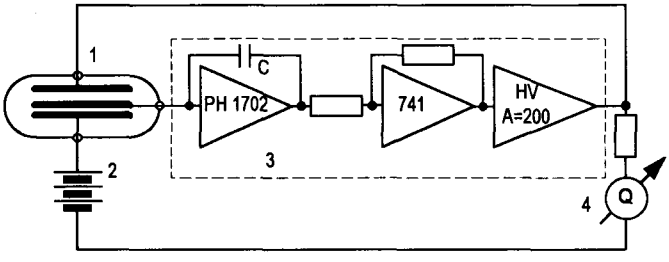

Figure 4. Functional circuit of the QF-meter: 1 - recombination chamber; 2 - high voltage source; 3 - high voltage electrometrical amplifier; 4 - voltmeter calibrated in units of quality factor.

When the chamber is placed in a radiation field, almost all the positive ions created in the smaller volume of the chamber, are collected to the middle electrode and charge the capacity C. This causes an increase of the negative voltage at the output of the high voltage amplifier. This voltage is applied to the electrode supplying the larger volume of the chamber. Therefore some negative ions from the larger volume can also reach the collecting electrode, reducing the effective current charging capacity C. An equilibrium voltage of the upper electrode is established when the resulting ionization current through the collecting electrode is equal to zero, i.e. when the ratio of the ion collection efficiencies in the larger and smaller parts of the chamber is equal to the volume ratio of the chamber. The recombination chamber works under conditions of local recombination of ions, so the value of the equilibrium voltage depends on LET and is approximately proportional to the radiation quality factor, as defined in ICRP Report 21.

ACKNOWLEDGEMENTS

The work was partially supported by State Committee for Scientific Research KBN grant No 4 S404 010 05.

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RESULTS ON $^{90}Sr + ^{90}Y$ APPLICATORS DOSIMETRY USING EXTRAPOLATION CHAMBERS

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ABSTRACT

The characteristics of four extrapolation chambers designed and constructed at the Calibration Laboratory of São Paulo were studied in this work. The relationship between the collecting electrode size and the entrance window density with the estimated absorbed dose-rate was investigated. Comparison with the Amersham calibration certificate indicate surface dose-rates agreement within 5%.

INTRODUCTION

^{90}Sr applicators have been successfully used for the treatment of superficial lesions. Different calibration techniques such as TLD, radiochromic dye films, fixed volume ionization chambers and scintillators have been reported (1,2). Nevertheless, significant discrepancies in the calibration of this sources are still observed (3).

At the Calibration Laboratory of São Paulo, four extrapolation chambers were projected and constructed with the aim of calibrate dermatological applicators. In this work, the influence of several parameters including the choice of the collecting electrode diameter and the entrance window density in the measurements of absorbed dose-rate was investigated. The absorbed dose rate of a $^{90}Sr + ^{90}Y$ beta radiation applicator was obtained using the different chambers, and the results were compared.

MATERIAL AND METHODS

The developed extrapolation chambers have a collecting electrode and a guard-ring of graphite. The chamber bodies are of Lucite and the entrance window of aluminized Mylar. Lucite was also used as insulating material between the electrode collector and the guard-ring . The main characteristics of the chambers are presented in Table1.

Table 1 Characteristics of the IPEN Extrapolation Chambers

Chamber	Collecting Electrode Diameter (mm)	Entrance Window Superficial Density ($mg.cm^{-2}$)	Effective Area (mm)
C1	10	0.84	78.5
C2	10	6.42	78.5
C3	3	6.42	8.30
C4	3	0.84	8.30

A Keithley 617 electrometer was used as measurement assembly. The applicator of the present study was from Amersham, Model SIQ, containing 1480MBq (1968) of $^{90}Sr + ^{90}Y$. The absorbed dose-rate in tissue ($7mg.cm^{-2}$) quoted by the manufacturer on 8.11.68 and corrected for the decay time to the measurement day and for the recent values of $\overline{W}/e$ and S_{air}^{tissue} is 31.01 $mGy.s^{-1}$.

RESULTS

a. Transmission Factors

The transmission factors were obtained covering the chambers with the polyethylene therephthalate (Hostaphan) foils and Plexigles plates with different thicknesses. The measurements were realized with the $^{90}Sr + ^{90}Y$ applicator placed as near as possible of the chambers and also at a distance of 1,0cm. The maximum relative standard deviation in these measurements was 1%.

The transmission factors determined for typical values of the tissue equivalent material with the chamber C4 are presented in Table 2. Similar results were verified with the other chambers; a comparison between the obtained transmission factors showed a difference lower than 1%. The results of Table 2 also indicate that, for a null source-detector distance (a=0), the effect of adding 7mg.cm^{-2} of plastic to the extrapolation chambers has been found to lower the measured dose rate by about 5%.

Table 2. Transmission factors for beta radiation $^{90}Sr + ^{90}Y$
a: Chamber C4-source distance

Tissue Thickness mm	Superficial Density mg.cm^{-2}	a = 0	a = 1cm
0	0	1.000	1.000
0,02	2	0.984	1.002
0,04	4	0.972	1.008
0,05	5	0.968	1.010
0,07	7	0.956	1.012
0,10	10	0.936	1.016
0,20	20	0.888	1.020
0,50	50	0.748	0.992
1,00	100	0.554	0.816

b. Absorbed Dose-Rates to Tissue

The absorbed dose rates to tissue were determined from current measurements at air gaps from 0.10 to 0.30mm. The measurements were realized with the applicator positioned as near as possible from the chambers. The extrapolation curves were obtained measuring the ionization current for both potential polarities applied to the chambers and plotting the average of this values as a function of the chamber depth. A linear function was fitted to the current-versus-air-gap data and the slope was used to determine the average surface-absorbed-dose-rate over the central area of the source. A constant potential gradient of 100V/mm was employed for all air gaps.

The absorbed dose rate to tissue in Gy/s is given by (3):

$$\dot{D}_z = \frac{(\overline{W}/e).S_{air}^{tissue}}{\rho_0 . A}\left(\frac{\Delta I_c}{\Delta d}\right) \qquad (1)$$

where $\dot{D}_z$ is the absorbed-dose rate to tissue at depth z, $\overline{W}/e$ is the average energy required to produce an ion pair in dry air (33.97 J/C), S_{air}^{tissue} is the ratio of the average mass stopping power of tissue to air (1.12), ρ_o is the density of dry air at reference conditions of 22 °C and 101.3kPa, A is the area of the collecting electrode and $\left(\frac{\Delta I_c}{\Delta d}\right)$ is the fitted slope of the corrected current versus air gap function. The transmission factors were applied in the determination of the absorbed-dose rate at 7mg.cm^{-2}.

The uncertainties in this procedure include the measured $(\frac{\Delta I_c}{\Delta d})$ ratio as well as the uncertainties in the chosen values for the average energy per ion pair, the stopping power ratios and other correction factors. The overall uncertainties of the absorbed dose rates were estimated to be approximately 13%. Table 3 presents the obtained results of the four chambers.

Table 3. Absorbed dose rate to tissue at 7mg.cm^{-2}

Chamber	$\dot{D}_m$ (mGy.s^{-1})	Δ (%)
C1	30.91	0.3
C2	30.85	0.5
C3	30.18	2.7
C4	30.98	0.1

where:

$\dot{D}_m$: absorbed dose rate to tissue at 7mg.cm^{-2} obtained experimentally;

Δ: percentual difference between $\dot{D}_m$ and $\dot{D}_0$ for the null distance.

Comparing the absorbed dose rates to tissue, determined using the four chambers with those of the source calibration certificate, corrected for the more recent values of $\overline{W}/e$ and S_{air}^{tissue}, a maximum percentual difference of approximately 2.7 % was verified.

CONCLUSION

The results show that the dose-rate determination using the developed extrapolation chambers are in agreement with the Amersham evaluations (corrected for the recent constants) with the overall uncertainties. In this study, the influence of the collecting electrode size of the chambers was negligible. However, in this case all chambers used an electrode smaller than the source size. The comparison between the results obtained with the four chambers indicate a percentual difference lower than 3%, showing a good agreement. From the experiments, it was also verified that the dose rate for tissue irradiation falls off by approximately 5% between the surface and a depth of 7mg.cm^{-2}.

ACKNOWLEDGMENTS

The authors acknowledge the valuable technical assistance of Mr. Marcos Xavier and the financial support of Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPQ).

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PAPER TITLE

Dose calculations by electron transport Monte Carlo simulation using ELSIM computer codes package

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MAJOR SCIENTIFIC TOPIC NUMBER 3 ..... (see page 7)

ABSTRACT (See instructions overleaf)

For the purpose of solving electron transport problems by Monte Carlo simulation, a computer codes package was develloped (ELSIM). ELSIM is based on condensed history Monte Carlo algorithm. In order to get reliable results over a wide range of electrons energies, a variety of electron transport techniques was implemented like: Molière and Goudsmit-Saunderson multiscatter angle distributions, Blunk-Leisegang multiscatter energy distribution, sampling of electron-electron and bremsstrahlung individual interactions. Also are included path-length and lateral displacement corrections algorithms and a module for computing collision, radiative and total restricted stopping powers and ranges of electrons. Each component was tested and rezults obtained with different techniques was compared. Also is presented the solving of some typical electron transport problems which apear in radioprotection and dosimetry.

A MICRODOSIMETRIC INTERPRETATION OF LET-DEPENDENT RADIATION EFFECTS *

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ABSTRACT

LET dependent effects in LiF TL-dosemeters seem to be similar to effects observed in biological systems. Analyses of the glowcurves of LiF- TLDs reveal a significant dependence of the peak hight ratios on the linear energy transfer (LET) of absorbed radiation. Several types of thermoluminescence dosemeters (TLDs) were calibrated in beams of heavy charged particles with various energies. In order to simulate the high energy densities around high LET particle tracks the dose response of the TLDs was recorded up to 40 kGy using Sr-90/Y-90 beta irradiation. A track structure model was applied in order to calculate the radial energy density gradients for the heavy charged particles. The energy density distribution around particle tracks was foulded with the dose characteristics of the dominant peak (200°C) as well as with the dose characteristics of the high temperature peaks (230-350°C). The applied model describes the decreasing TL-efficiency as well as the increasing peak ratios with increasing LET of absorbed radiation. The peak height ratios of LiF-dosemeter material is determined by the microdosimetric energy density distribution and may be used for determination of average LET respectively the mean specific energy density in mixed radiation fields. The biological relevance of these effects will be discussed.

INTRODUCTION

In former investigations a method of LET determination with TLDs was developed. This method was approached to measure the average LET in complex mixed radiation fields like in spacecrafts and aircrafts. The method utilizes the change of the peak height ratios in glowcurves from LiF thermoluminescent dosemeters. Irradiations of different types of TLDs were carried out in fields of different high energetic heavy ions as well as with alpha particles, neutrons, beta and gamma radiation and in mixed fields. It was found, that the ratio of the high temperature peaks (peak 6, 7) to peak 5 increases steady with increasing LET of absorbed radiation. This effect was calibrated and used for determination of the average LET of space radiation (1). It is common known, that the LET is not the optimal parameter to describe the biological effectiveness of radiation. For this reason in microdosimetry the LET is replaced by statistical parameters like the lineal energy or the distributions of energy density or the specific energy. Of course, from the point of view of microdosimetry, „LET"-dependent effects in TLDs are also a result of the microdosimetric distribution of energy density. For a better undestanding of these effects and in order to use TLDs as micro-dosemeters we started to investigate the microdosimetric properties of TLDs. For this purpose we applied a model of track structure theory.

SCIENTIFIC METHODS

Track Structure Theory was developed in order to understand the behaviour of different detectors as well as the sensitivity of biological material in connection with the absorption of radiation with high LET (2). This theory is based on the assumption, that the macroscopic parameters of detectors (efficiency, dose response etc.) or the damage in biological material is determined by the microdosimetric distribution of energy density within the tracks of heavy charged particles (HCP). In contradiction to the concept of LET this theory takes the radial distribution of energy density around the track of a charged particle into account. Several authors have applied this theory on thermoluminescent dosemeters in order to calculate their efficiency against HCP (3).

Zimmerman suggested a model to calculate the alpha efficiency of TL-materials (4). He foulded the dose dependent efficiency, measured after absorption of sparely ionizing beta radiation of several TL-materials, with the calculated energy density distribution around the tracks of alpha particles and integrated over the radius of the track. This integral gives the TL-response per unit track length. This has to be compared with the TL-response for the equal absorbed energy caused by a low dose of beta radiation. The ratio of these two responses is the relative efficiency of the material against alpha radiation. With this model Zimmerman obtained good theoretical results for the alpha efficiency of various TL-phosphors. We followed the suggestions from Zimmerman but used a more sophisticated track structure model introduced by Zhang (5). The energy density around particle tracks, calculated using this algorithm, was numerically integrated under consideration of sensitive site diameters to get the average grain dose suggested by Hansen (6). A computer code was developed for the calculation of the average grain dose distribution and foulding with the measured, dose-dependent beta efficiencies of TL-crystals, for various projectile ions. With this computer code it is possible, to do calculations for various diameters of the sensitive sites.

EXPERIMENTAL

Irradiation of various types of dosemeters with high energetic heavy ions (HCPs) were carried out at the accelerator center in Dubna. For irradiation beams of 5.12 GeV oxygen ions, 5.95 GeV and 5.23 GeV fluorine-19 ions were used.

* Work supported by the Austrian Society for Aerospace Medicine (ASM)

The LET-spectrum in the fluorine ion beams was changed by using different absorber thicknesses varying from 0 to 9.53 g/cm^2 and the LET of fluorine ions was shifted from 30 to 80 keV/µm in tissue. The LET spectrum during irradiation was recorded by a semiconducter detector. To get additional calibration points in a lower LET region, carbon-12 ions with different energies (150, 300 and 600 MeV/nuc) were used with a LET of 9.1, 12.5 and 19.4 keV/µm. The results of irradiation with different LET-radiation are shown in fig. 1. The dosemeters were irradiated with low doses in the linear range far below 10 Gy. To line out the differences in the high temperature range, the glow curves were normalized to peak 5 at 200 °C. In order to measure the dose response with sparely ionizing radiation the dosemeters were irradiated with a Sr-90/Y-90 beta source. The dose characteristic of peak 5, 6 and 7 were recorded up to 40 kGy. Some glowcurves after high dose beta irradiation are shown in fig.2. Glowcurves were normalized to peak 5.

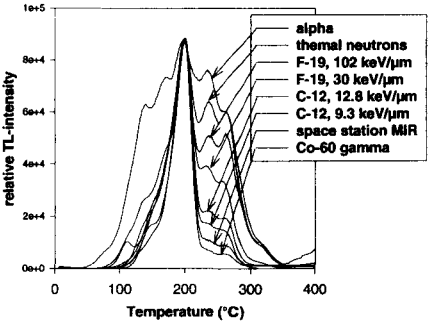

Fig.1: Glowcurves from TLD-600 after absorption of radiation with different LET (low dose)

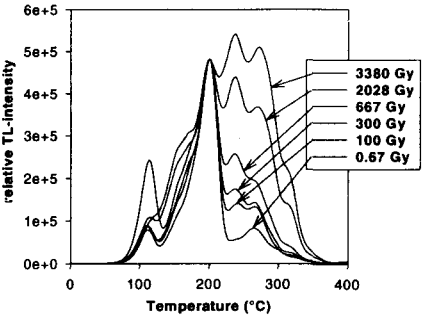

Fig.2: Glowkurves from TLD-600 after absorption of high dose beta radiation

RESULTS

A comparison of fig.2 with fig.1 lines out how similar the glow curves look in the high temperature region, measured after absorption of low dose high LET radiation and high dose beta irradiation respectively. This fact leads to the assumption, that the microdosimetric distribution of energy density determines the shape of the glowcurve and it is possible to use this information in order to obtain important microdosimetric parameters.

Therfore we started to check the assumption, that only the energy density distribution determines the efficiency of the peaks in the glow curve, with track structure calculations. For this reason the dose dependent efficiency of the crystals was fitted in order to get a function for calculation. The plot of the measured relative efficiencies of the peaks 5, 6 and 7 of TLD-600 against dose (fig.3) shows, that after reaching the maximum of supralinearity at about 300 Gy the efficiency decreases rapidly.

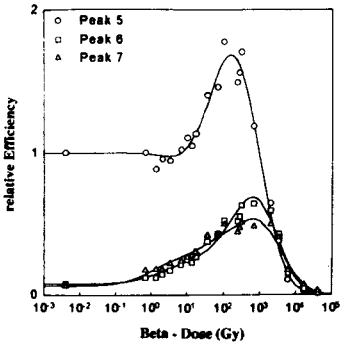

Fig.3: Measured dose dependent efficiencies for peak 5, 6, and 7 after beta irradiation. The least square fit was used for calculation.

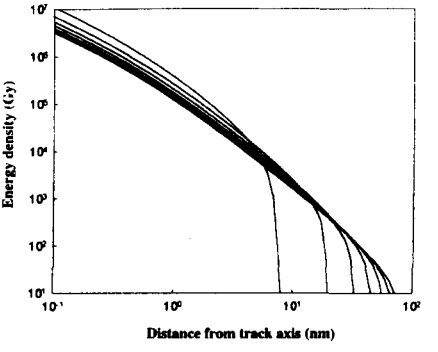

Fig 4: Calculated radial energy density in an alpha track. The distribution of energy density was calculated in 7 layers from the surface to the end of the range.

The efficiencies of the peaks 6 and 7 increase up to 1000 Gy and start to decrease above about 3000 Gy. The efficiency of peak 6 decreases more rapidly than the efficiency of peak 7. Because the peaks are strongly overlapping and deconvolution is not very reproducible, peak maximum was choosen as parameter. The next step was to calculate the energy density around particle tracks. We started to test our model with Am-241 alpha irradiated chips. Because the LET of 4.5 MeV alpha particles increases within the range, calculation was carried out for 7 layers within the range. Fig. 4 shows the change of calculated track structure. The track diameter is decreasing from about 120 nm in the first

layer to 14 nm in layer 7. The energy density in the center of the track is increasing towards Bragg-peak. Grain dose distribution was calculated in each layer and foulded with the beta efficiency. Target radius a_0 was varied in the calculation from 5 to 30 nm. The best correspondence between observed and calculated values was found for target radii of 15 nm (peak 5 and peak 6) and 11 nm for peak 7. Table 1 shows the excellent agreement between measured and calculated peak efficiencies.

Table 1: Measured and calculated peak efficiencies of TLD-600 after alpha irradiation:

Dosemeter	Peak number	Measured efficiency	Taget radius a_0 (nm)	Calculated efficiency
TLD-600	5	0.125	15	0.124
TLD-600	6	0.093	15	0.094
TLD-600	7	0.070	11	0.070

The target radii (radius of the sensitive site) obtained from this experiment were used for further calculations. Table 2 shows the relative peak efficiencies measured after irradiation with different high energetic heavy charged particles (HCP) in comparison with the results from track structure calculations.

Table 2: Measured and calculated relative peak ratios after HCP irradiation:

Irradiation			Peak - ratios, TLD-600			
Particle	Energy	LET	Peak 6/Peak5		Peak 7/Peak5	
	(MeV)	(keV/µm tiss)	measured	calculated	measured	calculated
C-12	1800	19.4	0.42	0.25	0.42	0.21
C-12	3600	12.5	0.25	0.21	0.22	0.18
C-12	7200	9.1	0.19	0.18	0.16	0.17
F-19	1235	80	0.46	0.30	0.46	0.24
F-19	2280	50	0.43	0.28	0.40	0.23
F-19	5225	30	0.39	0.25	0.36	0.21

Although our calculations for the alpha efficiencies of single peaks lead to very good results in agreement with measured and reported values of $\eta_{\alpha\beta}$, the model is underestimating the efficiencies after absorption of high energetic ions. Nevertheless the model describes the relative increase of the high temperature peaks (peak 6 and 7) with increasing LET of absorbed radiation.

DISCUSSION

The excellent agreement of experimental results and results of the applied track structure model confirm, that the peak ratios in glowcurves from LiF-TLDs are determined by the microdosimetric distribution of the energy density. For high energetic ions the model does not describe the absolute values of peak efficiencies sufficiently, but qualitatively the increase of peak ratios with increasing local energy density is described. The deviation of the calculated results from the experiments with high energetic ions is caused by at least two reasons:

1. The model used for the calculation of energy density in the particle tracks is neglecting the angel distribution of delta electrons and is assuming that all electrons leave the track normal to the track axis. This leads to greater track diameters in the model. This error is especially high for high energetic particles.
2. The model is assuming homogeneous tracks. This is more or less true for low energy alpha tracks. For high energetic particles the statistical nature of energy deposition has to be taken into account. High energetic delta electrons generate tracks itselves causing an energy deposition with a treelike shape. For this reason the homogeneous discription causes considerable errors.

Therefore improvements of the model are intended by the intoduction of statistical models for particle tracks. Furthermore a better understanding of the microdosimetric properties of TLDs and a consequent substitution of macroscopic parameters (like LET) by microdosimetric values (like specific energy) opens the opportunity to use TLDs as micro-dosemeters in order to get important information for a better understanding of biological effects after absorption of different kinds of radiation.

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PAPER TITLE Monte Carlo Simulations of Development of Electron Avalanches in Proportional Counters at high E/N Values.

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

Monte Carlo code AVALAN was developed to simulate an electron avalanche development under different operating conditions namely different counter geometries, gas pressures and electric fields. The first simulations were performed in low pressure proportional counters filled with methane gas. They allowed us to study all temporal and space aspects of the electron avalanche development. The dependence of gas gain, radial and time extend of the avalanche on gas pressure are described. We explain our results by the existence of *seeding* region in the avalanche where a relatively small number of electrons is created. Each of these electrons gives subsequently rise to a *partial* avalanche in the proper avalanche region. Results of the simulations are compared to measured data and to Boltzman equation results. The effects of the electron-molecule cross-sections uncertainties on the final results are discussed.

CALCULATION OF PHOTON DOSE FROM POSITRON SOURCES

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Abstract
This paper reports the results of detailed calculations of the different contributions to the total dose, where bremsstrahlung, annihilation and decay are considered. The calculations are carried for some ten radionuclides and for radiation protection purposes . The results were obtained by using analytical methods and Monte-Carlo techniquues. In order to have a presentation being consistent with gamma- emitting radionuclides, the results are given in terms of kerma rate constant in air of a shielded positron source.

INTRODUCTION

Positron emitting sources are becoming increasingly important in many applications of radiation. Sources are positron emitting radionuclides, positron beams, and induced as contamination at accelerators. Positron emitting radionuclides are used in nuclear medicine, e.g. positron emission tomography (PET) or in scientific applications. They are produced by accelerators (e.g. N-15(p,n)0-15 or by reactors (e.g.Cu-63(n,γ)Cu-64.
Since these sources both emit and produce some different kinds of radiation (positrons, bremsstrahlung, annihilation radiation), they contribute in different forms to all problems interesting in radiation protection, where dose assessment and shielding are the most important.

In order to make quantitative assessments, some aspects will considered in turn. The beta spectrum of the radionuclide were recalculated. Since detailed knowledge of positron interaction with matter is required to calculate quantities needed for shielding and dosimetry.
The work was separated into the following subchapters: calculation of positron spectra, assessment of the required absorber thickness, annihilation and bremsstrahlung production in the absorber leading to photon spectra for moneenergetic positrons. Radionuclide spectra, taking into account decay photons, are used for dose rate constant calculation

CALCULATION OF PHOTON PRODUCTION

Photons are produced in the absorber while the positron is slowing down via bremsstrahlung and annihilation processes. In addition, there are the decay photons in many of positron emitting nuclides, which are characteristic of the individual nuclide. From the Monte Carlo calculations, Fig. 2 shows as an example the relative intensity of the bremsstrahlung and the annihilation photons from a 1MeV positron beam with the respect to the thickness of the aluminium slabs.

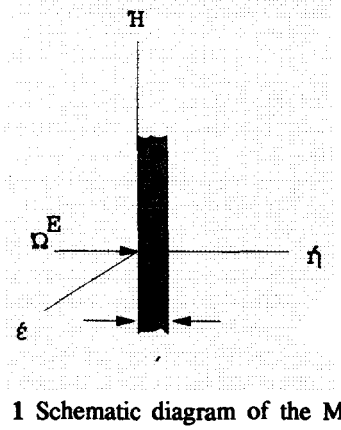

Fig. 1 Schematic diagram of the Monte Carlo simulation in which a positron beam is normally incident on the foil along the Z-axis.

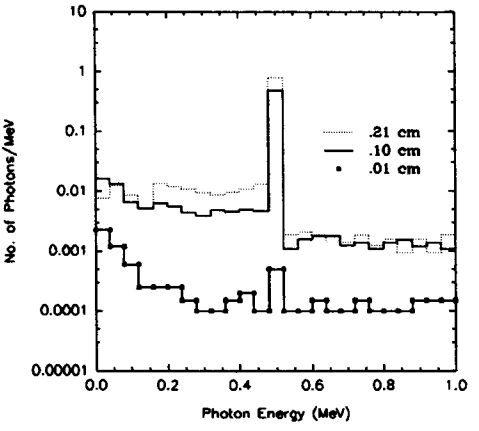

Fig. 2 Transmitted spectra of bremsstrahlung and annihilation photons from 1 MeV positrons on aluminium foils 0.1, 1.0, 2.1 mm thick.

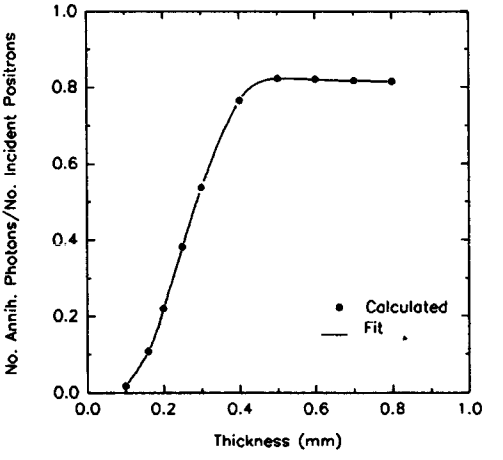

Fig. 3 Number of annihilation photons per incident positron from Cu-64 produced in aluminium foils.

PRODUCTION OF ANNIHILATION PHOTONS

The production of annihilation photons increases with the increase of the absorber thickness reaching a maximum when the thickness is equal to or more than the range of the maximum energy of the positrons, as shown in Fig 3 for Cu-64 source incident on aluminium absorber.

PRODUCTION OF BREMSSTRAHLUNG PHOTONS

As the production of the bremsstrahlung photons increases with the increase of the atomic number, and since only low Z-material is of concern here for the shielding of the positron emitting nuclides, it is expected that the contribution from bremsstrahlung would be minimal. However it is instructive to see how much this contribution is, especially that most of the work in the literature dealt with electrons only. Calculations of the bremsstrahlung production were first done for beams of 1, 2, and 3 MeV normally incident on aluminium slabs of different thicknesses, to obtain the

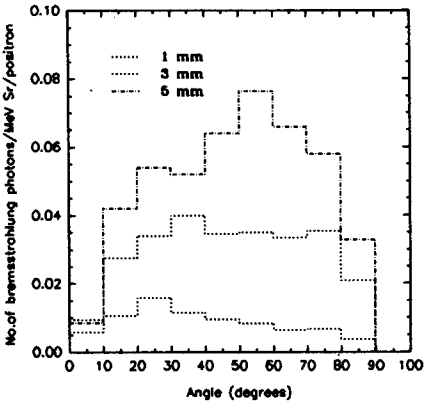

Fig. 4 Transmitted angular distribution of bremsstrahlung photons produced by 2 MeV positrons in aluminium foils.

average energy and number of photons produced per incident positron. For bremsstrahlung production in aluminium from a 1 MeV positron beam, there is a preference for transmission in the forward direction between 20° and 30° for a slab of 0.01 cm thickness. Increasing the thickness by a factor of 10, the magnitude of the spectrum increases by more than a factor of 2, and maximum direction is in 50°- 60°. Further increase of the thickness of the slab by a factor of 2, increases the bremsstrahlung production by about a factor of 2, and the maximum direction is shifted to 60°- 70°, but in a broader direction. In the case of the 3 MeV positron beam, the bremsstrahlung photons are transmitted in the forward direction with maximum between 10° and 20° when the thickness is 0.05 and 0.10 cm. The increase in the maximum is about a factor of 2 between 0.05 and 0.10 cm. With the increase in thickness the scattering increases and the photons leave the slab with higher probability of larger angles as can be seen in Fig. 4.

TOTAL PHOTON SPECTRA

As an example, transmitted spectrum of photons from a positron emitting nuclide is given below in Fig. 5 for Na-22, incident on aluminium slab whose thickness was equal to the range of the respective nuclide. It is clear from the spectra that the bremsstrahlung is low energy photons. These bremsstrahlung photons are comparable to the 0.511 MeV annihilation photons in number.

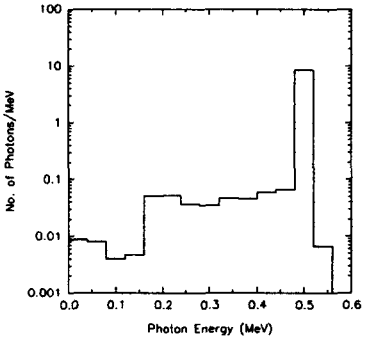

Fig. 5 Transmitted photon spectrum of Na-22 through aluminium foil whose thickness is equal to the range of the endpoint energy of Na-22.

DOSE RATE CONSTANT

The results of the calculation are compiled in the following table.

NUCLIDE	Air kerma rate constant (mGy m²/GBq hr)			
	Decay	Annih.	Brems.	TOTAL
C-11	0	2.309E-1	3.587E-4	2.313E-1
O-15	0	4.038E-2	2.200E-4	4.060E-2
F-18	0	1.308E-1	8.412E-5	1.309E-1
Na-22	1.220E-1	9.950E-2	2.318E-4	2.217E-1
Al-26	4.124E-1	2.401E-1	1.865E-3	6.544E-1
Sc-44	3.817E-1	3.628E-1	3.902E-3	7.484E-1
Cr-49	1.113E-2	1.281E-1	1.725E-3	1.410E-1
Fe-52	2.387E-1	1.039E-1	4.535E-4	1.282E-1
Mn-52	1.567E00	3.500E-2	8.887E-5	1.602E00
Ni-57	2.424E-1	7.883E-2	3.772E-4	3.216E-1
Cu-61	4.745E-2	1.888E-1	1.584E-3	2.378E-1
Cu-62	3.376E-3	7.765E-1	1.751E-2	7.974E-1
Cu-64	7.898E-4	2.515E-2	7.865E-5	2.602E-2
Zn-65	2.707E-2	7.576E-4	7.649E-7	2.783E-2
Ga-68	5.922E-3	1.574E-1	2.636E-3	1.660E-1
Zr-89	1.778E-1	4.939E-2	2.582E-4	2.274E-1

References see: K.A.Nahdi(1993) Thesis Dr. techn Technical University of Vienna 1993

EVALUATION OF THE W-VALUE OF VARIOUS GASES INCLUDING He-3 AND H-2 USING TRITIUM BETA PARTICULER

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1. Introduction

A new technique was studied and developed to aim at precise determination of W-values[1-4] of various gases using unit litter ionization chambers and tritium, which emits low energy β-ray. What was studied and developed :
(1) Ionization chamber and measuring gases filling system were designed and constructed. Characteristics of the ionization chambers were observed.
(2) Wall effect was examined.
(3) Effects of hydrogen as impurity on the ionization current were examined.
(4) The W-values of $^1H_2$, $^2H_2$, ^3He, ^4He, N_2 and Ar were respectively estimated.

2. Experimental Method

2.1. Ionization Chamber

The cross-sectional view of the measuring ionization chamber is shown in Fig.1. The chamber consisted of a stainless steel cylindrical high voltage supplying electrode and a collecting electrode, which was set on the central axis of the cylinder. The two electrodes were air tightly connected by pure alumina insulator with a guard ring. The inside of the cylindrical chamber was well polished like a mirror finish. The ionization chamber was electrically isolated from the filling system by the insulator tube.

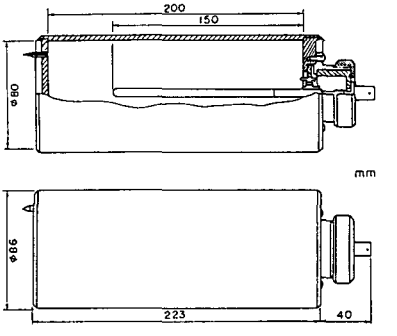

Fig.1 Unit litter ionization chamber

2.2. Measuring System

The block diagram of a measuring system is shown in Fig.2. Current measurements were made of a vibrating-reed electrometer. The ionization chamber electrodes were joined to the preamplifier of the vibrating-reed electrometer with a flexible shielded co-axial cable. The guard ring was operated at the level of the ground potential. A high voltage source supplied negative voltage from zero to 5kV to the cylindrical electrode.

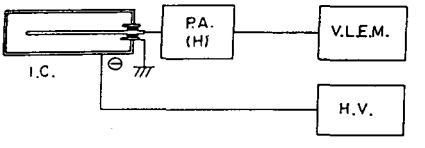

Fig.2 Block diagram of measuring system

I.C.: Ionization chamber, P.A.(H): Preamplifier of vibrating-reed electrometer, H.V.: High voltage supply, V.L.E.M.: Vibrating electrometer

2.3. Gas Filling System

The schematic diagram of the gas filling system is shown in Fig.3. This system was used for supply, recovery and storage of various gases and tritium gas. The system composed of stainless tubing, tube fittings and metal bellows valves, was joined to a vacuum pumping system. The gas filling system was evacuated routinely below 3×10^{-7}mbar. Tritium gas was supplied from a glass ampoule with a breakable seal. Tritium gas was stored into four unit litter vessels. Then, tritium in the vessel was conducted into an empty ionization chamber at first. Measuring gas was added to it.

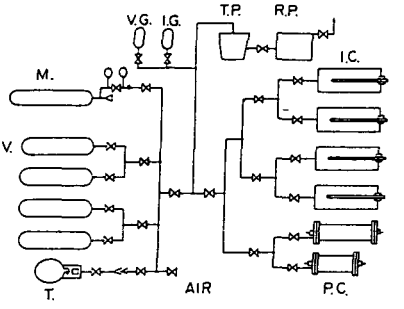

Fig.3 Schematic diagram of gas filling system

R.P.: Rotary pump, T.P.: Turbo molecular pump, V.G.: Baratron gage, I.G.: Ion-gage, M: Measuring gas container, V: Cylinder vessel, T: Tritium ampoule, I.C.: Ionization chamber, P.C.: Proportional counter

2.4. Gamma-Ray Irradiation

The γ -ray irradiation facility, which has 89TBq of ^{60}Co, belongs to Nagoya University. The ionization chambers were installed at a distance of 2.0m from the center of the ^{60}Co source.

2.5. Experimental Procedure

Saturation curves of various gases filled in the ionization chambers were measured. Tritium filled in the vessel was introduced to four ionization chambers at the same pressure to repeat other gases in the same procedure. Then, $^{1}H_2$, $^{2}H_2$, ^{3}He, ^{4}He, N_2 and Ar were added into the each ionization chamber at 5×10^{3}mbar respectively, afterwards, negative potential was applied to the cylindrical electrode and elevated stepwise. The collection current was measured out in each chamber. Measurements of the wall effect of various gases were carried out. In the same tritium content, ionization currents were measured by stepwise increased pressure and various gases respectively. Used tritium was provided a mixture of $^{1}H_2$ and $^{3}H_2$. In order to examine impurity effect in the gases used in the chambers, ionization currents of the chambers filled with mixed gases, $N_2 + {}^{1}H_2$ and $Ar + {}^{1}H_2$, were measured in the ^{60}Co γ -ray field.

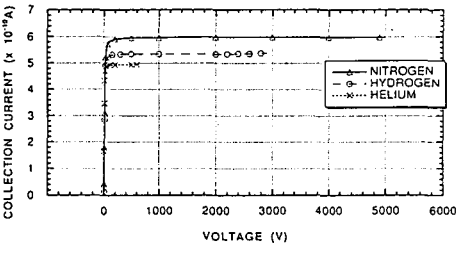

Fig.4 Saturation curves

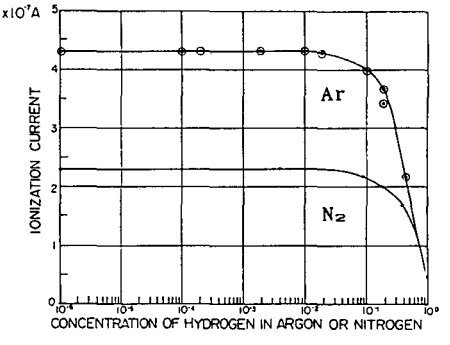

Fig.5 Ionization current versus concentration of H_2 in N_2 & Ar

3. Results and Discussion

3.1. Saturation Characteristics

The saturation-currents on N_2, H_2 and He are shown in Fig.4. The voltage ranges of the flat saturation-currents on N_2, H_2 and He were 5×10^2 to 4.9×10^3, 2.0×10^2 to 2.8×10^3 and 5.0×10 to 6.0×10^2V at 5.0×10^3mbar respectively. The ionization chambers were operated at the middle voltage of each range.

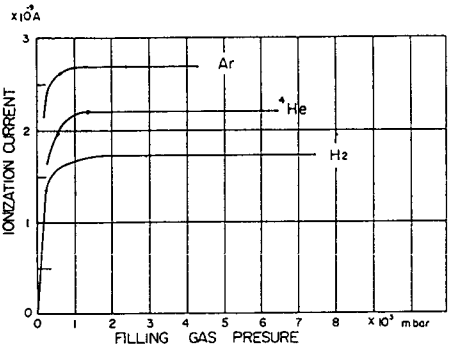

Fig.6 Ionization current versus filling gas pressure

3.2. Effects of Impurity Hydrogen

The ionization currents versus to hydrogen concentration in argon or nitrogen by γ -ray irradiation are shown in Fig.5. The ionization currents were unaffected below 1% of hydrogen mixture. Therefore, each character of the ionization chamber was unaffected to add $^1H_2$ & $^3H_2$ below 10^{-2}% in measuring gases.

3.3. Wall Effect

Results of the measurement of the wall effect are shown Fig.6. The ionization current decreased as low as to the pressure of the measuring gas. The decreased current was mainly occurred by absorption of tritium β-ray on the chamber wall. Though the wall absorption was decreased at high pressure. Above 1×10^3mbar, the ionization currents were saturated. The wall effect was neglected above the 2×10^3mbar.

From above results, the W-values of the various gases will be obtained by the techniques easily. The W-values of N_2, $^1H_2$ and He were estimated 34.6, 37.0 and 43.5eV respectively.

4. Conclusions

From the above experimental results, it was concluded that the tritium adding method was satisfactorily used to measure the ionization current in the various gases to determine the W-values. The conclusions about the present studies are the followings;

1) Ionization chambers were designed and manufactured.
2) A metallic gas filling system was constructed.
3) Saturation-currents on N_2, H_2 and He were flat over a wide voltage range.
4) When the gases contained hydrogen below 1%, the ionization current did not change, in the case of γ -ray irradiation.
5) Wall effect was neglected at pressure above 2×10^3mbar.
6) W-values of N_2, $^1H_2$ and He were obtained and the value of other gases will be reported at the IRPA9.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Mini-Presentation

PAPER TITLE Spectral Distributions of Ambient Dose Equivalent for Environmental Radiation Fields

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

Spectral distributions of an ambient dose equivalent for different environmental radiation fields were modelled using a Monte Carlo method. The radiation fields considered include both the natural as well as the man-made fields. Further, conversion factors from the air kerma into the ambient dose equivalent, H^*/K_a, were calculated for the fields. These were compared with experimental values of conversion factors obtained from *in-situ* gamma spectrometry. Variations of H^*/K_a and their influence on dosimetry of the environmental radiation fields will be discussed.

SEPARATION OF SR/Y-90 AND TL-204 BETA PARTICLES WITH MODIFIED TELEDYNE P-300-AS BADGE TO MEET THE REVISED ANSI N13.11(1993) REQUIREMENTS

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ABSTRACT

This paper describes a method to separate Sr/Y-90 and Tl-204 beta particles by a simple modification of Teledyne P-300-AS TLD badge system, which is currently used in the Korea Atomic Energy Research Institute for personnel dosimetry, by attaching small ^{7}LiF disk to the back-up open window filter area(A5). This modified badge system can evaluate the delivered dose from both Sr/Y-90 and Tl-204 within ±5% error bound even in the randomly selected irradiation condition

INTRODUCTION

The response of many personnel thermoluminescence dosimeters is extremely energy dependent when they are used in beta particle fields of somewhat significant levels in many working places. This energy dependence is generally taken into account by considering the response of the TL dosimeters as a function of the beta energy, and for this reason, low energy Tl-204 beta source was newly added in personnel dosimeter performance test criteria in addition to high energy Sr/Y-90 beta source in the revised ANSI N13.11(1). Both sources are recommended for use as reference beta sources by ISO Standards 6980(2).

In most cases the dosimeters for measuring beta doses use only one filter area, generally called open-window area, and usually evaluate beta doses by gross amounts of energy response not by separating their energies. To separate beta particles by energies, it is possible to improve the badge design by adding one more supplementary dosimeter with different filter in the badge case.

The Teledyne P-300-AS TL dosimeter badge system, used in the KAERI for personnel dosimetry, could not separately evaluate the beta doses from Sr/Y-90 and Tl-204 beta particles because this badge had an originally thick beta shield area(A2) enough to stop all the weakly penetrating beta particles. If we evaluate the beta dose only using open window filter area(A1) for beta particle mixtures, it would result in underestimation for Tl-204 beta dose because of the shielding effect by 7 mg/cm^2 mylar on the open window filter area(A1).

So, the purpose of this paper is to describe a method to separate Sr/Y-90 and Tl-204 beta particles by attaching small ^{7}LiF disk to the back-up open window filter area(A5) of the original Teledyne P-300-AS badge system, and to verify the performance of this method.

EXPERIMENTAL DETAILS

Teledyne P-300-AS multi-element badge case and C-300-A beta/gamma TLD card are used to determine the response of the TLDs to the beta particles. This badge system consists of 4 main areas and 4 back-up areas. The open window filter area(A1) simulates body surface and consists of a 7 mg/cm^2 black polyethylene mylar on the open window of the badge case. The beta shield filter area(A2) attenuates most beta energies less than approximately 2.2 MeV. To separate the beta particles with energies, it was covered with small ^{7}LiF disk on the back-up open window filter area(A5) to attenuate most of the Tl-204 beta particles and to measure Sr/Y-90 beta only. The filter materials in badge case are given in Table 1.

Table 1. Filter Materials for Each Area of Modified P-300-AS Badge Case.

Area	Filter Materials and Thicknesses
A1	7 mg/cm^2 Black Polyethylene
A5	7 mg/cm^2 Black Polyethylene + 211 mg/cm^2 ^{7}LiF
A2 & A6	863 mg/cm^2 , Plastic
A3 & A7	866 mg/cm^2 , Al
A4 & A8	1,245 mg/cm^2 , Al + Cu + Sn/Pb

The dosimeters were placed on a 30 X 30 X 5 cm PMMA slab phantom and irradiated by the PTB certified Sr/Y-90(74 MBq) and Tl-204(18.5 MBq) source in the Beta Secondary Standard System(Buchler Co., Germany) at the Calibration Laboratory in KAERI. The absorbed dose rate was given by the PTB certificate of each source at specified distances.

RESULTS AND DISCUSSIONS

The TL response as a function of absorbed dose was measured for Sr/Y-90 and Tl-204 radiations between 1.5 mGy and 10 mGy according to the requirements of ANSI N13.11. The linearity of TL response in this range is shown in Fig. 1. The response for Sr/Y-90 betas is linear and the measured doses are quite the same as the delivered doses. But even though the response of the TL dosimeters to the Tl-204 is linear, the measured doses are about 20 % lower than the delivered. From this a beta correction factor(BCF) for Tl-204 was derived to be 1.25, and the measured TL response for Tl-204 was then corrected by this BCF. The corrected dose shows good agreement with the delivered as seen in the same figure.

If the types of beta particles are known, the beta doses can be easily evaluated by multiplying the BCF obtained above. But in the case of randomly choosing beta sources to irradiate each dosimeter as defined in ANSI N13.11, the delivered dose would be underestimated by about 20 % for the Tl-204 beta if it is not separated into Sr/Y-90 and Tl-204 betas as shown in Fig.1. To separate two betas, the ratios of A1 to A5 for the case of covering and uncovering ^{7}LiF disk on A5 were obtained and the results are given in Table 2.

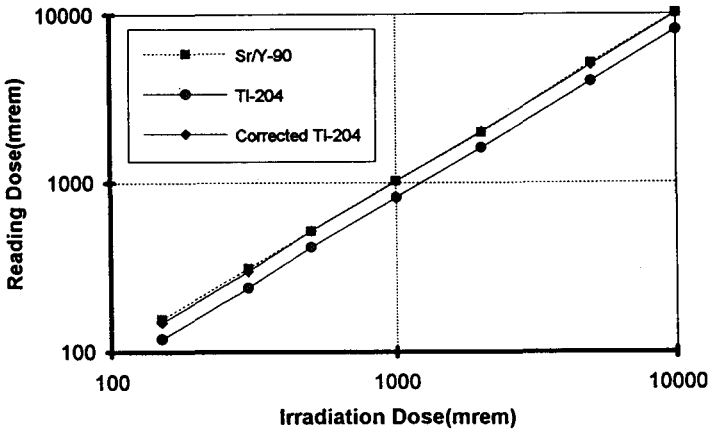

Figure 1. Linearity of Sr/Y-90 and Tl-204 Beta Particles.

Dose evaluation algorithm detects the beta or beta mixtures at first, separates the type of beta sources by A1/A5 ratio, and then evaluates beta dose by multiplying the relevant beta correction factor which is 1.0 for Sr/Y-90, 1.2 for Tl-204.

Table 2. Delivered to Evaluated Dose Ratios with and without Cover of ^{7}LiF Disk.

Sources	A1	A5 Without Cover	A5 With Cover	A1/A5(With Cover)
Sr/Y-90	1	1	0.9	1.1
Tl-204	0.8	0.8	0.05	16.0

To verify the performance of the modified badge system, a simple performance test was conducted and the results are given in Table 3. The test categories and delivered doses were selected from ANSI N13.11. It is shown that all results meet the performance test criteria even without any modification of original P-300-AS badge case, but the performance index (bias plus standard deviation) is much larger than that of modified P-300-AS badge case, by which the accuracy can be improved by more than 20 % compared with the original Teledyne design.

Table 3. Results of Performance Test for the Modified P-300-AS Badge Case.

ANSI N13.11 Test Categoriy	Delivered Dose(mSv)	Reported Dose(mSv)	Test Results | B | + S[(1)]	Tolerance Level
VA Sr/Y-90	10	9.8 (9.8)[(2)]	0.03 (0.04)	0.5
VB Tl-204	10	9.7 (7.8)	0.04 (0.25)	0.5
VC Sr/Y-90 + Tl-204	10	9.8 (8.7)	0.05 (0.16)	0.5

(1) B : Bias, S : Standard Deviation
(2) Values in parentheses are results from original P-300-AS Badge Case.

CONCLUSION

Teledyne P-300-AS badge case and dosimeters used for personnel monitoring of beta radiations suffer from an energy dependent problem because this does not provide the filters needed to evaluate different types of betas separately. To improve this, back-up filter area A5, having same filter material and thickness as open window filter area A1 for beta measurement, was covered with a small ^{7}LiF disk to provide multi-element filter areas for separating beta energies.

The TL response to beta radiation energy showed that the evaluation data for Tl-204 was about 80 % of the response obtained with Sr/Y-90 source. However, by separating Tl-204 and Sr/Y-90 beta sources with a modified badge system, the delivered dose can be evaluated within $\pm$5 % error bound even in the randomly selected irradiation condition.

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TRITIUM DETECTION WITH $BeO:TiO_2$ -CERAMIC TL DETECTORS

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INTRODUCTION

The problems associated with registration of alpha and very soft beta radiation are well known and they spur a search for new ways of their solution (1). We used new thermoluminescent (TL) detectors on the basis of $BeO:TiO_2$ ceramics for ^3H ($E_{\beta\ max}$ =18 keV) detection. A characteristic feature of the detectors is their complete optical opacity and existence of a thin (20 μm) sensitive)layer on the $BeO:TiO_2$ surface (2). Thanks to these properties the detector sensitivity to irradiation with short-range charged particles was much higher than to irradiation with accompanying quanta (3).

EXPERIMENTAL

Thermoluminescence was measured using a DTU-01 instrument (Lithuania) The TL peak of the glow curve in the temperature range from 420 to 440 K is used only for the read-out procedure.

Two methods of tritium detection were used in the study. According to one method the detector was irradiated by a standard source of tritium arranged at a distance of 0.5-1 mm from its surface. The TL signal amplitude increased linearly with an increasein the exposure time. The measurement results showed that the lower detection limit is 10 of β-particles found on the detector surface.

According to the other method the detector was brought into a momentary mechanical contact with the surface of the source. It was found that the detector based on the $BeO:TiO_2$ ceramic possesses the ability to accumulate and retain for a long time tritium diffusing into the sensitive layer of the material. The TL signal of such a detector is induced by its self-irradiation. The measurements showed that the TL output is proportional to the time of the detector self-irradiation, i.e. the effect of "dose accumulation" takes place. This property persisted for several weeks, even when a few tens of cycles of TL read-outs were performed during that period. Shown in the Figure is the measured TL response vs. the "self-irradiation" time of two detectors (curves 1 and 2).

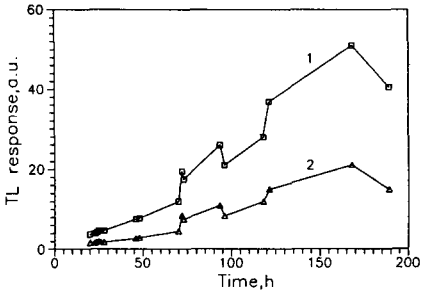

Figure. TL response vs. self-irradiation time of two $BeO:TiO_2$ ceramic detectors

These experiments were carried out using purposefully chosen detectors, which had different TL sensitivities determined by the former method as described above. It follows from the Figure that an increase in the self-irradiation time is indeed accompanied by an almost linear rise of the TL response. A comparison of the two methods under discussion shows that an increase in the TL output due to the "dose accumulation" effect provides an additional 10- to 100-fold rise of the tritium detection sensitivity. Besides, as is seen from the Figure, the curves 1 and 2 have sharply nonlinear portions. Most surprisingly, the said portions are fully coincident. It is unlikely that this synchronous change in the TL sensitivity is caused by migration of tritium in the detector surfacc. Thcn diffusion constants of tritium in thin surface layers of two different detectors should be ideally coincident, which is, in our opinion, highly improbable. We believe the most plausible explanation of the presence of synchronous "steps" in "TL response - accumulation time" curves 1 and 2 obtained for two detectors with different sensitivity is instability of the TL read-out device. Specifically, we attribute the observed effect to instability of the detector heating unit. During the measurement procedure sensitivity of the recording circuit before and after heating of the detectors was constant as determined using a radioisotope reference light source. It may be assumed with a certain degree of confidence that the said two detectors helped us reveal a time fluctuation of the stability of the read-out device; the time fluctuation could not be detected with the reference light source.

The self-irradiation effect was examined additionally by the method of thermally stimulated exoemission (TSEE). The measurements were made in vacuum. The effect was fully reproducible for several hours but its sensitivity dropped with every heating-measurement cycle. The effect ceased if the samples were kept in vacuum for a few days. When TL was measured in air, sensitivity of the detectors lowered only 1.6-1.8 times in 2 months. This fact can be accounted for by a more intensive withdrawal of tritium from the surface layer of the detectors subject to heating and storage in vacuum. This inference does not contradict the mechanism of tritium diffusion in the lattice of the BeO ceramic (4).

CONCLUSION

Our preliminary studies of TL in BeO:TiO_2 -based detectors show that they may be useful for determination of tritium contamination and serve as TL standards for testing of read-out devices.

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Poster No.: P35 - 26

RADIATION VOICE MONITORS

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1. ABSTRACT

Radiation voice monitors, which serve to announce the dose equivalent (or dose rate) by voice to a worker wearing bulky protective clothing in high-radiation level and high-contamination areas, were developed to reduce the exposure to radiation of the worker. Examples of the applications of such monitors are introduced hereunder.

2. PURPOSE OF DEVELOPMENT

In developing radiation voice monitors, the aim was to enable a worker and his supervisor, while working in high-radiation level and high-contamination areas, to hear the dose equivalent (or dose rate) in real time and select working locations which would minimize the dose equivalent, increase working efficiency and reduce the collective dose equivalent (man Sv).

3. BACKGROUND

In Japan where the utmost efforts are being made to reduce exposure to radiation, the average yearly dose per worker at nuclear power stations is as low as 1.0 mSv/year. However, the need inevitably arises for workers wearing heavy protective clothing to perform duties at certain times in high-radiation level and high-contamination areas.

In the past, an alarm meter carried inside the pocket of the protective clothing sounded when the specified period of time was completed, whereupon the worker ceased working and exited from such areas. Time controls were in place to prevent the worker's dose rate from exceeding the planned dose. However, the dose rate in the working environment varied widely depending on both location and time.

4. PERFORMANCE

Radiation voice monitors enable the worker to continue his duties while being informed through his earphone of the changing dose equivalent or dose rate encountered, this being particularly convenient in cases where, due to the bulky clothing, it is difficult to see the dose equivalent or dose rate displayed on the dosimeter. As the new type of monitor transmits such audio-data wirelessly, a supervisor far away can hear the dose equivalent of the worker in real time and provide him with adequate instructions.

The radiation voice monitor is equipped with an additional audio-communication function enabling communications to be exchanged between workers and their supervisors.

The specifications of the radiation voice monitor are shown in and Table-1.

5. EXAMPLE OF APPLICATION

Place	:	Nuclear facility A
Dose equivalent	:	0.5 ~ 20 mSv/h
Surface contamination density	:	$\sim 8 \times 10^2$ Bq/cm^2
Noise	:	70 ~ 90 dB
Protective clothing	:	Anorak suit, full-face mask
Planned collective dose equivalent	:	13-man mSv/period
Actual collective dose equivalent	:	8-man mSv/period

6. STUDIES ON REDUCTION OF DOSE EQUIVALENT

It was determined that the dose equivalent in the application example given in the above section 5. was reduced as a result of the following:

(1) Workers could perform their duties while constantly being informed by earphone of their dose equivalent. Therefore, work was not suspended as it became unnecessary to exit the radiation area merely to satisfy the time control of the dose equivalent. 40%

(2) As the high-dose equivalent locations were known, workers were able to avoid working in these locations. 20%

(3) Because the supervisors heard the workers' dose equivalents in real time, they were able to supply adequate instructions.20%

(4) Workers were able to carry out work while hearing other co-workers' dose equivalents; that is to say, respective dose equivalents could readily be identified. 20%

100%

Furthermore, studies on the following index were conducted from the standpoint of optimization of radiation protection.

$$\frac{\Delta X}{\Delta S} \leq \alpha$$

ΔX: Cost of radiation voice monitor (¥)

ΔS: Collective effective dose-equivalent (man Sv)
α : Dimensional constant expressing monetary cost assigned to a unit of collective dose for radiation protection rate (¥/man Sv)
Δ : Finite increment

Application time (total time) of radiation voice monitors will affect the index ΔX .

7. CONCLUSION

Radiation voice monitors were developed to optimize radiation protection and have been used over the past five years. The results of the application of the monitors indicate a very satisfactory reduction of dose equivalent.

Table 1: SPECIFICATIONS
(FM RADIATION VOICE MONITOR)

Detector	Silicon semiconductor
Radiation detected	γ (X) ray (80 Kev - 2 Mev)
Measuring accuracy	+/-20%, 662 Kev for Cs-137
Measurement range	0.01 ~ 99.99 mSv, 0.01 ~ 99.99 mSv/H
Audio	Speaker, earphone, FM radio
Audio output	Max. 0.2W
Transmission frequency	FM radio, 70 ~ 100 MHz
Power	Four dry batteries
Battery life	8 hours
Size and weight	2.5W x 32D x 103H (mm), 250g

DOSIMETRY CHARACTERISTICS OF GLASSES FOR GAMMA HIGH DOSES

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INTRODUCTION

The vitreous materials present non-comum characteristics; they also own special and useful technological properties, because of their atypical nature. Although a glass doesn't present the elevated crystal order, it is not destituted of a structure.

Ionizing radiation induces in glasses absorption bands that depend on the material constitution and on the irradiation conditions. The colour change is due to the mechanism of oxidation and reduction of their elements and to the colour centers formation (1). Several high dose dosimetry methods have been developed, because of the increasing radiation use in industrial processes, as the sterilization of medical and food products(2).

In the present work the possibility of use of commercial glasses in gamma dosimetry was studied between 18Gy and 7.5kGy, through optical density measurements, using a densitometer and a spectrophotometer.

MATERIALS AND METHODS

Commercial glass samples, with dimensions of 11.50 x 50.00 mm and thicknesses between 2.00 and 2.07mm, were initially analysed (neutron activation method) and they showed several components as SiO_2 (71%), Al_2O_3 (0.90%), Fe(0.13%), Mg (4%), Na (15%), K_2O (0.40%) and Ca (0.13%).

Reutilization of the material was achieved by thermal treatments at 300° C during 15 minutes. All irradiations were made in air, using a panoramic ^{60}Co source Yoshizawa Kiko Co. Ltd. (10.8 Tbq).

Optical absorption spectra were obtained (air as reference) between 300 and 700nm, always 20 minutes after the irradiations, using the spectrophotometer Femto, model 482, Brazil, simple beam. A special densitometer was provided by MRA, Brazil, in order to allow measurements up to high doses using glass samples, with transmission between 400 and 550 nm.

RESULTS

In Figure 1 the optical absorption spectra of the studied glasses are shown. The glass samples are transparent in the visible region, before the exposition to ^{60} Co radiation, presenting an optical absorption band at about 310nm. After irradiation they become light brown coloured and a main large band at 420nm is shown; its intensity increases with the absorbed dose.

In order to test the batch uniformity and the response repeatability, the samples were 10 times irradiated with 1 kGy (^{60}Co). The batch uniformity presented variation coefficients of 1.8 and 3.5%, and the response repeatability, of 1.8 and 3.6%, for measurements taken at respectively the densitometer and at the spectrophotometer.

The thermal fading at room temperature can be seen in Figure 2 for both cases of densitometer and spectrophotometer. A steady initial decay (25% in 24 hours) and a subsequent slow decay can be observed. Only after 18 days the samples showed a constant response.

Figure 3 presents the calibration curves of the glass samples irradiated between 18 Gy and 7.5 kGy. A linear behaviour can be observed from 30 Gy to 7.5 kGy, for the spectrophotometer measurements. In the case of the results obtained with the densitometer, the linear region resulted from 0.1 to 1.5 kGy, with a subsequent supralinear behaviour up to 7.5 kGy and a tendency to saturation.

The low dose limits were determined for these glass samples as 19 and 100 Gy using as measuring instruments respectively the spectrophotometer and the densitometer. These same results can be obtained from Figure 3.

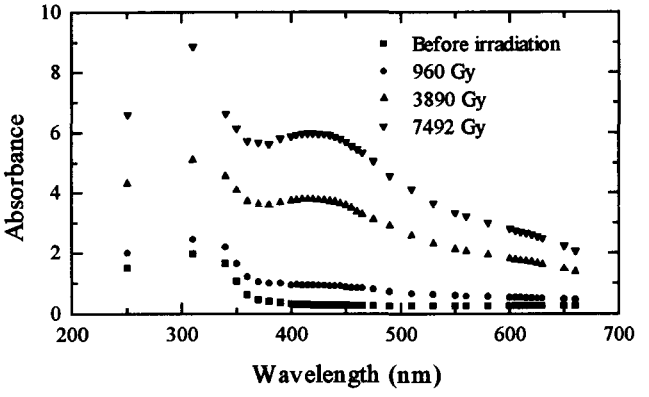

Figure 1 - Absorption spectra of the glasses before and after irradiation.

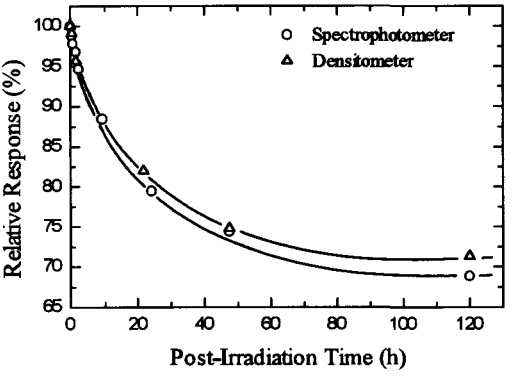

Figure 2 - Thermal fading at room temperature of the glasses irradiated with ^{60}Co.

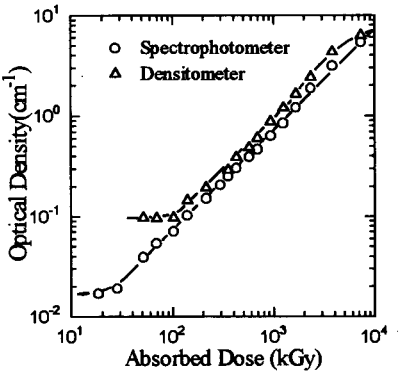

Figure 3 - Calibration curves of the glasses irradiated with ^{60}Co (2.3 kGy)

CONCLUSIONS

The commercial glass samples studied in this work present adequate dosimetric characteristics as batch uniformity, response repeatability, reutilization and linear behaviour in determined dose intervals, which make them suitable detectors for routine high dose dosimetry in the range of 18 Gy to 7.5 kGy. However, these glasses present an undesirable thermal fading effect, which does not constitute a great problem even in routine work, because the dose evaluations are normally made after a certain time interval. A correction factor has to be applied to the results.

The main advantages of these radiation detectors for gamma high doses are the very low cost and the measuring simplicity. The glasses can also be used as Yes/No detectors, because of their colour change with the absorbed dose.

ACKNOWLEDGEMENTS

The authors acknowledge the partial financial support of Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq).

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MEASUREMENT OF GLASS DOSIMETER RESPONSE FOR LOW ENERGY PHOTON USING SYNCHROTRON RADIATIONS

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INTRODUCTION

Radiophotoluminescent glass dosimeters made of silver activated metaphosphate glass are widely used for dosimetry of gamma-ray fields because of the unique characteristics such as repeatability, integrity during intermittent monitoring and reading, and extremely low fading. In addition, by utilizing a pulsed N_2-gas laser in readout system, the performance of the glass dosimeter has been improved on reproducibility of low-dose measurement(1) and expected to be widely used for personnel dosimeter.

The characteristics of the dosimeter, including the energy response for photons up to about 20 keV, had reported. However, few experimental data were presented for monochromatized low energy photons. Recently, high intensity photons with low energy has been widely used in synchrotron radiation facilities and the dosimetries including the personnel one is one of the serious safety problems for the low energy photons. Therefore, the responses of glass dosimeter badge, made by Toshiba Glass Corp., have been measured for low energy photons by using monochromatized synchrotron radiation and compared with the results by using X-ray tube. In order to get the knowledge of characteristics of the detector response, radiation transport phenomena have been simulated and compared with the experimental results.

EXPERIMENT

The GD-403 type glass dosimeter(2) consists of 4 elements with respective different filters (Al of 1mmt,Sn of 1mmt, polyester films of 35 μ mt and 250 μ mt), as illustrated in Fig.1. The response of glass dosimeter for low energy photon ranging from 7 keV to 37keV have been

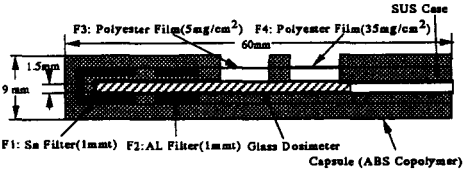

Fig.1 Cross-sectional view of GD403 type glass dosimeter badge

measured by using monochromatized synchrotron radiations at a vertically polarized wiggler beamline, BL-14C, at Photon Factory in KEK. Synchrotron radiation from the wiggler was monochromized to $\Delta E/E \sim 10^{-3}$ - 10^{-4} band width. Absolute photon intensities were monitored with a free air ion chamber calibrated with a new type calorimeter(3). The dosimeters were scanned horizontally in order to uniformly irradiated. The higher harmonic components of monochromatized photons reduced to less than 0.3% by detuning the double crystal monochromater, which components were monitored by a HP-Ge detector.

The detector was irradiated by using an X-ray tube operated at 30kV with Al

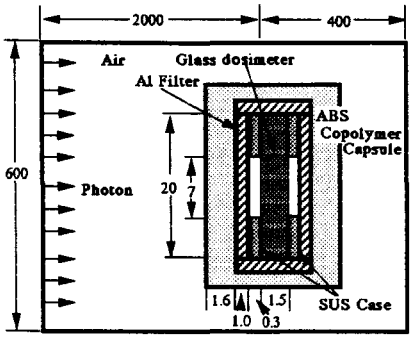

Fig.2 Calculational configuration of glass dosimeter with Al filter

filter of 4.01mm thickness for 24.1keV effective energy and at 50kV with Cu filter of 0.59mm for 39.4keV. The quality indices that is photon effective energy to tube voltage ratio, are 0.8 and 0.79, respectively. Gamma-ray of ^{137}Cs (662keV) was also used for normalization.

In readout operation, the FDG-60 automatic system(4) was used. On the system, the N_2-gas laser incidents with slightly inclined on the side of glass dosimeter and the radiophotoluminescences were detected outside the front of the dosimeter.

MODEL CALCULATION

CYLTRANP was used in ITS3.0 and 4 models were employed for the 4 filters. For example, figure 2 shows the calculated configuration of slab-cylinder model of glass dosimeter with Al filter. To obtain an effective radius and thickness, calculations were made under the following conditions; (a)deposited energy is independent of the depth of detector for 37keV and 662keV photons, (b)deposited energy is reduced by shadow effect of SUS case , and (c) deposited energy is dependent upon the readout thickness of the detector for 7keV photons.

The effective radius of the detector can be expressed as a function of the ratio of the energy deposition with 37keV photons to 662keV photons and determined by the interpolation using the function. After knowing the effective radius, the effective thickness can be estimated from the ratio of the energy deposition with 7keV photons to 37keV photons. Energy responses of the glass dosimeter were calculated by using the effective radius and the thickness.

Table1 Comparison of the measurements with X-ray tube and synchrotron radiation for GD403 type dosimeter with various filters. The data are normalized at 662 keV photon energy per unit exposure.

	Filter1 (Sn)	Filter2 (Al)	Filter3 (Poly.5mg/cm^2)	Filter4 (Poly.35mg/cm^2)
X-ray (39.4keV)	0.0418	3.20	3.35	3.57
SR(36.86keV)	0.028	3.02	3.42	3.50
Ratio	1.49	1.06	0.980	1.02
X-ray (24.1keV)	-	1.83	2.98	3.05
SR(23.74keV)	0.007	1.56	2.69	2.75
Ratio	-	1.17	1.11	1.11

RESULTS

The responses of the dosimeter are indicated in table 1, and good agreements are obtained between the responses for 39.4keV effective photon by X-ray tube and 36.86keV photon of synchrotron radiation, except for the case of the Sn filter element. At 24keV photon energy, the responses by using the X-ray tube are more 10% higher than those of synchrotron radiation. This is pointed out that the higher energy components of X-ray are introduced to high sensitivities.

On the calculation of the energy deposition as a function of the depth of detector, it is confirmed to be depth independent of the photon energies above 30keV, that is, the ratio of the energy deposition for 37keV photon to that for 662keV photon is independent of the depth of the detector. From the model calculations mentioned above, 5.1±0.3mm was obtained for the effective radius of the detector. In Fig.3, the two solid lines show the calculations of the functions of the ratio of the energy deposition with 7keV photons to 37keV photons, indicating that the ratio depends on the detector thickness and

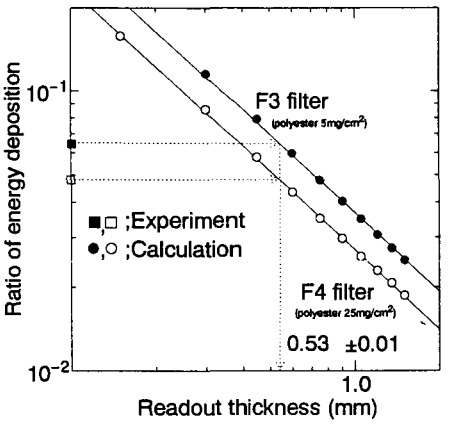

Fig.3 Dependencies of Proportion of deposited energy upon the readout thickness of the detector with 7keV to 37 keV photons.

experimental data for the detectors with polyester film filters, and obtained the effective thickness as 0.53±0.01mm.

The energy responses of the detector with each filters are shown in Fig.4 as the comparison between the experimental results and calculations with the effective radius and the thickness. As shown in Fig.4, fairly good agreements are obtained between the experiments and calculations, except the response of the detector with Sn filter because of the insufficient accuracy in the calculation.

CONCLUDING REMARKS

The energy responses of the glass dosimeter badge for monochromatized low energy photons ranging from 7 to 37 keV have been measured by using synchrotron radiation and the sensitivities have been investigated accurately. The comparison between the experimental results of synchrotron radiation and the Monte Carlo calculations, showed close agreements as to the energy response.

We found that sensitivity of detector was estimated highly in the results for low energy photons, especially below 24keV in conventinal method of X-rays, and that the response functions depended on the readout system, especially readout radius and thickness.

Dose estimation for the personnel within fields of low energy photons mixed with β-rays is currently underway for future studies.

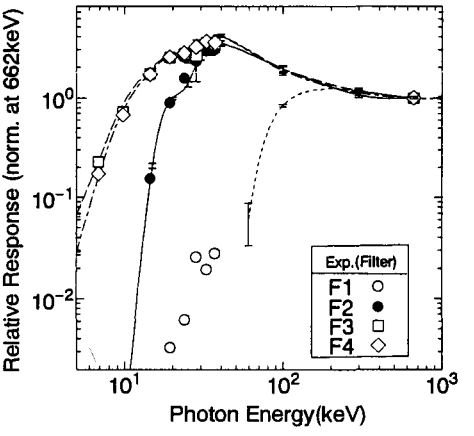

Fig.4 Energy response of glass dosimeter badge within free air .(○,●,□ and ◇ are experimental data with each filters and lines are calculated data by ITS3.0)

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PAPER TITLE Dosimetry of low energy photon radiation in the energy range from 6 to 15 keV.

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MAJOR SCIENTIFIC TOPIC NUMBER .3... (see page 7)

ABSTRACT (See instructions overleaf)

Photon radiation in the energy range below 15 keV occurs in many practical situations encountered in research, industry and medicine, and like beta radiation, belongs to the class of low-penetrating radiations. The dosimetry of such radiations is a concern in radiation protection as well as in radiation diagnostic when areas of unprotected superficial organs, such as the skin, are exposed. In order to monitor individuals in terms of the personal dose equivalent Hp(0.07), the operational quantity recommended by ICRU, specific thin detectors are required and their energy and angle responses must be known. The investigation of the dosimetric characteristics of such detectors implies the availability of well-characterised calibration beams. This paper describes the establishment and characterisation of low-energy photon calibration beams based on two radiation facilities, namely an X-ray facility providing 3 narrow spectrum photon beams centered at resp. 6.5, 8.5 and 12 keV, and a Fe-55 (6 keV) irradiation facility. As shown by Boehm et al.(1), an air-filled extrapolation chamber can be used as a primary standard from which absorbed dose to a given tissue equivalent material can be derived reliably. In the present work the extrapolation chamber method was used to characterise photon beams in terms of Hp(0.07;α°), the dose equivalent at a depth of 0.07 mm in the 4-element ICRU tissue slab for different angle of incidence (α) of the radiation. Furthermore, the air kerma Ka was measured at the same point enabling the determination of the necessary reference conversion coefficients from Ka to Hp(0.07;α°). Measured values are compared with calculated values (2) and, in addition, results are presented concerning the application of the established calibration fields to investigate and typetest individual dosemeters and to check how they comply with required performances.

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THEORETICAL CALCULATION OF GAMMA DOSE CONVERSION FACTORS FOR PERSONNEL EXTREMITY DOSIMETER

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ABSTRACT

In late 1996, The ANSI is planning to release a standard for the performance testing of personnel extremity dosimeters as ANSI N13.32 standard. The standard includes experimentally-determined gamma dose conversion factors used for determining the dose equivalents to the extremities at the reference depth of 7mg/cm^2. In this study, theoretical calculations of the gamma dose conversion factors for two types of extremity phantom, finger and arm phantoms which were specified in the ANSI N13.32, were performed using a MCNP code and compared with the experimental values of the ANSI N13.32. Finally, it is concluded that the all conversion factors for the two types of the extremity phantom show good agreement with the values of the ANSI N13.32 for each energy within 4.8% except for about 12% at less than 50keV energies.

INTRODUCTION

In Korea, over past one decade, personnel dosimetry researches for external, whole-body dose evaluation were primarily focused with little attention to dosimetry research for extremity dose evaluation. In the case of the United States, the draft American National Standards Institute(ANSI) N13.32 "Standard for the Performance Testing of Extremity Dosimeters" established the methods for testing extremity dosimetry systems, specifies the sources, energy ranges and irradiation geometries to be used during the performance testing[1]. In this study, for determination of the gamma dose conversion factors theoretical calculations of them for two types of extremity phantom, finger and arm phantom which were specified in the ANSI N13.32, were performed using a Monte Carlo N-Particle(MCNP) code and compared with the experimental values of the ANSI N13.32[2].

THEORETICAL BACKGROUND

In order to provide the accurate calculation necessary for dose evaluation with different extremity dosimeter systems, the use of well-determined gamma dose conversion factors is required. With computational geometry in mind, the coefficient, gamma dose conversion factor, $H_p(d)/K_a$ for converting air kerma to dose equivalent for primary photon beam of mono-energy(E) can be calculated according to the following equation:

$$\frac{H_p(0.07)}{K_a} = \frac{C\int \phi_{\text{ICRU tissue}}(E,d)\cdot E\cdot\left[\frac{\mu_{tr}(E)}{\rho}\right]_{\text{ICRU tissue}}}{\phi(E)\cdot E\cdot\left(\frac{\mu_{tr}}{\rho}\right)_{air}}$$

where C is the unit conversion factor [10^{12} Gy/(J/kg)],

$\phi_{ICRU\ tissue}$ is the fluence in the ICRU soft tissue material of extremity cylindrical phantom,

$\left[\frac{\mu_{tr}(E)}{\rho} \right]_{ICRU\ tissue}$ is the mass energy transfer coefficient of ICRU tisssue,

$\left(\frac{\mu_{tr}}{\rho} \right)_{air}$ is mass energy transfer coefficient of air, and

K_{air} = air kerma.

COMPUTATIONAL METHODS

Theoretical calculations of the gamma dose conversion factors for two types of extremity phantom, finger and arm phantoms using a MCNP transport code[3]. The gamma dose conversion factors in the range between 15keV and 1.5MeV were calculated using the MCNP code with the kerma approximation. The back-scatter factors are not considered in this calculation, since all dosimeters do not respond equally to backscatter. Each phantom was assumed to be irradiated by the idealized case of broad, unidirectional, and monoenergetic source particles normally incident on the front face of the phantom, which was assumed to be surrounded by vacuum. The computational irradiation geometry and cell arrangement for each extremity phantom is shown in Figure 1. The mass energy transfer coefficients of tissue and air for theoretical calculation of gamma dose conversion factors were determined using Storm and Israel data and Hubbel data, respectively[4, 5].

RESULT & DICUSSIONS

The gamma dose conversion factors calculated in this study and specified in the ANSI N13.32 for each phantom are shown in Figures 2 and 3. In the case of finger phantom, the gamma dose conversion factors theoretically-determined by using the MCNP code show good agreement with the values of the ANSI N13.32 standard for all energies within 3.5% except for about 10% in the range of less than 50keV energy as shown in Figure 2. And in the case of arm phantom, the gamma dose conversion factors theoretically-determined by using the MCNP code also show good agreement with the values of the ANSI N13.32 standard for all energies within 4.8% except for about 12% in the range of less than 50keV energy as shown in Figure 3. The cause of bigger difference at less than 50keV energies than at higher than 50keV energies for both phantoms may primarily be due to inherent difficulty of experimental measurements occurred by using a x-ray generator.

CONCLUSIONS

As a result of comparing theoretically calculated gamma dose conversion factors with experimental data specified in the ANSI N13.32, it is shown that there is a reasonable good agreement with the maximum difference ratio, 4.8% at all energies except for about 12% at less than 50keV energies. Therefore, it is concluded that gamma dose conversion factors determined by using a MCNP code in this study for finger and arm phantoms could be used for considerably reliable extremity dose evaluation in addition toss the experimental values of the ANSI N13.32.

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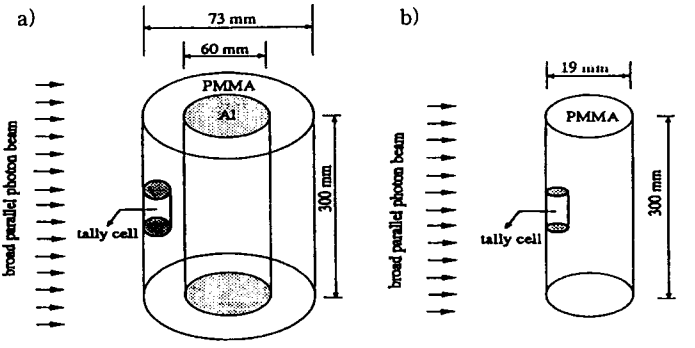

Figure 1. The computational geometry and cell arrangement for the extremity phantom: a)arm phantom, b)finger phantom.

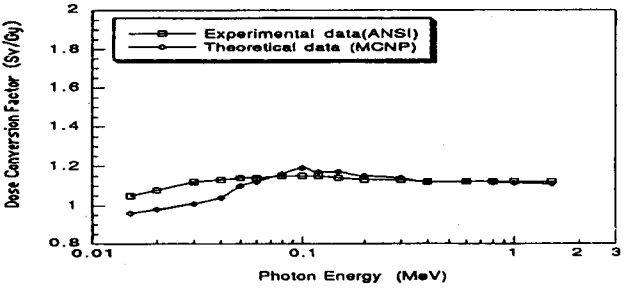

Figure 2. The dose equivalent conversion factor for the finger phantom of the ANSI.

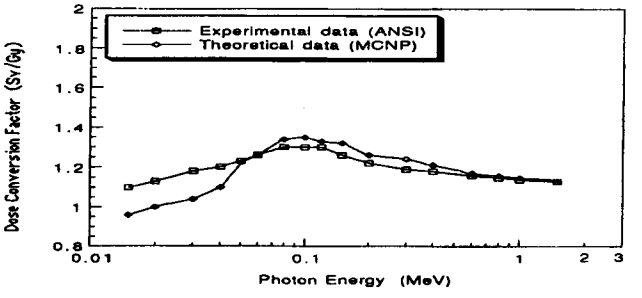

Figure 3. The dose equivalent conversion factor for the arm phantom of the ANSI.

PERFORMANCE OF A DIRECT ION STORAGE (DIS) DOSEMETER FOR INDIVIDUAL MONITORING

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INTRODUCTION

The development of the Direct Ion Storage dosemeter was triggered by the need to find an electronic successor for the aging Film- and TLD-dosemeters. As the basis of any passive dosemeter, there must always be a reliable means of storing information, such as the electron traps in TL materials. The new development was based on the same principle for information storage as is used in almost all microelectronic devices today; a nonvolatile FAMOS (Floating Gate Avalanche-injected MOS) memory cell. In electronics these devices are known as EPROMS, EEPROMS and Flash-Memories. In order for a memory cell to act as a dosemeter, it must be sufficiently sensitive to ionizing radiation to the extend that the small dose levels down to a few μGy encountered in radiation protection applications can be reliably detected.

In a nonvolatile solid-state memory cell the information is stored in the form of electronic charge being trapped on the floating gate of a MOSFET transistor. Original memory designs were only used to store digital information which meant that in each memory cell there was either a low amount or a high amount of charge stored to represent one of the two binary digits 0 or 1. In the recent years new types of nonvolatile memories have been developed and made commercially available to be used for storing analog information. This means that the amount of charge in each memory cell can be made fully variable and therefore, these memory cells can be used to store analog information directly. Analog-EEPROM memories are used today in speech recording applications for example in electronic telephone answering systems.

The radiation sensitivity of normal solid-state memory cells is inherently too low for use as detectors in radiation protection applications. The main reason for this is that these devices are specifically designed to be as insensitive as possible to ionizing radiation so that they could be used in space, military and nuclear industry applications without damage.

In an Analog-EEPROM memory cell, the charge on the floating gate can be set to a predetermined level by injecting electrons through the oxide layer. The charge is then permanently stored on the gate because in the normal operating temperature range the electrons have a very low probability to exceed the energy barriers in the metal-oxide and oxide-silicon interfaces. These types of memory cells are known to retain the stored charge for hundreds of years.

In order for ionizing radiation to have an effect on the stored charge, either new charge has to be brought to the gate or some existing charge removed. Ionizing radiation incident on the oxide layer does produce electron-ion pairs but due to the relatively poor mobility of charge carriers in the oxide, recombination occurrs and most of the free charge is neutralized before it has a chance to cross the metal-oxide interface. MOS dosemeters that are based on this principle are not sensitive enough for most radiation protection applications. The Direct Ion Storage (DIS) principle is based on allowing the surface of the floating gate to be in direct contact with the surrounding air. Ionizing radiation incident in the air space produces electron-ion pairs with extremely high mobility and if there is an electric field surrounding the floating gate, these charge carriers can very efficiently be

transferred to the gate before recombination occurrs. The electric field around the gate is generated by injecting an initial charge on the gate. By surrounding the whole memory structure with a conductive wall, effectively an ion chamber is formed between the wall and the floating gate (Figure 1). For photon radiation, the interactions take place mainly in the wall and the secondary electrons ionize the air between the wall and the floating gate. For other charged particle radiation, if the wall is sufficiently thin, the charged particles are allowed to transfer their energy directly into the air space. The dosimetric characteristics can therefore be adjusted by altering the properties of the wall material and other fill-gases instead of air could also be used[1].

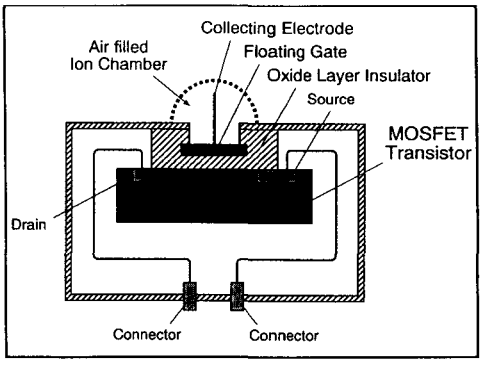

Figure 1. Schematic of a dosemeter based on DIS

Figure 2. The complete electronic Dosemeter

IONIZATION CHAMBER DESIGN

The operating voltage of an ionization chamber based on DIS is 25 to 30 Volt. To obtain an electric field strength in the order 10 kVm^{-1}, necessary for linearity up to high dose rates, the distance between electrodes has to be 2 to 3 mm. The extremely high sensitivity of the DIS methods allows for the use of chamber volumes of 1 cm^3 and less with a resulting detection limit well below 10 mSv. The size of a detector containing two sets of chambers, one to determine $H_p(10)$ and one to determine $H_p(0.07)$ is 35 mm x 50 mm x 7 mm. This detector may be used as a passive dosemeter or it can be combined with an on line readout device to build an electronic dosemeter[2] (Figure 2).

PRELIMINARY DATA ON DETECTOR CHARACTERISTICS

Photon energy dependence

The photon energy dependence of a prototype ionization chamber was tested on the ISO water slab phantom (30 cm x 30 cm x 15 cm, PMMA walls, water filling) using ISO narrow series (N) photon radiation qualities and the most recent conversion coefficients $h_{pK}(0.07)$ and $h_{pK}(10)$ proposed by ISO. The results show that after further optimization of the detector construction an excellent photon energy dependence down to 15 keV for $H_p(10)$ detectors and to at least 6-8 keV for $H_p(0.07)$ detectors can be expected (Figure 3).

Sensitivity to beta radiation

Tests to determine the sensitivity for beta radiation have been performed with ^{204}Tl irradiations of a NPL beta secondary standard ($E_{\beta_{max}}$ = 0.8 MeV). The response for $H_p(0.07)$ varied between 0.47 and 0.75 for different entrance window designs.

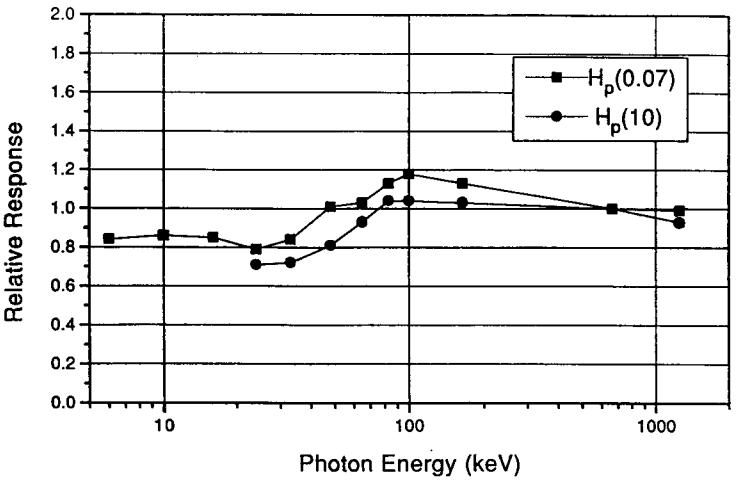

Figure 3. Photon energy dependence of a prototype DIS-dosemeter for $H_p(10)$ and $H_p(0.07)$.

Dose and dose rate range

Linearity of the dose rate has been tested up to 10 Sv/h with deviations of less than 10 %. The dose range can only be fully tested with a complete unit with a final design of the electronics. The intention is to reach a resolution for low dose values of 1 μSv and an upper range of 5 Sv.

CONCLUSIONS

The DIS based dosimetry system is quite simple but highly sensitive. It covers a wide range of photon and beta radiation energies and can be applied as a passive or an electronic device to measure $H_p(10)$ and $H_p(0.07)$. The system seems not to be sensitive to electromagnetic fields.

The tests were performed on prototype instruments, therefore, all test results obtained so far are preliminary. Further optimization of the detector design is possible. The system has a potential to fulfil the requirements of a legal electronic dosemeter. Applications in neutron and radon dosimetry as well as in other fields will be studied.

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A photon spectrometer to investigate the energy spectrum and the angular distribution of photon radiation incident on a personal dosemeter

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Introduction

The design of a personal dosemeter depends on the energy and the direction of the radiation to be measured by the dosemeter. The wider the energy range and the larger the solid angle of the incident radiation, the more complicated the dosemeter must be to maintain the same measurement accuracy. The angular distribution of the radiation incident on the dosemeter is strongly affected by the motion of the dosemeter's wearer in the radiation field. Measurements of the energy spectrum and of the angular distribution of photon radiation incident on personal dosemeters actually worn at workplaces are significant only if the measuring instrument follows these motions, i. e. if it is worn by the person in the radiation field. To our knowledge, such measurements have not, however, been carried out yet. To make such measurements possible, a portable photon spectrometer has been developed.

Design and calibration of the portable photon spectrometer

For a portable photon spectrometer to be worn together with a personal dosemeter by a person working in a radiation field it is most important to be sufficiently small and light in order not to handicap its wearer during his work. Hemispherical CdTe detectors with an active volume of about 50 mm^3 and an integrated charge-sensitive preamplifier were therefore chosen as detectors. These detectors are sufficiently small, light and robust and can be operated without any cooling, and their sensitivity and energy resolution are sufficient for this application (1). The photon spectrometer contains a stack of three hemispherical CdTe detectors (see fig. 1(a)). Each detector is surrounded by a cylindrical shielding of lead and copper to allow the photon radiation incident on the wearer of the spectrometer to be measured from three different cylindrical segments of the solid angle.

Fig. 1(b) shows the measured angular dependence of the response of the photon spectrometer for the ISO N-80 X-ray field (operating voltage of the X-ray tube: 80 kV, mean photon energy: 65 keV). The three angular ranges of the three different detectors, which do not overlap, can be clearly distinguished. Due to decreasing absorption by the shielding material, the separation of the angular ranges of the detectors becomes less pronounced with increasing photon energy.

As the radiation fields encountered at workplaces can be expected to consist mainly of continuous contributions such as scattered radiation or radiation transmitted through shieldings, the spectral distributions are rather broad and an unfolding procedure is necessary to obtain the spectral fluence from the measured pulse-height spectra. It is therefore necessary to determine the detector's response matrix including the effects originating from the electronic devices arranged behind.

The response matrix of the CdTe detectors was determined experimentally (2). For this purpose, the pulse-height spectra of 11 isotopes emitting monoenergetic gamma radiation were measured. The resulting spectra represent the response of the CdTe detectors to monoenergetic photons of the respective energy. These spectra were approximated by a model

function which makes use of 19 energy-dependent parameters. The interpolation of these parameters with respect to the energy and the calculation of the corresponding pulse-height spectra yields the response matrix. The response matrix and the unfolding procedure was tested by comparing measured and unfolded pulse-height spectra of different isotopes with the corresponding reference data.

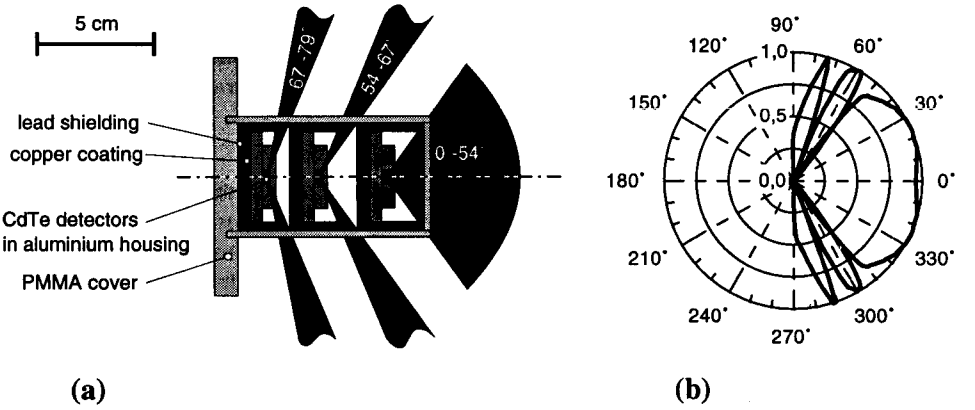

Figure 1. *(a) Setup of the portable photon spectrometer. The figure shows the arrangement of the CdTe detectors with their cylindrical lead shieldings which are coated on the inside with copper to suppress the lead's fluorescence radiation. The spectrometer is covered with PMMA. The whole assembly is of cylindrical symmetry. The dark grey areas indicate the angular range covered by the different detectors. The ratio of the sizes of the segments of the solid angle is 2 (0°-54°) : 1 (54°-67°) : 1 (67°-79°).*
(b) Angular dependence of the response of the photon spectrometer in the angular range from 0° to 360° for the ISO N-80 X-ray field (3).

Results

First measurements at real workplaces were carried out in the Physikalisch-Technische Bundesanstalt during the type testing of three different X-ray emitters, namely X-ray tubes with maximum operating voltages of 160 kV, 225 kV and 300 kV. Fig. 2 shows the measured angular distributions of the personal dose equivalent $H_p(10)$ for the three X-ray tubes. In the measurements with the X-ray tubes with maximum operating voltages of 225 kV and 300 kV, the contributions to the personal dose equivalent $H_p(10)$ of radiation incident from the different segments of the solid angle roughly correspond to the size of these segments, indicating an almost isotropic angular distribution of the incident photon radiation. In contrast, the measurement with the X-ray tube with a maximum operating voltage of 160 kV shows a pronounced maximum at angles of incidence ranging from 54° to 67°.

Fig. 3 shows the unfolded spectral distributions of the personal dose equivalent $H_p(10)$ corresponding to the angular distributions shown in fig. 2. For the X-ray tube with a maximum operating voltage of 160 kV, the main contribution to the total dose originates from the radiation transmitted through the housing of the emitter, whereas for the X-ray tubes with maximum operating voltages of 225 kV and 300 kV also scattered radiation and, especially for the X-ray tube with 300 kV maximum operating voltage, the primary bremsstrahlung considerably contribute to the total dose. The measurements agree in showing that the whole energy range below the maximum energy determined by the maximum operating voltage and all ranges of the solid angle, also those with very oblique angles of incidence, contribute significantly to the personal dose equivalent $H_p(10)$.

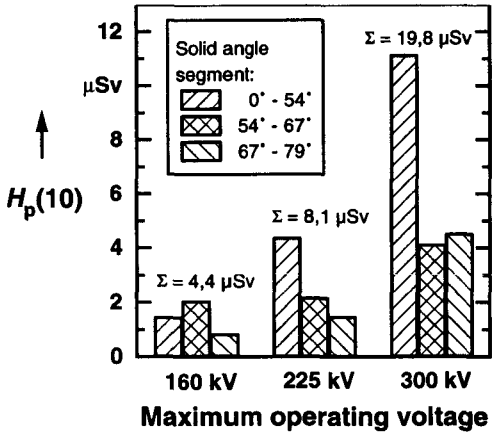

Figure 2. Angular distributions of the personal dose equivalent $H_p(10)$ during the type testing of three different X-ray emitters (X-ray tubes with maximum operating voltages of 160 kV, 225 kV and 300 kV). Σ denotes the total dose measured during the respective type tests.

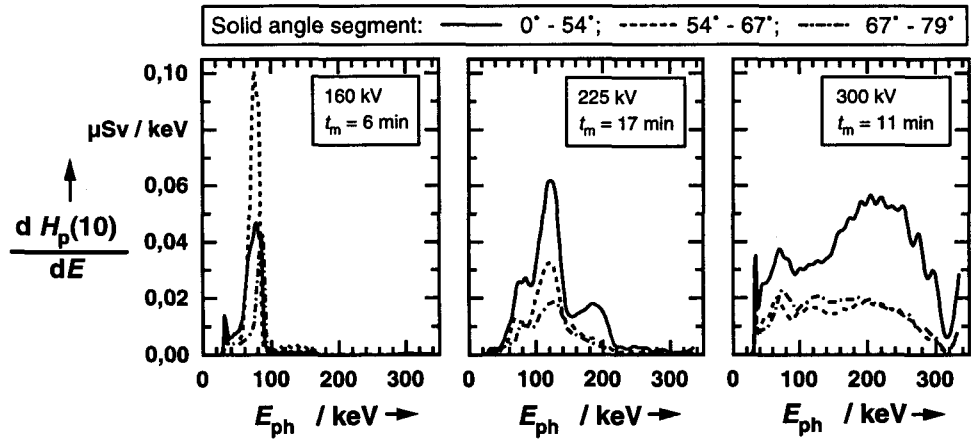

Figure 3. Unfolded spectral distributions of the personal dose equivalent $H_p(10)$ during the type testing (duration of measurement t_m) of three different X-ray emitters (X-ray tubes with maximum operating voltages of 160 kV, 225 kV and 300 kV). E_{ph}: photon energy.

Acknowledgement

This work was supported by the Minister of the Environment, Nature Conservation and Reactor Safety of the Federal Republic of Germany. The views and opinions expressed herein are those of the authors and do not necessarily represent those of the Minister.

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A new film badge for the measurement of the personal dose equivalent $H_p(10)$ using the gliding-shadow method

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Introduction

The gliding-shadow method (1) is a novel design principle for filters of a personal dosemeter badge which allows the energy and angular dependence of the dosemeter response to be considerably reduced. This is demonstrated with the aid of a new film dosemeter, because film dosemeters are used worldwide in routine individual monitoring predominantly because of their advantages in dose record documentation and in the retrospective investigation of the conditions of exposures. The new film badge was developed and optimized for the new measurand $H_p(10)$.

The gliding-shadow method

In the new film badge, filters are used which have been designed and optimized using the gliding-shadow method. With the gliding-shadow method a filter F of the size of detector D is placed at a certain distance d in front of the detector (see fig. 1). In the case of normal incident radiation, the shadow of the filter exactly hits the detector (fig. 1, on the left). If the direction of radiation incidence changes to oblique angles, the shadow of the filter glides apart and parts of the detector are exposed to unattenuated radiation (fig. 1, middle). The more oblique the angle of incidence becomes, the more the shadow glides beside the detector (fig. 1, on the right). Obviously, the response of the detector increases with increasing angle of incidence, depending on the distance and the thickness of the filter. The variation in the response of a detector shielded in this way is contrary to the variation of the response of detectors shielded with conventional filters, where the filter is much larger than the detector and very close to it ($d \cong 0$). Consequently, the combination of a conventional filter with a filter designed by the gliding-shadow method enables the construction of a personal dosemeter with a low variation of the response even at large angles of incidence over a wide energy range.

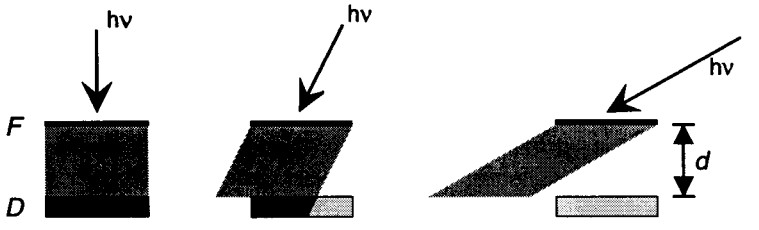

Figure 1. *Principle of the gliding-shadow method: with increasing oblique angle of incidence, the shadow of the filter F glides beside detector D depending on the distance d between filter and detector.*

Design of the new film badge

Due to the strong energy dependence of the response of the bare film, a satisfactorily low variation of the response cannot be achieved with a single filter, even if the gliding-shadow

method is used. It has, however, been found that two filters, a metal filter and a plastic filter, are sufficient. Fig. 2 shows the setup of the two optimized filters. The filters are of circular symmetry in relation to a vertical axis through the center of the respective filter and are designed for a diameter of 8 mm of the detector which is the area on the film directly below the filter. The metal filter is made up of two concentric discs, each consisting of an Sn and Pb layer. A PE disc serves as a spacer between both discs. The plastic filter consists of a thin PE layer (0,5 mm) with a PE ring 2 mm thick arranged below. Thus the gliding shadow is replaced by a gliding illuminated area. This design is necessary, because, at low energies, the bare film itself has nearly the required response to $H_p(10)$ at normal incidence, and only at oblique incidence is an additional shielding necessary to simulate the additional attenuation due to the longer path through the 10 mm ICRU tissue layer in the definition of $H_p(10)$.

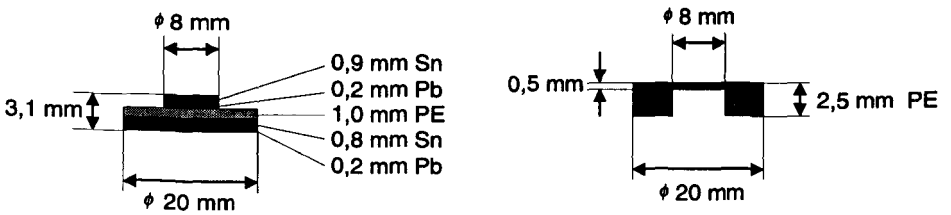

***Figure 2.** Design of the metal filter (on the left) and of the plastic filter (on the right) in the new film badge. A set of identical filters is placed below the film.*

Results and conclusions

The performance of the dosemeter was tested using X-ray fields of the ISO narrow spectrum series with mean photon energies ranging from 17 keV to 250 keV, gamma radiation of ^{137}Cs and ^{60}Co, and high-energy photon fields of 4,4 MeV and 6,7 MeV. The X-ray irradiations were carried out on slab phantoms (size 30 cm × 30 cm × 15 cm) according to ISO/CD 4037-3 at a distance of 250 cm from the focus of the X-ray tube, the irradiations with ^{137}Cs and ^{60}Co at a distance of 100 cm from the gamma source and the irradiations in the high-energy photon fields at a distance of 50 cm from the source. The angle of incidence α ranged from normal incidence (0°) to 75°.

The conventional true value of the personal dose equivalent for the ISO water slab phantom, $H_p(10)$, was calculated from the air collision kerma K_a using appropriate conversion coefficients from ISO/CD 4037-3. The dose values, H_{pla} and H_{met}, were obtained by first measuring the optical density on the film behind the plastic filter and the metal filter as mean values over areas 8 mm in diameter which are fixed irrespective of the angle of incidence. In the second step, these values of the optical density were converted to the dose values using calibration curves and corrections for the background. The calibration curves of H_{pla} were obtained with X-rays with a mean photon energy of 37 keV, and those of H_{met} with gamma radiation of ^{137}Cs.

The measured dose value M_p of the whole dosemeter results from a linear combination (2, 3) of the measured values, H_{pla} and H_{met}:

$$M_p = 1{,}05 \cdot H_{met} + 0{,}82 \cdot H_{pla}$$

In fig. 3 the response, i. e. the ratio $M_p / H_p(10)$, is shown as a function of the mean photon energy and of the angle of incidence. The response shows a maximum deviation of ± 34% (dashed lines) from unity over an energy range from 17 keV (N-20) to 1250 keV (^{60}Co) and

over an angular range from 0° to 60°. For angles of incidence of 75°, most of the measured data are within or slightly outside the range of ± 34% except for energies of 17 keV (N-20) and 48 keV (N-60). In the case of high-energy photon radiation with energies of 4,4 MeV and 6,7 MeV, the dosemeter shows an overresponse of between 70% and 110%.

From the ratio H_{pla} / H_{met} of the measured values, an estimation of the mean energy of the incident photon radiation is possible for energies up to about 300 keV.

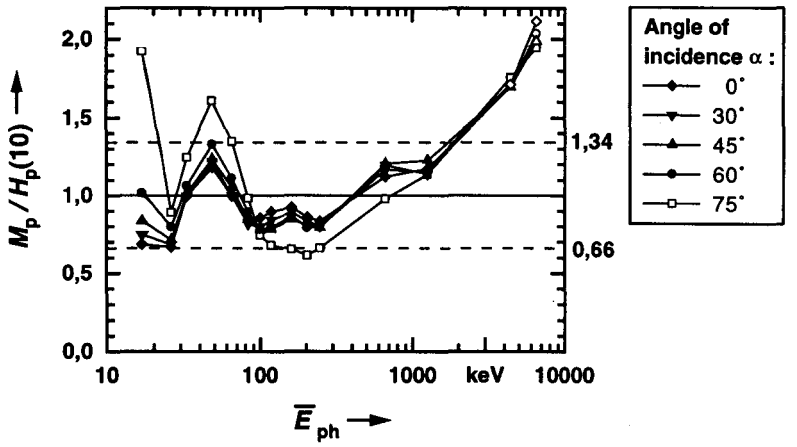

Figure 3. *Response of the new film dosemeter to $H_p(10)$ as a function of the mean photon energy $\overline{E}_{ph}$ for different angles of incidence α. The dashed lines indicate a deviation of the response of ± 34% from unity. The open symbols denote irradiation conditions outside the suggested rated range of use, which is 17 keV - 1250 keV for the energy and 0° - 60° for the angle of incidence.*

All these results can be obtained by a fully automated read-out procedure, because the measuring positions of the densitometer are fixed. The whole dosemeter is enclosed by a lead shielding frame to prevent incorrect measurements due to radiation incident parallel to the film plane. In addition, the badge contains two small filters which are not used to determine the dose but to detect beta radiation, and four lead pins to determine the direction of the incident radiation.

It is also possible to design other dosemeters, e. g. TL-dosemeters, using the new gliding-shadow method, and to calculate other measurands for the personal dose equivalent by exchanging the coefficients for the linear combination.

Acknowledgement

This work was supported by the Minister of the Environment, Nature Conservation and Reactor Safety of the Federal Republic of Germany. The views and opinions expressed herein are those of the authors and do not necessarily represent those of the Minister.

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CONVERSION COEFFICIENTS RELATING AIR KERMA TO $H_p(10)$ AND $H_p(0.07)$ FOR ISO X RAY NARROW SPECTRUM SERIES USING A PMMA SLAB PHANTOM

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INTRODUCTION

In 1985, ICRU (1) introduced two operational dose equivalent quantities, $H_p(d)$ and $H_s(d)$, for individual monitoring, which were recently (2) substituted by a single quantity $H_p(d)$ called the personal dose equivalent. These quantities are defined as the dose equivalent in soft tissue below a specified point on the body at depth d. Since its introduction, the implementation of these quantities in practical calibration procedures has encountered many difficulties, mainly because of the impossibility of reproducing them in a laboratory (3). To overcome this problem, ICRU, in its Report n° 47 (2), recommends the use of a 30 cm x 30 cm x 15 cm depth PMMA phantom in dosimeter calibrations. Some other phantoms are also accepted, thus the PMMA slab one is recommended to achieve uniformity in calibration procedures. To be able to determine the individual monitoring operational quantities, the quantity to be used for calibration purposes is defined as the dose equivalent in a phantom having the composition of ICRU tissue and the same size and shape as the calibration phantom. Consequently, conversion factors relating air kerma and photon fluence to personal dose equivalent $H_p(10)$ for normal photon incidence on a 30 cm x 30 cm x 15 cm depth slab phantom of ICRU tissue are given in ICRU 47. Corresponding values for $H_p(0.07)$ are given by Grosswendt (4).

Although ICRU 47 has clarified and simplified personal dosimeter calibration procedures, two main aspects remain unsolved. First of all, the report does not provide conversion coefficients for filtered X radiation specified by ISO 4037 and its amendment (5), which are very often used in personal dosimetry calibrations. Moreover, no correction is considered for backscatter differences in PMMA and ICRU tissue. The aim of this study is to obtain conversion coefficients relating air kerma to $H_p(10)$ and $H_p(0.07)$ for ISO X ray narrow spectrum series using an ICRU slab phantom and to propose a backscatter correction factor for ISO spectra when calibration is performed with the recommended PMMA slab phantom.

CALCULATION PROCEDURES

Conversion coefficients relating air kerma in free air to personal dose equivalent for ISO X ray narrow spectrum series have been calculated by means of equation 1, integrating the conversion coefficients for monoenergetic photons (4) over the X-ray spectra given by Seelentag et al (6), corresponding to the ISO narrow spectra.

$$\overline{F}_K = \frac{\int_0^{E_{max}} K_{air,E}\, F_K(E)\, dE}{\int_0^{E_{max}} K_{air,E}\, dE} \qquad (1)$$

where $K_{air,E}$ is the energy spectral distribution of air kerma, namely, $\frac{dK_{air}}{dE}$, E_{max} is the maximum energy of the considered spectrum and $F_k(E)$ is the conversion coefficient for energy E.

To simplify the calculation, the coefficients for monoenergetic photons $F_k(E)$ have been fitted to two analytical functions similar to the ones proposed by Wagner (7), for coefficients in the ICRU sphere. $F_k(10)$

and $F_k(0,07)$ for ISO X ray spectra are shown in table 1.

Calculations have also been performed using the spectral distribution of air kerma from experimental X ray spectra obtained in our laboratory and the differences with the values stated in table 1 were lower than 1 %.

Generating Voltage (kV)	GSF spectrum n°	Mean energy (keV)	$\overline{F}_K(10)$ (Sv/Gy)	$\overline{F}_K(0,07)$ (Sv/Gy)	k'_p
40	37	32,5	1,174	1,265	0,925
60	54	47,3	1,650	1,551	0,922
80	65	64,5	1,884	1,718	0,932
100	75	82,6	1,868	1,709	0,943
120	83	100	1,797	1,663	0,954
150	98	117	1,713	1,605	0,959
200	110	164	1,562	1,484	0,966
250	126	207	1,478	1,417	0,975
300	135	248	1,422	1,375	0,979

Table 1. Conversion coefficients and backscatter correction factors for calibrations using a 30 cm x 30 cm x 15 cm depth PMMA phantom for the X ray narrow ISO spectra series from reference (6).

A PMMA phantom produces a different scatter at its surface than an ICRU phantom. To estimate this difference the ratio between the backscatter phantom of both materials is calculated as

$$k'_p = \frac{B_m^{ICRU}}{B_m^{PMMA}} = \frac{(K_m)_{ICRU}}{(K_m)_{PMMA}} \qquad (2)$$

where k'_p is the backscatter correction factor which quantifies the difference of kerma in the dosimeter material m if it is situated on the surface of a PMMA slab or an ICRU slab. We consider that this factor should not be disregarded and that the dosimeter reading should be adequately corrected. Variations of k'_p in air and tissue are within 1%, and since most detectors are good substitutes of air or tissue k'_p is obtained as

$$k'_p = \frac{(K_{ICRU})_{ICRU}}{(K_{ICRU})_{PMMA}} \qquad (3)$$

In the energy range considered in this study, kerma ratio indicated in equation (3) can be substituted by air collision kerma ratio which in turn can be approximated to dose equivalent at 0 mm depth on the PMMA and ICRU tissue slab. k'_p for monoenergetic photon has been obtained from Grosswendt data (4) as the ratio of dose equivalent in the two phantoms. k'_p varies with energy, therefore, as it was done with $F_k(d)$, a mean backscatter correction factor is calculated for the ISO X ray narrow spectrum series as

$$\overline{k'}_p = \frac{\int_0^{E_{max}} K_{air,E}\, k'_p dE}{\int_0^{E_{max}} K_{air,E}\, dE} \qquad (4)$$

Calculated values for Seelentag spectra (5) are shown in Table 1. Correction factor can be seen to range between 2 % and 8 %, which we think is not negligible.

Another problem that arises with the use of personal dose equivalent quantities is the difficulty to assign an uncertainty to the conversion coefficients; obviously this is not suitable for a calibration procedure. To achieve the desired good uniformity, some agreed conversion coefficients should be established by international organisations such as ICRU or ISO and the associated uncertainty evaluated comparing different authors' calculations or when possible comparing experiment and calculation. At the moment, little information is available concerning this (8,9).

CONCLUSIONS

From these results it can be stated that the use of a backscatter correction factor improves uniformity between calibration procedures which is very desirable despite the uncertainty inherent in personal dosimetry. Consequently when a PMMA slab phantom is used for calibration the personal dose assigned to the dosimeter should be

$$\frac{H_p(d)}{k'_p} = \frac{F_k(d) \cdot K_{air}}{k'_p} \qquad (5)$$

and for calibration beams,

$$\frac{H_p(d)}{\overline{k'_p}} = \frac{\overline{F_k(d)} \cdot K_{air}}{\overline{k'_p}} \qquad (6)$$

Since ICRU recommends a practical phantom such as the PMMA slab phantom, we believe that calibration procedures in this field should make this the standard phantom. If this was so it would be much easier to establish a set of internationally-agreed conversion coefficients for all ISO calibration qualities and to reasonably estimate the associated uncertainty.

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The evaluation of multi-element personal dosemeters using the linear programming method

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Introduction

Multi-element dosemeters are frequently used in individual monitoring. Each element can be regarded as an individual dosemeter with its own individual dose measurement value. In general, the individual dose values of one dosemeter vary according to the exposure conditions, i. e. the energy and angle of incidence of the radiation. The (final) dose measurement value of the personal dosemeter is calculated from the individual dose values by means of an evaluation algorithm. The best possible dose value, i. e. that of the smallest systematic (type B) uncertainty if the exposure conditions are changed in the dosemeter's rated range of use, is obtained by the method of linear programming.

Mathematical basis

The mathematical problem of determining the measured dose value H of the entire dosemeter from the individual dose values of each detector element $\{D_k,\ k = 1, \cdots, m\}$ has been described by Bermann and Chanourdie [1] in the form of a set of linear algebraic equations:

$$D_k = \sum_{i=1}^{n} a_{i,k} \cdot H_i \qquad k = 1, \cdots, m \tag{1}$$

$$H = \sum_{i=1}^{n} H_i \tag{2}$$

Here n denotes the number of the different radiation fields used for the calibration, and m denotes the number of the individual detector elements in the dosemeter. In the evaluation of the dose value H, m individual dose values $\{D_k,\ k = 1, \cdots, m\}$ are therefore measured. The response matrix $a_{i,k}$ contains the response of the m individual detector elements of the dosemeter to the n different radiation fields used for calibration. In the evaluation of H it is assumed that the dosemeter has been exposed to a radiation field which can be described as a linear combination of these n calibration fields with dose contributions H_i, see eq. (2).

The solution of a set of linear algebraic equations of the form of eq. (1) and (2), i. e. here the determination of a set of dose contributions $\{H_i,\ i = 1, \cdots, n\}$, is completely covered by the theory of linear programming and was described in [2] including the following two problems:

- Frequently the set of equations (1) cannot be solved for a set of measured individual dose values.
- If the set of equations (1) can be solved, usually the solution is not a unique one. In this case, the dose values H, see eq. (2), form an interval:

  $$H_{min} \leq H \leq H_{max} \tag{3}$$

  with all values of H within the interval being solutions of the set of equations (1) and (2). This problem has been described in [3].

At present, in radiation protection the dose evaluation algorithms listed in table 1 are applied to determine the dose value H from the individual dose values $\{D_k,\ k = 1, \cdots, m\}$. Each of these evaluation algorithms can be interpreted as an attempt at finding a solution of the set of

equations (1) and (2). The algorithms differ with respect to the information made available concerning the uncertainty of the dose value H and concerning the dose contributions $\{H_i, i = 1, \cdots, n\}$ to the dose value H, i. e. the spectrum of the radiation field.

***Table 1.** Additional information resulting from the various dose evaluation algorithms.*

Evaluation method	Additional information about:	
	uncertainty	spectrum
(A) Methods using ratios of the individual values	No	Yes
(B) Linear combination method	Yes	No
(C) Linear programming method	Yes	Yes

At least in Germany, above all the algorithm (A) is applied at present. To support the dose evaluation by algorithms (B) and (C) a program named »LINOP« and the corresponding manual were developed [4]. The program allows both the determination of the optimum coefficients for the linear combination method and the dose evaluations by linear programming using the simplex method to be carried out in the routine monitoring of a dosimetry service.

In the following, the evaluation algorithms mentioned previously are described in more detail. The results of a test of »LINOP« by corresponding measurements are presented.

Methods using ratios of the individual values

By the methods using ratios of the individual values (A), one single dose value compatible with the set of equations (1) and (2) and the corresponding spectrum are determined. The set of equations is not completely solved, i. e. the interval $[H_{min}, H_{max}]$ is not determined. The uncertainty of the measured dose value must therefore be assumed to be of the size found in performance tests carried out with well-defined radiation fields and well-defined irradiation conditions. The actual uncertainty cannot, however, be determined by these methods.

Linear combination method

By the linear combination method (B), the set of equations (1) and (2) is solved neglecting the information concerning the spectrum, which are contained in the individual dose values:

$$H = \sum_{k=1}^{m} \sigma_k \cdot D_k \qquad (4)$$

The optimum coefficients σ_k minimizing the uncertainty of measurement of this method are obtained by linear programming [2, 4] using eq. (1) and (2) together with the minimization of the maximum uncertainty of measurement of all calibration measurements. The uncertainty of measurement to be specified in the evaluation of the dose value is independent of the individual dose values $\{D_k, k = 1, \cdots, m\}$ of a single measurement, but is equal to the maximum interval $[H_{min}, H_{max}]$ resulting from all individual dose values of all calibration measurements. Neglection of the information about the spectrum therefore leads to an increasing uncertainty of measurement.

Linear programming method

For every measurement the method of linear programming (C) [2, 4] furnishes the set of dose values compatible with the set of equations (1) and (2), i. e. the dose interval $[H_{min}, H_{max}]$ depending on the individual dose values $\{D_k, k = 1, \cdots, m\}$ of the respective measurement. The spectra corresponding to H_{min} and H_{max} can be specified. This method is applicable to

every measurand and to every multi-element dosemeter and, in every case, yields the minimum uncertainty of measurement.

The results of the »LINOP« program were verified by measurements with a four-element laser TLD dosemeter [5] (see table 2). The response matrix $a_{i,k}$ of this TL-dosemeter consists of the responses of the four single elements to X-ray radiation of the ISO narrow spectrum series (N-30 to N-300) and to gamma radiation of ^{137}Cs and ^{60}Co at angles of 0°, 30°, 45° and 60° on an ISO water slab phantom. The measured value M_p of the personal dose equivalent is the geometric mean of H_{min} and H_{max}. The maximum possible dose interval $[H_{min}, H_{max}]$ was calculated by »LINOP«. The calculation of the maximum dose interval not only furnishes the corresponding calculated individual dose values D_k^c of the four single elements but also the irradiation conditions, i. e. the dose contributions H_i of the radiation qualities Q_r at the angles of incidence α, which lead to these individual dose values D_k^c. These irradiation conditions are calculated for $H = H_{min}$ and $H = H_{max}$, H being the dose to which the dosemeter is exposed. Irradiation of the dosemeter in a radiation field satisfying these calculated irradiation conditions and subsequent evaluation of the dosemeter lead to the measured individual dose values D_k^m which are slightly different from the calculated dose values D_k^c which then result in a slightly different dose interval $[H_{min}, H_{max}]$.

The comparison of the calculated and measured individual dose values and the resulting dose interval listed in table 2 yields good agreement between calculation and experiment. The calculated and measured data differ only slightly within the uncertainty which can be achieved with the TL-detectors used.

Table 2. *Comparison between calculated irradiation data and the corresponding measured irradiation data, both obtained by linear programming. For the symbols, see text.*

	Irradiation conditions					Individual dose values in mSv				Dose in mSv			
$H=$	Q_r	α	H_i in mSv	H in mSv		D_1	D_2	D_3	D_4	H_{min}	H_{max}	M_p	$\frac{M_p}{H}$
H_{min}	N-30	45°	0,08	2,00	calc.	6,70	2,29	4,77	6,50	2,00	2,69	2,32	1,16
	N-150	60°	1,92		meas.	6,72	2,15	4,48	6,52	1,90	2,56	2,21	1,11
H_{max}	^{60}Co	0°	2,21	2,69	calc.	6,70	2,29	4,77	6,50	2,00	2,69	2,32	0,86
	N-60	30°	0,48		meas.	6,60	2,30	4,66	6,57	2,08	2,73	2,38	0,88

Acknowledgement

This work was supported by the Minister of the Environment, Nature Conservation and Reactor Safety of the Federal Republic of Germany. The views and opinions expressed herein are those of the authors and do not necessarily represent those of the Minister.

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DEVELOPMENT OF NEW METHODOLOGY FOR DOSE CALCULATION IN PHOTOGRAPHIC DOSIMETRY

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INTRODUCTION

The personal dosemeter system of IPEN is based on film dosimetry. Personal doses at IPEN are mainly due to X or gamma radiation.

The use of personal photographic dosemeters involves two steps: firstly, data acquisition including their evaluation with respect to the calibration quantity and secondly, the interpretation of the data in terms of effective dose .

The effective dose was calculated using artificial intelligence techniques by means of neural network. The learning of the neural network was performed by taking the readings of optical density as a function of incident energy and exposure from the calibration curve (1). The obtained output in the daily grind is the mean effective energy and the effective dose .

MATERIALS AND METHODS

The photographic dosemeters used at IPEN are made up of two parts:
a) the Personal Monitoring 2/10 film (Agfa - Gevaert), consisting of two emulsions. b) the holder (badge) containing four filters (plastic, lead, copper and cadmium) and an open window. The area of the filters is sufficiently large to avoid edge effects.

The irradiations were performed using a ^{60}Co (1.0 GBq) ,^{137}Cs(53 TBq) and a X-ray machine Stabilipan model 250 Siemens.The quality control was carried out using a ^{14}C plane source.

The measurement of radiographic film density was performed using a transmission densitometer MacBeth TD 904.

The calibration curves (OD x Exposure) were ploted out as a function of the radiation energy for each filter of the badge. In this case , for each lecture **j** , due to a certain filter, it is possible to construct a family of response curves of OD (d_{ij}) as a function of Exposure (X_i) for each energy value (2).

NEURAL NETWORK STRUCTURE

Different structures of neural networks can be used, each one associated with one application. In this work it was used the backpropagation paradigma, were the neuronios are arranjed in layers: input, output and intermediate layers as shown in Fig. 1 In this work it was used the Neural Works Professional II/Plus software (3,4).

All values used for the network training are the average of five lectures.

For the training of the network the data were arranjed as a matrix as shown bellow . The matrix was presented to the network 10^6 times, during 30h in a PC-AT 486. The obtained average square error was less than 0.1%.

Different from the training, the use of the trained network is very fast, only a fraction of seconds (5).

In order to apply the trained network it was developed a software in Visual Basic Language for Windows.

$$\begin{bmatrix} DO-Cd_1 & DO-Ja_1 & DO-Cu_1 & DO-Pb_1 & DO-Pl_1 & E_1 & D_1 \\ \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot \\ \cdot & \cdot & \cdot & \cdot & \cdot & \cdot & \cdot \\ DO-Cd_i & DO-Ja_i & DO-Cu_i & DO-Pb_i & DO-Pl_i & E_i & Di \end{bmatrix}, \quad i = 179$$

IMPROVEMENT OF A BETA/GAMMA DOSE RATE METER FOR MEASURING $\dot{H}^*(10)$ AND $\dot{H}'(0.07)$

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The portable dose rate meter DL 1 was developed at the Research Centre Jülich approx. 14 years ago for measuring the area dose rate in mixed β-γ-radiation fields. This instrument with a small ionization chamber is particulary well suited for measurements near contaminated surfaces where the contribution of β-radiation to the dose rate can be very high.
An adaption of the device is required for measuring the new operational quantities $\dot{H}'(0.07)$ and $\dot{H}^*(10)$. Ionization chambers of different configuration were tested. By a suitable choice of the chamber form and wall material it is possible to measure $\dot{H}'(0.07)$ for β-radiation within $\pm$ 20 % for energies from 0.23 to 2.2 MeV and up to angles of incidence of 60°. $\dot{H}^*(10)$ is determined within $\pm$ 20 % for photons with energies > 50 keV. Furthermore, the microphonic effect was reduced.

INTRODUCTION

In handling unsealed radioactive sources, the skin dose due to the β radiation can exceed many times that caused by γ radiation. For measurements in mixed β-γ radiation fields, the portable β-γ dose rate meter DL 1 was developed at the Research Centre Jülich approximately 14 years ago [1]. Its detector is a small (15 cm^3) thin-walled ionization chamber suitable for measuring the dose rate especially near contaminated surfaces. For β radiation the dose rate is determined at 70 μm tissue depth. An additional build-up cap serves to measure the photon dose equivalent rate for γ radiation.
Due to the planned introduction of the new operational quantities *directional dose equivalent* $H'(0.07)$ and *ambient dose equivalent* $H^*(10)$ [2] it is necessary to modify the above dose rate meter. Apart from an energy-independent response with respect to either $\dot{H}'(0.07)$ or $\dot{H}^*(10)$ particular attention was paid to a response independent of the direction of radiation incidence. Whereas in the past an isotropic response was also demanded for β radiation fields, investigations into the new operational quantities revealed a quite pronounced dependence of the dose rate on the angle of incidence [3]. Depending on the energy of β radiation, both higher and lower values can be obtained for the dose rate in the case of inclined radiation incidence relative to vertical incidence (e.g. the response is 15 % higher for an angle of incidence of 60° than of 0° in case of $^{90}Sr/^{90}Y$ β radiation).
Based on the existing measurement system, new chamber configurations were designed according to basic physical and geometrical principles, constructed and subsequently tested in β and γ radiation fields. The response with respect to $\dot{H}'(0.07)$ and $\dot{H}^*(10)$ was optimized by successively improving the configuration.

EXPERIMENTAL

Work was aimed at determining the operational quantity $\dot{H}'(0.07)$ within $\pm$ 20 % for maximum β energies between 0.2 MeV and 2 MeV and for angles of incidence up to 60° [3]. An isotropic response with respect to $\dot{H}^*(10)$ within $\pm$ 20 % was also demanded for photon radiation with energies > 60 keV. Investigations concerning the response with respect to $\dot{H}'(0.07)$ for β radiation were carried out using the β secondary standard [4] with three different β emitters (^{147}Pm, 0.23 MeV; ^{204}Tl, 0.765 MeV; $^{90}Sr/^{90}Y$, 2.2 MeV). $\dot{H}^*(10)$ was measured at an X-ray unit with filtration for radiation quality A [5]. The reference for all data specified is the response to ^{137}Cs γ radiation (662 keV).
Our investigations were based on the existing dose rate meter DL 1, at which the different chamber

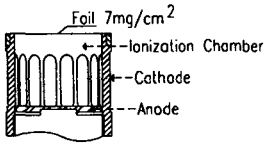

Fig. 1a DL 1 with original ionization chamber

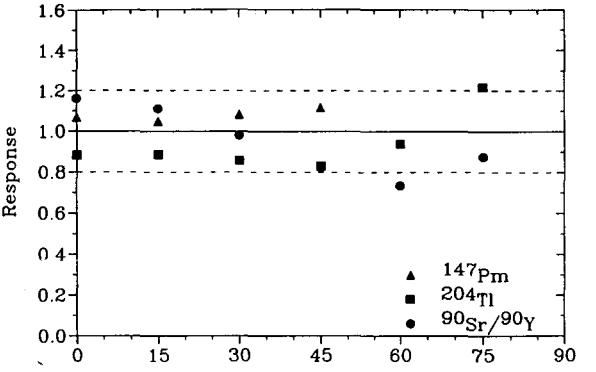

Fig. 1b

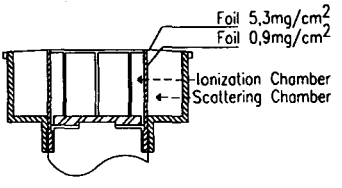

Fig. 2a Configuration with additional scattering chamber

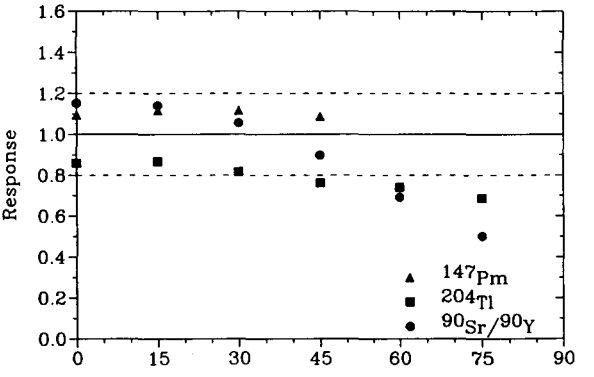

Fig. 2b

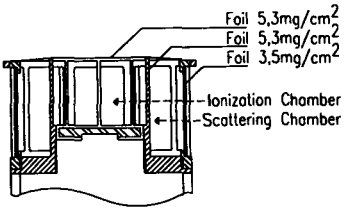

Fig. 3a Final chamber configuration

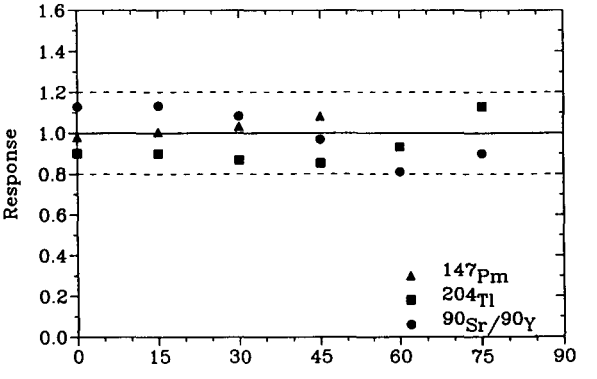

Fig. 3b

Fig. 1-3 Different chamber configurations (a) and related response with respect to Ḣ'(0.07) (b) as a function of radiation incidence

types were tested. The range of the instrument extends from 1 mSv/h (full scale) to 10 Sv/h in five measuring ranges. This also applies to the chamber types described below since the active measuring volume was kept approximately constant.

The original DL 1 ioniziation chamber (Fig. 1a) is formed by an aluminium cage with lateral windows and one end window, which are covered by an electrically conductive plastic foil with a mass/unit area of 7 mg/cm^2. In this chamber, as well as in the other types, the outer wall of the measurement chamber forms the cathode and a graphite disk of 25 mm diameter the collecting anode. The response with respect to $\dot{H}'(0.07)$ of this chamber is shown in Fig. 1b. It displays an inadmissibly strong dependence especially on the angle of incidence. Various experiments with a changed geometry of the measurement chamber as well as variations of the foil thickness did not initially yield any significant improvement.

In order to achieve a better adaptation of the response with respect to $\dot{H}'(0.07)$, it was attempted to adapt the measuring principle to real exposure conditions, where laterally scattered radiation in the upper layer of the skin increases the dose rate in the radiation-sensitive basal layer of the epidermis. On the basis of these considerations, the original chamber was first enclosed in a very large additional scattering chamber (19 cm diameter). The optimum front wall proved to be a plastic foil with a mass/unit area of 7 mg/cm^2. This configuration enabled $\dot{H}'(0.07)$ to be measured within $\pm$ 15 % for β radiation independent of energy up to an angle of incidence of 75°. Due to the size of the scattering chamber, however, this configuration is not suitable for practical application.

In a next step, therefore, the diameter of the scattering chamber was reduced (Fig. 2a). Shading of the bulk lateral walls deteriorated the response for larger angles of incidence (Fig. 2b).

A reduction in height of the lateral walls of the scattering chamber can reduce this shading effect. It proved to be difficult, however, to cover the scattering chamber with a foil, since it was practically impossible to avoid wrinkling.

This problem can be solved by a scattering chamber with high lateral walls with twelve windows that are covered with a plastic foil (Fig. 3a). The energy dependence of the response was optimized for β radiation by varying the individual foil thicknesses (Fig. 3b).

For this chamber configuration, the response to photon radiation as a function of energy was determined using an additional 10 mm thick build-up cap made of PMMA. The required tolerance of $\pm$ 20 % is also maintained here. The dependence on the angle of incidence also shows deviations within $\pm$ 20 % from the target value up to angles of 60°.

As an additional effect, the measurement chamber is protected by the scattering chamber against vibration due to sudden air pressure fluctuations. This leads to a considerable reduction of microphonic effects.

CONCLUSIONS

An additional scattering chamber around the measurement chamber allows the determination of $\dot{H}'(0.07)$ within $\pm$ 20 % for β radiation with energies from 0.23 MeV to 2.2 MeV and for angles of incidence up to 60°. A build-up cap ensures that by measuring $\dot{H}^*(10)$ the response to photon radiation is also constant within $\pm$ 20 % for energies > 50 keV.

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SILICON STRIP DETECTORS FOR DETERMINATION OF RADIONUCLIDES IN VARIOUS SUBSTANSES

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Abstract. Silicon strip detectors for the coordinate determination of short-range particles can be effectively used for measurements of radionuclide spectra in various substances because of their high energy resolution. In comparison with usual detectors they have much lower noise level at the same value of the sensitive area.

INTRODUCTION

Silicon strip defectors are widely used for the coordinate determination of ionizing particles in high energy physics (1-3). They were also applied for measurements of alpha-particles spectra with high spatial resolution (3-6]). Strip detectors allow to obtain higher energy resolution in comparison with usual large area detectors (for instance , surface barrier detectors) because the capacitance and reverse current of a strip, which determine the energy resolution , are much smaller than those of usual detectors. In the present communication the main attention is paid to measurements of energy spectra of alpha -active nuclides by silicon strip detectors.

DETECTOR FABRICATION

Detectors were fabricated on the base of (111) oriented wafers of high resistivity n-type silicon. The p^+-n junction were made by the ion implantation of boron and rearside n^+ -n junction was obtained by the ion implantation of phosphorus with subsequent aluminium metallization (aluminium layer thickness was 0.3 μm). Passivating SiO_2 layers have thickness 0.15 - 0.20 μm . Some other characteristics of silicon strip detectors are presented in Table 1.

Table 1. Characteristics of fabricated strip detectors.

Detector type	1	2	3	4	5 *)
Resistivity , kOhm·cm	1.0	12.5	4.0	4.0	4.0
Wafer thickness , μm	280	340	330	280	330
Numbers of strips	28	40	128	8	32
Pitch , μm	250	350	330	3100	3100
Strip width, μm	200	300	280	3000	3000
Interstrip distance, μm	50	50	50	100	100
Strip length, cm	2.5	40	42	42	**)
Strip capacitance, pF	1.9	3.7	3.8	50.6	**)
Energy resolution, keV	25	15	4	20	20
Full depletion voltage, *V*	250	29	85	62	85
Testing source	^{226}Ra	^{241} Am	^{106}Ru	^{90}Sr + ^{90}Yr	^{90}Sr + ^{90}Yr

*)the annular strip detectors.
**)Strip length and capacitance are dependent on the strip radius.

RESULTS AND DISCUSSION

The experimental set up and the measurement technique were described in details in ref .(3). Fabricated detectors were tested with alpha sources ^{226} Ra , ^{241} Am and minimum ionizing particles (m.i.p.) from sources ^{106}Ru and ^{90}Sr + ^{90}Yr .

Fig. 1 shows the result of measurements of charges Q_i and Q_{i+1} collected at two neighboring strips. The total charge collected at the strips $Q = Q_i + Q_{i+1} = const(E)$ represents events from particles of the same fixed energy *E*. Charge generated in the detector by short-range particles (alpha particles of ^{226} Ra and ^{241}Am) was collected at one or two strips. Hence, the capacitance and the reverse current for strip detectors are much smaller than those for usual detectors with the same total sensitive area.Four loci in Fig.1 correspond to the following alpha particle energies 7.69, 6.0 , 5.49 and 4.82 MeV (^{226} Ra source) .

Noise of detectors strongly depends on the capacitance and reverse currents of strips. The strip capacitance C at applied voltage V in the case $V < V_{depl}$ (V_{depl} is the full depletion voltage) is equal to (see ref.(7))

$$C = 1.054\ S/d\ ,\ d = (2\varepsilon\mu_e\rho V)^{1/2}\ ,\ V_{depl} = w^2/(2\varepsilon\mu_e\rho)\ ,$$

where $\varepsilon = 1.054\ 10^{-12}$ F cm^{-1}, μ_e is the electron mobility, ρ -is the silicon resistivity, S is the strip area. d - is the depletion layer thickness. The main contribution to the reverse current of a strip gives the generation- recombination current (7)

$$I_{gen} = qn_i\Omega/(2\tau)\ ,$$

where q is the elementary charge, n is the intrinsic concentration, τ is the lifetime, $\Omega = Sd$ is the volume of the depleted region.

Capacitances and reverse currents in strip detectors are in S_{strip}/S_{total} times smaller than in large area detectors , so one can expect a higher energy resolution for strip detectors. However, a real gain was smaller because a limiting factor was the input capacitance of preamplifiers and observed values of reverse currents were larger than current estimated by expression for I_{gen} . This indicates a significant role of the edge effects and inhomogeniety of original silicon as well as some decrease in lifetime τ during the detector fabrication. The measured energy resolution was ΔE = 25 keV for type 1 detector (E = 5.8 MeV) and ΔE = 15 keV for type 2 detector (E = 5.8 MeV).These values are much better than those obtained with large area detectors (ΔE = 70 keV for a detector with area of 4 cm × 4 cm). The equivalent noise measured by means of m.i.p. Landau distributionwas in the range of 4 keV for different detectors (the signal to noise ratio ≈ 20).

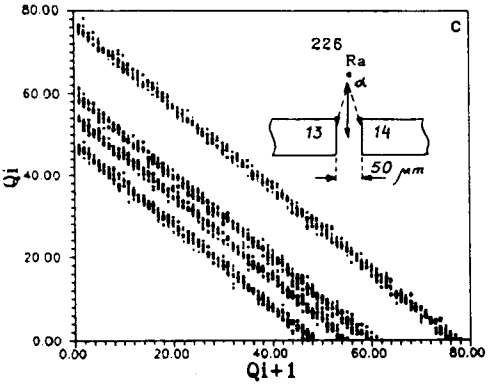

Figure 1. The two-dimensional distribution of charges Q_i and Q_{i+1} collected by neighboring strips (charges are in arbitrary units) .

CONCLUSIONS

Investigated silicon strip detectors have some advantages over usual detectors with the same total area. They have a better signal to noise ratio which is determined by much smaller capactance and reverse current of one strip. Such detectors can be used for determination of the alpha spectra of various substanses. They also can give information on spatial location of alpha sources with high position accuracy.

ACKNOWLEGEMENTS

We express our gratitude to Dr. Frolov and V.Pervertaylo for the fabrication of some strip detectors (the firm " Detector". This research has been partly supported by the International Atomic Energy Agency under research contract No. 7200/R2/RB.

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PROPERTIES IN HIGH RADIATION FIELDS OF HIGH PURITY SILICON DETECTORS

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INTRODUCTION

Some types of semiconductors are meant to operate under heavy radiation conditions. It may take place in the destroyed Chernobyl NPP and also in high energy hadron collider experiments. The properties of detectors are mainly determined by electrical characteristics of high purity silicon and also by the peculiarities of technology of their fabrication.

Peculiarities of the volume inhomogeneity distribution of τ and ρ in silicon samples, manufactured by different producers (Wacker, Topsil, ZTMC etc) have earlier been studied (1). Hear an effort was made to establish the correlation between revealed volume inhomogeneities of $\tau-$ and $\rho-$ parameters and the presense of growth sructural microdefects.

The present study has an aim to investigate the peculiarities of electrical properties evolution in high ohmic silicon and detectors manufactured on its base being in high radiation fields of γ-rays.

EXPERIMENTAL PROCEDURE AND RESULTS

The samples cut from float-zoned silicon (FZ-Si) were choosen for investigation. To achieve the above goals the Hall-coefficient measurements have been performed. When studying the evolution of electrical properties, the silicon crystals have been irradiated by the γ-rays at successiv steps with dose ranging from 10^6 to 10^9 rad. After each step the samples were removed to perform the necessery measurements. The effect of γ-irradiation on carrier concentration of high purity n- and p- silicon, produced by Topsil, Wacker and ZTMC (Ukraine) firms are shown at Fig.1.

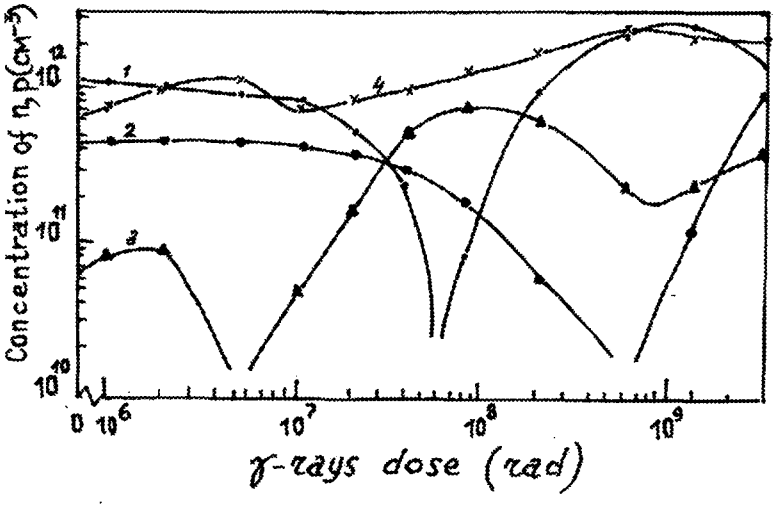

Figure 1. Dependence of carrier concentration on γ-irradiation doses for : 1 - ZTMC, 2 - Wacker, 3 - Topsil (all n - Si), 4 - ZTMC (p - Si).

As evidenced by this figure in n-Si the carrier density decreases with dose increasing, achieves intrinsic value and then invert into p-type conductivity where the hole concentration increases reaching the level of saturation near $2 \cdot 10^{12}$ cm^{-3}. Earlier it was shown that the hole density saturation occurs mainly due to creation of divacancy (2). The value of inverting dose in certain extent

corresponds to the initial carrier concentration in samples. The results of measurements are given in Table 1.

Table 1

Sample	Initial concentration n_0, cm^{-3}	Inverting dose Φ, rad
Topsil	$6 \cdot 10^{10}$	$5 \cdot 10^{6}$
ZTMC	$1 \cdot 10^{12}$	$6 \cdot 10^{7}$
Wacker	$5 \cdot 10^{11}$	$6 \cdot 10^{8}$

It can be seen that the only exception from the above established rule is Wacker n-Si. Comparation of the carrier removal rates $\varepsilon = -\Delta n/\Delta\Phi$ in range $(1 - 2) \cdot 10^7$ rad shows that the Wacker n-Si considerably distinguishes from all other samples. Parameter ε for the Topsil, ZTMC and Wacker crystals equals $4 \cdot 10^{-5}$ cm^{-1}, $2 \cdot 10^{-5}$ cm^{-1} and $3.5 \cdot 10^{-6}$ cm^{-1} respectively. As can be seen the Wacker n-Si has the most radiation hardness among all examined n-type silicon. This phenomenon may be attributed to the presence of electrically nonactive impurities that serve as rather effective drains for the primary radiation-induced defects.

As far as the p-type silicon samples are concerned (Fig. 1, curve 4) the test that were perfomed revealed the ZTMC p-Si (with $p_0 = 10^{12}$ cm^{-3}) as a most radiation-damage resistant material of all examined p- silicon crystals under γ-irradiation.

In order to test the electrical and spectrometrical properties of detectors made on the basis of investigated materials, the detectors with diode struture were fabricated using both surface-barrier and planar technology. As a function of γ-irradiation from ^{60}Co the I-V, C-V and R-V measurements were perfomed. Detectors were exposed at successive steps by γ-irradiation as it was previously done for the silicon materials. All detectors were measured being biased at 50 V. The results achieved show that the both diode reverse current and energy resolution for all examined detectors as a function of irradiation dose were found to be similar in the dose ranging from 10^6 to $1.2 \cdot 10^7$ rad. Besides, there were no evidence of direct connection between the reverse current and energy resolution also as a function of dose.

It should be however noted that at initial range of irradiation there can be seen the improving of detector characteristics, but this effect takes place at different doses for different silicon materials. For Wacker n-Si this effect reveals at dose of 10^6 rad, for ZTMC - $6 \cdot 10^6$ rad (n-Si), $2 \cdot 10^6$ rad (p-Si) and for Topsil n-Si - near 10^6 rad. The effect of characteristics improving is confirmed also by the increasing of carrier concentration deduced from the C-V measurements. This the so-called effect of little doses can be associated with the non-equilibrium states, which are creating in the process of detector fabrication. The γ-irradiation leads to the proceeding of radiation-stimulated reactions with the point defects involved. This reaction brings the diode structures to be well regulated and to the disappearance of non-equilibrium states.

Concernig the technology, it should be noted that for planar detectors fabricated by different producers (Ukraine and Australia) we observed significant increase of the reverse current and deterioration of energy resolution. At dose of 10^7 rad the energy resolution deteriorates by 4 times and reverse current increases by one order while the characteristics of surface-barrier detectors did not change considerebly.

CONCLUSION

1. Wacker n-Si crystals show the most radiation hardness among all examined n- silicon under γ-irradiation.
2. There is no direct connection between the reverse current and energy resolution as a function of γ-irradiation dose for all detectors examined.
3. The radiation hardeness of detectors fabricated by the surface-barrier technology without high temperature treatments is higher than that for planar detectors where such treatments have been applied.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE THE USE OF COMMON PHOTODIODES FOR DEPTH DOSE MEASUREMENTS IN PHOTON AND ELECTRON FIELDS

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2. (see page 7)

ABSTRACT (See instructions overleaf)

Semiconductor detectors are widely used for radiation dosimetry due to their high sensitivity, small size, and absence of external bias. In previous investigations the authors found that common photodiodes, which have the same junction characteristics of semiconductor detectors, present a linear dose-response relationship. In this paper, the BPW-34, a PIN photodiode manufactured by Siemens, was tested for measuring the central axis depth doses in both photon and electron fields.

Measurements were carried out with the photodiode connected in the photovoltaic mode to the input of an integrating electrometer developed in the Nuclear Energy Department at the Federal University of Pernambuco, Brazil. Depth dose distribution curves were obtained in a slab perspex phanton measuring 30cm x 30cm x 20 cm. The doses were measured in a ^{60}Co gamma beam, in 6-MV and 10-MV X-ray beams, and in a 10-MV electron beam, for field sizes of 5 x 5 cm^2, 10 x 10 cm^2, and 20 x 20 cm^2.

The results of the measurements with the photodiode agree with those obtained with the ionization chamber at the same irradiation conditions within less than 1%. These results show that the BPW-34 photodiode can be used for mapping the depth dose distribution in both photon and electron fields. Its small size makes the BPW-34 photodiode suitable for patient dose measurements during radiotherapic treatments.

IMPROVEMENT OF THE SENSITIVITY OF CdTe SEMICONDUCTOR DETECTOR IN THE HIGH ENERGY REGION

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INTRODUCTION

Cadmium Telluride, CdTe, semiconductor detectors have sufficient band gap energy (1.47 eV) to use at room temperature, and their atomic number are so large (48 and 52) that their photon detection efficiency is more excellent than that of Si or Ge. Recently CdTe crystals have become easily available because of improvements in the crystal growth method. It is a useful X-ray detector, because it has good energy resolution and high efficiency at the full energy peak at less than a few hundred keV of incident photon energy. However, if the incident photon energy become higher, the efficiency of the full energy peak become worse, and it is very difficult to distinguish the full energy peak above 1 MeV, because the mobility of charge carriers in the CdTe crystal is much smaller than in Si and Ge and it is difficult to produce a larger volume element.

In order to analyze the energy of several radioisotopes, it is necessary to improve the sensitivity of CdTe detectors in high energy regions. We have previouly suggested a multi-layered structure of CdTe elements.[1] This paper describes a simulation and experiment to improve the efficiency of the full energy peak in the high energy region above 1 MeV.

METHOD OF SENSITIVITY IMPROVEMENT

It is common to increase the detector volume in order to improve the efficiency of the full energy peak in the high energy region. However, the charge carrier mobility of CdTe, particularly in holes, is much smaller than that of Si and Ge.[2,3] Most holes drifting in the CdTe crystal are trapped at trapping centers, for example lattice defects, and their mean free path is shorter than the thickness of the detector. Therefore most holes can not reach the collecting electrode and hence all the energy deposited in the detector by photons does not contribute to the output pulse height.

In order to collect more holes, we suggest a multi-layered structure of CdTe elements as shown in Fig. 1. The output pulse from each layer is superimposed by connecting them in parallel. By this method only one amplifier is needed and therefore minimizing the cost to improve sensitivity. There is only air between each element in order to minimize the photon interaction other than with CdTe elements. The sensitivity in the high energy region is improved, because the collection efficiency of charge carriers become higher and the detector volume is increased.

SIMULATION

The response function of the CdTe detector was predicted by simulation with the Monte-Carlo calculation code, EGS4 (Electron Gamma Shower Ver. 4)[4] to evaluate the degree of sensitivity

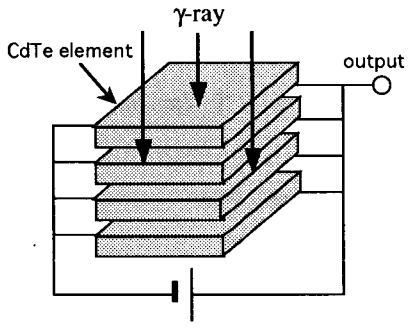

Fig. 1. Multi-layred CdTe detector.

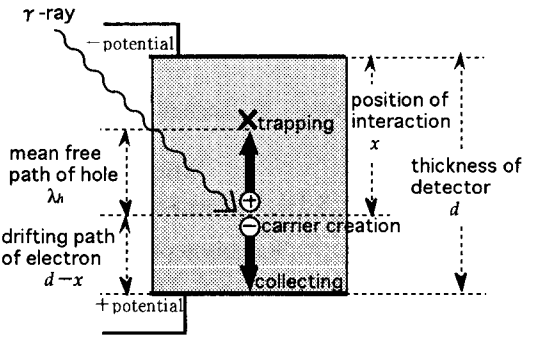

Fig. 2. Simulation model considering the hole trapping in the semiconductor detector.

improvement. In this calculation, the hole trapping in the CdTe crystal was considered. In the case of a planar type detector, the collected charge $Q(t)$ on the electrode is presented as,

$$Q(t) = \frac{q_0}{d}\left(\text{drifting length of electron} + \text{drifting length of hole} \right) \quad (1)$$

where d is the thickness of detector, and t is the time from creation of charge carriers q_0 by photon interaction . In Fig. 2, the charge carriers are created at the interaction position and the distance from this point to the negative electrode is indicated by x . However the holes are trapped as they drift in the CdTe crystal (the mean free path of holes is λ_h), the collected charge $Q(x)$ is presented as,

$$Q(x) = q_0\left[\frac{d-x}{d} + \frac{\lambda_h}{d}\right] \quad (2)$$

When the energy deposition in the CdTe detector was calculated, the energy calculated by EGS4 was multiplied by the weight function, $[(d-x)/d+\lambda_h/d]$. Therefore the energy derived from this model is less than the energy when the hole trapping is not considered. The simulated result with this model was in good agreement with the experimental result of a one-element CdTe detector as shown in Fig. 3 which indicated that the simulation model was correct.

By using this model, we predicted the degree of sensitivity improvement of a multi-layered CdTe detector in the high energy region. Figure 4 shows the energy dependence of the relative sensitivity by the simulation. The relative sensitivity is defined as the ratio of the full energy peak efficiency of four layers to that of one layer. It was predicted by simulation that the relative sensitivity would exceed four in the energy region of more than 1 MeV and therefore the improvement effect would be larger than the simple multiplication of the number of elements.

EXPERIMENT

Figure 5 shows a block diagram of the present experiment. The applied voltage to the superimposed CdTe elements was adjusted so that the measured voltage of 1 layer and that of 4 layers was the same. A check source of ^{60}Co (3.29MBq) was used in this experiment. The distance between detector and source was kept long enough so that the direction of the gamma-rays was parallel and that the dose rate errors at the detector became a minimum. The output pulse height distribution was measured and the full energy peak efficiency was determined by counting the 1.33 MeV peak of ^{60}Co.

The result is shown in Fig. 6. The sensitivity of the 1.33 MeV peak in the case of 4 layers was 4.9 times better than the case of 1 layer. The improvement of the sensitivity is in good agreement with the simulation result as shown in Fig. 4.

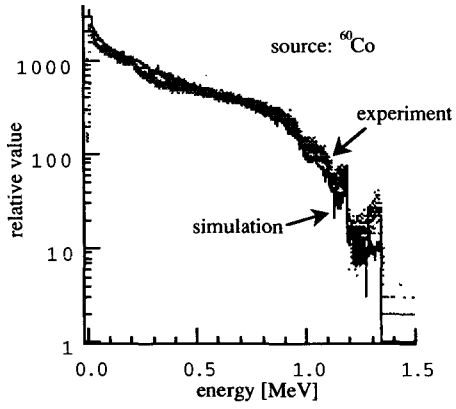

Fig. 3. Simulation results considering the hole trapping for the 1 element detector compared with the experiment.

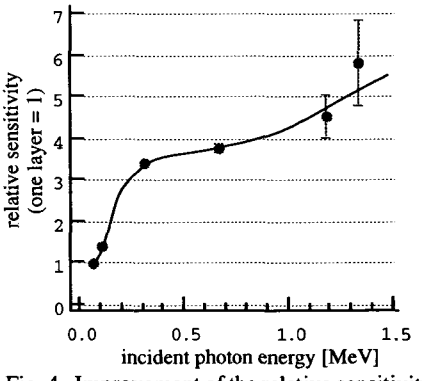

Fig. 4. Improvement of the relative sensitivity for the 4 layered detector predicted by simulation.

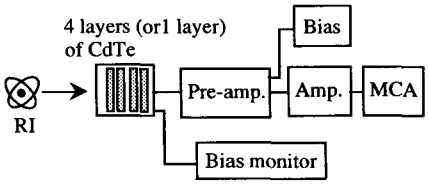

Fig. 5. Block diagram of the experiment.

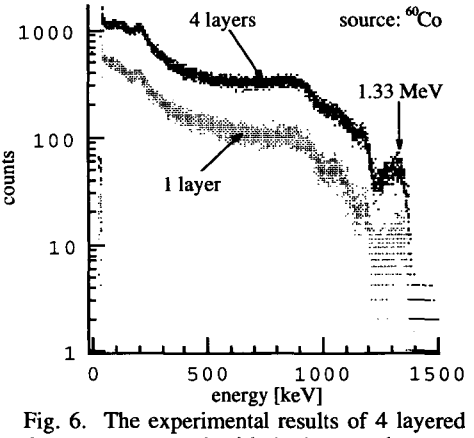

Fig. 6. The experimental results of 4 layered detector compared with 1 element detector.

CONCLUSION

A multi-layered structure of elements was proposed to improve the sensitivity of a CdTe semiconductor detector in high energy regions. The response function of a semiconductor detector whose charge carrier mobility is even as small as CdTe could be simulated, and the degree of sensitivity improvement in the high energy region above 1 MeV was predicted by using this model. The results of both simulation and experiment indicated that the improvement effect of the full energy peak with a multi-layered CdTe detector was larger than the effect of simply multiplying the number of elements.

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CALIBRATION OF PORTABLE HPGe SPECTROMETER FOR INDOOR DOSE RATE ESTIMATIONS

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INTRODUCTION

In situ gamma spectrometry is a powerful tool to study indoor and outdoor dose rates. A high resolution spectrum allowing analysis of individual photo peaks provides valuable information on the relative contribution of the various nuclides to the total exposure rates. In situ techniques for measuring the dose rate, resulting from the gamma radiation and characterizing its sources, with a gamma ray spectrometer have been used successfully in the outdoor environment (1-4). In principle, the same methods can be applied to indoor radiation measurements. However, due to complex or generally unknown geometry in indoor areas, the methods for outdoor environments can not be applied directly to indoor ones. In the present work we investigate the applicability of the calibration factors derived from field geometry to indoor geometry.

METHODOLOGY

In order to check whether calibration factors for field geometry can be applied to indoor radiation measurements, we applied to our measured indoor and outdoor spectra a geometry independent method, the spectral stripping method (5), (6), for determining the gamma dose rates, which does not require any assumptions concerning the source geometry. The measured indoor and outdoor spectra were performed with a portable coaxial HPGe detector (10% efficiency).

A count registered by the detector can be caused by the full or partial absorption of an incident photon or by the passage of a cosmic ray produced charged particle. In order to convert to gamma dose rate, the spectrum must be stripped of the partial absorption and cosmic-ray events leaving only the events corresponding to the full absorption of a gamma ray. The resulting spectrum, which represents both primary as well as scattered photons, can then be converted to the total incident flux spectrum by applying the full absorption efficiency curve of the detector.

The flux spectrum is then used to calculate the total gamma dose rate which can be compared with the sum of the individual dose rates for each radionuclide (natural and man-made) obtained with the use of the calibration factors for field geometry. When these two values are in agreement, one can assume that the calibration factors being used are correct.

Concerning the cosmic-ray events induced in the detector we followed the methodology given in ref. (5), (6). The cosmic ray count rate in the region below 3 MeV in a typical Ge detector is generally small compared to the gamma count rate. Since there is no natural gamma line present between 3 and 4 MeV, this region is used to estimate the cosmic ray count rate which is then extrapolated back to 0 MeV and subtracted out from the spectrum. Although there may be some variation in the energy distribution, we have assumed as in ref. (5), (6) that the cosmic continuum is flat in this region.

Concerning the partial absorption events induced in the detector we followed the methodology given in ref. (5), (6) and ref. (7).The events corresponding to partial absorption in the detector make up a large fraction of the counts in the spectrum and are not as simple to strip out. For the most part, these events are the result of compton scatter in the crystal itself or the surrounding cryostat. For single scatter events, a continuum of counts will form in the spectrum reaching a maximum energy at the compton edge given by

$$E_c = E_p - \left(\frac{2}{0.511} + \frac{1}{E_p} \right)^{-1} \quad (1)$$

where E_c is the energy of the Compton edge and E_p is the energy of the incident gamma ray.

Okano et al (7) have used a distribution of counts with energy which is essentially a single step function between the low end of the spectra and E_c. An example of the actual distribution of counts in the continuum for normal incident 0.662 MeV photons in our detector is shown in Figure 1. It can be seen that: (a) the energy dependence below Ec can not be described by a constant and (b) there are a number of counts that lie above E_c, which are a result of multiple scatter events within the crystal. Based on this, Miller (5), (6) incorporated into the stripping routine, a multiple step function fit to the continuum. In our method which is simple, quick and acceptable accurate, a two function fit to the continuum was incorporated into the stripping routine. The region below E_c is fitted with a polynomial function of second order and the region between E_c and E_p with the sum of an exponential and linear function. The different parameters of the polynomial and exponential function depend on the incident photon energy and have been obtained assuming a constant Peak to Compton ratio for all energies. For incident photons 0.662 MeV, the polynomial and exponential functions used in the stripping routine are also shown in Figure 1. The correlation coefficients between measured and fitted data for the fist region, (100 keV until E_c) and the second one (E_c until E_p) are 0.76 and 0.99 respectively. It should be noted also, that when we write Peak to Compton ratio, we mean the number of counts at E_p divided by the average number of counts in the channels for the range from $0.99 \cdot E_c$ through E_c, and not the one given by the manufacturer, which is determined as the ratio which is found by dividing the number of counts in the 1.332 MeV peak channel by the average number of counts in the channels for the range from 1.040 through 1.096 MeV. One must have in mind that the number of counts in the peak

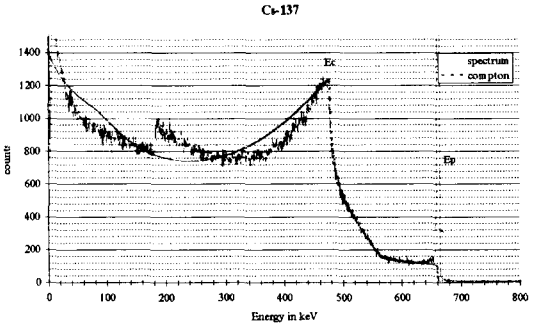

Figure 1. Cs-137 spectrum and compton fit.

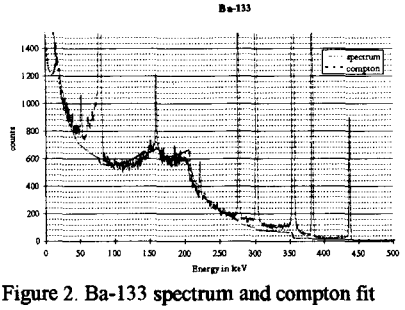

Figure 2. Ba-133 spectrum and compton fit

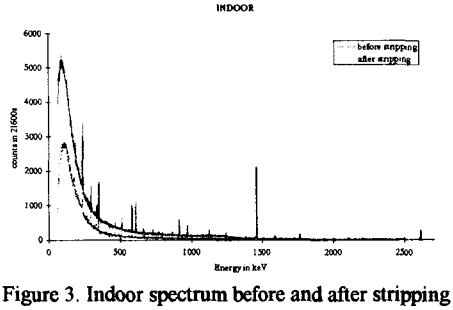

Figure 3. Indoor spectrum before and after stripping

(E_p) are considerably higher that the counts in the Compton edge (E_c).

Using the assumed shapes for the cosmic and partial absorption continua we proceed to a computerized stripping operation which subtracts out from the measured in-situ spectrum those counts that represent cosmic-ray events or partial absorption of gamma rays in the detector. The stripping operation is initiated at the highest energy gamma peak (2.615 MeV) and involves subtracting the assumed continuum of counts which is lower in energy. The operation continues for succeeding lower energies gamma peaks down to 100 keV. In order to check our stripping operation we test it in a spectrum obtained with incident γ rays emitted from a Co-60 and Ba-133 point sources. It can be seen in Figure 2, that the compton distribution deduced from the stripping operation describes successfully the experimental values.

An example of a stripped indoor spectrum as compared to the original spectrum is shown in Figure. 3. Typically about 50% of the counts are removed. These counts are removed from the continuum portion of the spectrum, while the peaks due to the full absorption of primary flux are preserved. The stripped spectrum of Fig. 3 is converted to incident flux by applying the full absorption efficiency of the detector.

Having calculated the flux energy distribution $\varphi(E)$ the absorbed dose rate in air due to the gamma radiation can be easily calculated by

$$\dot{D} = \int_0^{E_{max}} \varphi(E) \cdot E \cdot \mu(E) \cdot dE \qquad (2)$$

where $\mu(E)$ is the mass absorption coefficient for air at energy E and E_{max} is the highest energy gamma line (2.615 MeV).

RESULTS AND DISCUSSION

In order to check the applicability of the field calibration factors to indoor geometry we performed long duration measurements (six hours). Specifically 5 measurements took place in buildings and 6 in outdoor environments, with 3 of them on roads and pavements. For each spectrum we calculated the total external dose rate (nGy/h) using 3 different methods. (a) The field calibration factors assuming equilibrium along the various nuclides in the U and Th series, so that the measurement of any one spectral line yields concentration values for the entire series, although averaging several spectral lines will give a more precise value. For the U-series we used the lines of 0.352 MeV (Pb-214) and 0.609 MeV (Bi-214) and for the Th-series the lines of 0.583 MeV (Tl-208) and 0.911 MeV (Ac-228). We also measured the lines 1.461 MeV (K-40) and 0.662 MeV (Cs-137). (b) The field calibration factors without assuming equilibrium along the various nuclides in the U and Th series, using equation 2 for the unscattered photons and multiplying the resultant dose rate for each energy with the appropriate build up factor (8). (c) The spectral striping method using equation 2. The results are shown in Table 1.

	using field factors (assuming equilibrium)	**using eq. 2 (without assuming equilibrium)**	**using stripping method**	**field factors to stripping method ratio**
Building	46.7	48.0	53.6	0.87
Building	48.5	49.3	53.23	0.91
Building	49.3	46.3	45.33	1.09
Building	58.4	59.7	63.65	0.92
Building	61.7	62.8	63.98	0.96
Road & Pavement	16.5	15.0	22	0.75
Road & Pavement	31.4	30.7	35.85	0.87
Road & Pavement	32.3	33.5	38.43	0.84
Over Soil	39.0	38.0	36.9	1.06
Over Soil	44.8	44.0	45.13	0.99
Over Soil	44.9	44.2	44.05	1.02

Table 1. Total external dose rate in air (nGy/h)

The fact that the field factors have been verified for the outdoor environment indicate that the stripping operation used is accurate. This can be seen from the ratio between the dose rates estimated by field factors and the stripping operation for measurements over soil, in Table 1.

One can see, observing mainly the results for buildings and houses, that disequilibrium in the U series due to exhalation of Rn-222, from the soil surface or the building materials, does not affect the total dose rate significantly. Also, it is notable, that the use of field calibration factors for roads and pavements underpredict the dose rate

The generally good agreement between the two dose rate estimates between calibration factors and stripping method, which has been observed also for almost all indoor spectra, used for this test, justify the applicability of field geometry factors to indoor geometry at least in the case of masonry structure which is typical for most of the buildings and houses in Greece.

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COMPUTERISED GLOW CURVE DECONVOLUTION:APPLICATION TO THERMOLUMINESCENCE DOSIMETRY

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INTRODUCTION

Computerised glow curve fitting methods (CGCD) used to improve the dosimetric applications of thermoluminescence have come of age. A recent monograph (1) outlines the glow curve deconvolution of various TLD materials and then discusses the many and varied applications to dosimetry. These include: dramatic improvements in precision and minimum measureable dose, dose re-estimation using phototransferred thermoluminescence (PTTL), simultaneous estimation of dose and elapsed time following irradiation, estimation of fading and sensitivity corrections, optimal annealing procedures, mixed field beta/gamma and neutron/gamma dosimetry, and various quality control applications at both the growth, material characterization and readout stages. At Ben Gurion University we are currently investigating some of the more recent developments in CGCD concerning quality control and neutron/gamma dosimetry. In addition, we present some preliminary results concerning abnormal glow curves obtained for low energy x-rays and beta rays due to oxygen contamination during 400oC in air annealing. These latter results should be of concern to users involved in superficial dose estimation using LiF-TLD.

CGCD APPLIED TO QUALITY CONTROL

This application, although still in its infancy (2), is potentially one of the most. important applications of CGCD. One can envisage important material-selection improvements in the manufacturing stage of the material, in the manufacturer's quality control of the final product and in the final stage of glow curve readout involving verification of dose data as well as diagnostics of the reader malfunctions. The final stage is especially important since glow curve analysis automatically involves a form of permanent storage of the dose information and completely answers one of the complaints in the use of TLD regarding a permanent record of the dose information. For these reasons many operational dosimetrists have become enthusiastic proponents of glow curve storage in order to be able to test for abnormalities and dose verification. Hoot and Landrum (3) were the first to publish a library of error conditions to be verified by glow curve analysis. Liu et al (4) in their study of the optimization of the readout procedures for the Harshaw 8800 automatic TL dosimetry system, report occasional irregular glow curves of thin TLD-700 dosemeters which were easily identified by the CGCD routine currently operative in the 8800 system. Tan et al (5) are developing computerized programs to screen for abnormal LiF glow curves. Bos et al (6), studied the kinetic trapping parameters of different Harshaw Production batches. This study has a direct relevance to the question of production quality control and the non-universality of dosimetric properties in Harshaw LiF products. Generally, users request uniform sensitivity in batch selection and ignore details of glow curve shape. It is clear that users interested in maintaining dosimetric characteristics over the years in their procurement of dosimetric material should specify not only overall batch sensitivities but also should place specifications on glow curve shapes.

At Ben Gurion University we have studied how to retrieve dosimetric information from distorted glow curves using CGCD (2). Fig.1 illustrates a type of distortion involving distortion over a temperature range of approximately 20^{o}C in the maximum of peak 5 in a "four-glow-peak glow curve".

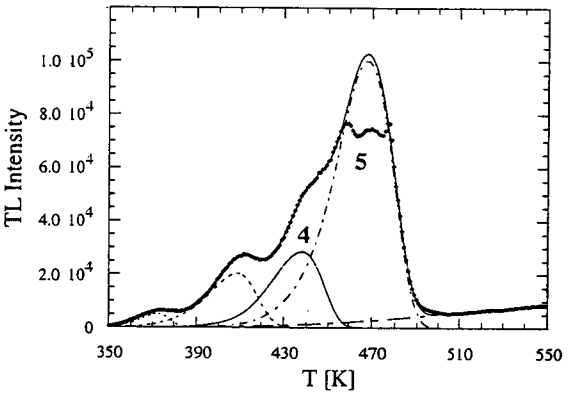

Figure 1. Gamma induced glow Curve of TLD-100 illustrating system malfunction in the central portion of peak 5. (•) experimental

The analysis of the glow curve was carried out by excluding the distorted region from the computerised fitting procedure. In this particular exercize, and for demonstration purposes, this exclusion procedure is carried out by visual inspection. It is no problem to configure the fitting procedure to automatically identify and exclude the distorted regions. The results of the computerised analysis in all the cases studied allowed us to retrieve the dosimetric information to an accuracy of better than 2%.

CGCD APPLIED TO NEUTRON-GAMMA DOSIMETRY

This application has been discussed in detail by Horowitz and Yossian (1). The dominance of TLDs in the personal dosimetry of photons and beta rays suggests that further efforts must be invested in the use of TLDs in fast neutron environments simply because of the increasing availability of automated TL facilities in nuclear, medical and industrial installations where fast neutron dosimetry is a subject of continuing concern. Most previous applications centered on the use of TLDs in albedo dosimetry or in the use of the fast-neutron-sensitive high temperature glow peaks in LiF:Mg,Ti, but these approaches suffer from several disadvantages. At Ben Gurion University we are currently studying the possibility of the application of peak 4/5 ratios in the glow curve of TLD-600 in the separation of gamma and neutron dose. A typical glow curve following fast neutron irradiation is shown in Fig. 2.

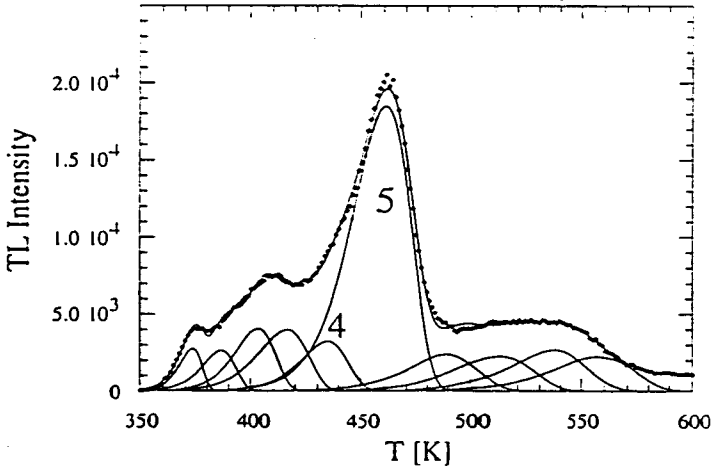

Figure 2. Glow curve of TLD-600 following fast-neutron irradiation

The ratio of peak 4/peak 5 is approximately 15% compared to approximately 30% following gamma irradiation. The success of this technique depends critically on the reproducibility of these 4/5 ratios which is the subject of our current investigations.

CGCD APPLIED TO SURFACE DOSE ESTIMATION

Dosimetric measurements of low energy radiation using LiF-TLD may be subject to difficulties due to surface contamination by OH^- ions during 400°C air-annealing. Differences in the TL efficiency between air and N_2 annealing have been well documented in the literature (7), but not much attention has been given to glow curve analysis. A typical glow curve of TLD-100 following irradiation by ^{63}Ni beta rays (air-annealing) is shown in Fig. 3. The intensity of the low temperature peaks 2-3 is greatly enhanced relative to peak 5, and in addition, new low-temperature peaks appear. Similar glow curves have been obtained following ^3H or alpha-particle irradiation. The effect increases with subsequent air-anneals and is totally absent when the chips are annealed in N_2. Dosimetrists involved in superficial dose estimation of low-energy radiations, should, therefore, pay especial care to the purity of the N_2 atmosphere used in the high temperature annealing procedure and during readout. Changes in sensitivity to low energy-radiation may occur which will not be detected by high energy, volumetric, calibration.

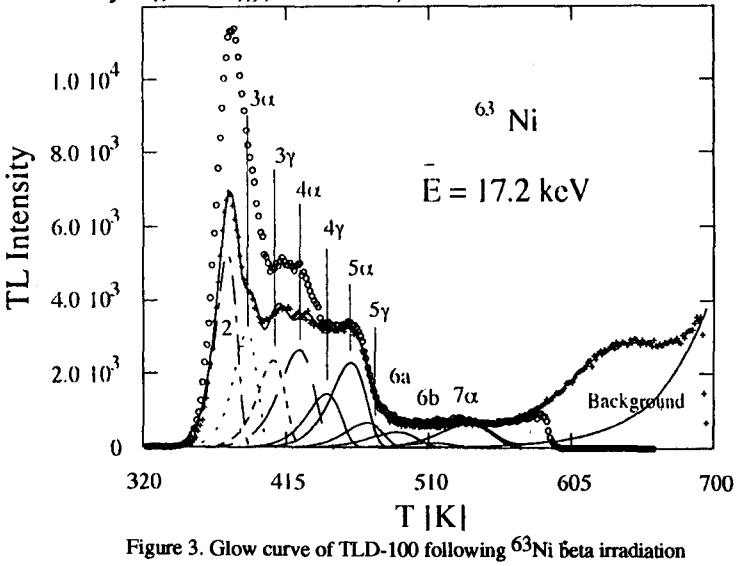

Figure 3. Glow curve of TLD-100 following ^{63}Ni beta irradiation

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OPTIMIZATION OF MAGNESIUM BORATE THERMOLUMINESCENT MATERIAL FOR RADIATION PROTECTION DOSIMETRY

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Abstract - The dosimetric performances of developed MgB_4O_7:Dy,Na, optimized magnesium borate TL material have been evaluated and the results are presented. Three types of thin active layer TL dosemeters based on new magnesium borate are developed to provide beta and neutron dosimetry.

INTRODUCTION

Valuable features of magnesium borate TL material such as near tissue-equivalency, high sensitivity, and its good performances for X, gamma, beta and neutron dosimetry, were the ground that systematic studies were carried out for further improvement of this material. Previously prepared sensitized magnesium borate thermoluminescent materials activated with rare earth elements, dysprosium or thulium, are highly sensitive phosphors (1). However, these TL materials show remarkable fading, multiple peak structure of the main TL peak and slight chemical instability. These shortcomings of magnesium borate phosphors have been entirely overcome by development of the original preparative procedure and development of MgB_4O_7:Dy,Na - thermoluminescent material with optimal TL characteristics.

Also, three types of thin active layer TL dosemeters based on new magnesium borate are developed to provide beta and neutron dosimetry.

The performances of these TL dosemeters are presented.

MgB_4O_7:Dy,Na - TL performances

TL material, MgB_4O_7:Dy,Na differs considerably in its chemical composition from all types of magnesium borate, TL materials previously synthesized. TL material is nearly tissue equivalent, with an effective atomic number for photoelectric absorption, Zeff = 8.55.

MgB_4O_7:Dy,Na shows single TL peak at about 185°C-190°C, and the sensitivity which is, depending on the reader type used (2) 6-15 times higher than that of LiF:Mg,Ti (TLD-100). The technology developed proved to be efficient for evident modification of the main, multistructure peak which was characteristic for the previously prepared magnesium borate TL materials, with the maximum at about 165°C, to simpler TL peak with the maximum at about 185°C-190°C.

MgB_4O_7:Dy,Na is a complex, composed of two phases, of about 70% of magnesium tetraborate, MgB_4O_7, and of about 30% of magnesium meta borate, MgB_2O_4. The X-ray diffraction analysis shows that a complex comprising the phases, MgB_4O_7 and MgB_2O_4. With increasing of the content of the phase MgB_2O_4, the glow TL peak is shifted to higher temperatures. Consequently, fading for the optimized magnesium borate TL material was found to be only of the order of less than 2-5% for three months, at the ambient temperature of 30°C in dark, compared to 35% for two months fading for previously prepared magnesium borate TLDs. In fact, this very low fading of MgB_4O_7:Dy,Na are comparable to the uncertainties due to the homogenity of the batches and also comparable to this defined for LiF TL phosphors.

The dosemeters are in form of solid sintered pellets with a diameter of 4.5 mm. The measurements were performed on Harshaw Model 2000 TL Analyser and partly on ALNOR Dosacus TLD reader.

Thin layer TL detectors for beta and neutron dosimetry based on MgB_4O_7:Dy,Na

TL Detectors for Beta-Ray Dosimetry

The optimal thermoluminescent dosemeter for beta dosimetry is thin detector of a few mg.cm^{-2} active layer with the detection threshold lower of 20 μGy, nearly tissue equivalent, insensitive to humidity, with low fading, excellent reusability and mechanically rugged.

Graphite mixed MgB_4O_7:Dy TL detectors with 2-3% of graphite were introduced in beta dosimetry by Prokić (3). A general problem with the graphite mixed TL dosemeters is that the TL sensitivity is relatively low. Particularly, the problem of high fading of previously prepared MgB_4O_7:Dy was the reason for development of TL detectors for beta dosimetry based on new, MgB_4O_7:Dy,Na TL material.

Two types of thin active layer TL detectors are developed to provide beta dosimetry: (a) The first type is an active thin TLD layer, difusion bonded onto graphite mixed magnesium borate base, and (b) A type of ultra thin TLD layer infiltrated on the surface of non-thermoluminescent sintered magnesium borate pellets, by their immersion in the activator (Dy) water solution.

The gamma ray sensitivity of the first type of thin layer dosemeters, with the TLD layer thickness of about 5 mg.cm^{-2} are rather high; in fact, relative sensitivity amounts 28, normalized to TLD-700 for which sensitivity to ^{60}Co gamma rays is taken as 100. The second type of TL dosemeters prepared by the variation of the period of immersion of sintered pellets (and by changing the concentration of the activator solution), the active TLD layer is selected to be ultra thin, 1 μm, to about 4 mg.cm^{-2}, or more. After second sintering, the relative TL sensitivity for ultra thin TLD layer, about 4 mg.cm^{-2} to ^{60}Co gamma rays is 16. For comparison, graphite mixed magnesium borate TLDs with the similar effective thickness has relative gamma ray sensitivity of about 6 to 7, with the excellent low beta energy response of 0.70 to 0.83 (4). Beta ray response of the two types of developed thin TLDs for low energy beta radiation of ^{147}Pm (Emax = 0.225 MeV) is 0.68 and 0.77, respectively, i.e. similar to that of graphite mixed magnesium borate TLDs. Beta ray response is given as the ratio of the response of a bare detector to low energy beta radiation (^{147}Pm), to the response to gamma radiation (^{60}Co).

Recent progress in the development of thin layer TL detectors has made it possible to improve the sensitivity without changing (no more than for 10%) the efficiency for beta dose measurements, with a flat beta ray energy response curves for skin dose measurements when applied Mylar foil of corresponding thickness to fit the 0.07 mm tissue for beta ray maximum energies above 0.2 MeV.

TL Detectors for Personal Neutron Dosimetry

At the present, only ^{6}LiF:Mg,Ti, as commonly used TL detector, and also $Mg^{10}B_4O_7$:Dy,Na can be used as a neutron dosemeters. Sintered TL dosemeters based on magnesium borate ^{10}B enriched, along with ^{10}B depleted magnesium borate TLDs can be used successfully for separate measurements of neutron and gamma doses in an albedo type personal neutron dosemeter system (5). In fact, thermal neutrons are absorbed mostly in the first 0.2-0.3 mm layer of neutron sensitive TL dosemeter, while absorbtion of photons is uniform through the detector thickness. Hence, by the application, in principle, the technologies developed for the preparation of thin layer TLDs for weakly penetrating radiation, it was prepared thin neutron sensitive TLD layer, $Mg^{10}B_4O_7$:Dy,Na with the active TLD thickness of about 65 mg.cm^{-2}.

However, the main intention of this project was an attempt to improve the thermal neutron sensitivity of the $Mg^{10}B_4O_7$:Dy,Na, but not only to obtain high ratio of neutron to gamma

response. The neutron highly sensitive TL dosemeter is prepared by infiltration of the thin active layer on the surface of non-thermoluminescent magnesium borate sintered pellet. The isotop separated lithium (^{6}Li) and boron (^{10}B) content is added in the active TLD layer and as a very thin enamel wrapper, over the pellet before the second sintering. With this highly efficient method, neutron sensitivity is enhanced for nearly the factor of even 2 to that of ^{6}LiF:Mg,Ti (TLD-600). This value presents only the preliminary results but it pointed at the correct direction of the method under development.

In the preliminary test, in order to measure the neutron sensitivity, moderated ^{252}Cf source was used. Detectors are placed on the surface of the polyethylene cylinder. Since for moderated ^{252}Cf neutron spectrum, the fluence and neutron dose equivalent were not known, the relative neutron sensitivity is normalized to TLD-600. Further study will be performed by irradiation with the thermal neutrons.

CONCLUSION

Thermoluminescent dosemeter MgB_4O_7:Dy,Na differs considerably in its chemical composition from all types of magnesium borate TL detectors previously synthesized. It presents the optimal TL material based on magnesium borate with negligable fading, which is comparable to that defined for LiF TL phosphors.

MgB_4O_7:Dy,Na as a high sensitive nearly tissue-equivalent TL material are attractive for various dosimetric applications, with TL performances suitable for photon, beta and neutron dosimetry.

Two types of thin active layer TL detectors are developed to provide beta dosimetry with high sensitivity to gamma, and the high response of low energy beta radiation. Also, an unique neutron dosemeter is under development which exhibit a significant improvement in neutron sensitivity.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE THE DOSIMETRIC CHARACTERISTICS OF THE NEW HARSHAW HIGH SENSITIVITY, COPPER DOPED LITHIUM FLUORIDE

AUTHOR(S) NAME(S) R. A. Tawil, M. Ramlo, N. A. Karpov, K. J. Velbeck

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2. (see page 7)

ABSTRACT (See instructions overleaf)

HARSHAW/BICRON◆NE has developed the capability of producing high sensitivity, copper doped, lithium fluoride (LiF:Mg,Cu,P) material for thermoluminescence dosimetry (TLD). This material has been proven to be about twenty times more sensitive than standard lithium fluoride (LiF:Mg,Ti), and it can be produced with ^{7}Li, ^{6}Li, and natural Li. TL chips made of this high sensitivity lithium fluoride can be mounted in standard four-element cards in either Teflon® or Kapton®. This paper reports the results of the testing done to characterize the dosimetric properties of this material, both as bare elements and in cards. The energy response was found to be relatively flat and fade is negligible. Additional results reported in this paper are detection threshold, reproducibility, batch uniformity, residual signal, linearity, ambient light-induced thermoluminescence and fade, and spectral response. The results of these tests show this high sensitivity material to be an almost perfect material for environmental, personnel, and medical dosimetry applications.

THE INFLUENCE OF THE ANNEALING PROCEDURE ON THE THERMOLUMINESCENCE READER CALIBRATION OF A PERSONAL DOSIMETRY SERVICE

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INTRODUCTION

Personal dosimetry services are required to distribute monthly a large quantity of dosemeters: their handling has thus to be easier and safer so that detectors could be read by an automatic reader. As a solution, badges have been devised containing more plastic encapsulated chips. Alternatively, individual detectors have been sealed into a plastic envelope with an appendix containing the identification bar code. A problem arises when these arrangements are applied to LiF:Mg,Ti (TLD-100) detectors: annealing is impossible because the procedure requires a 400°C heating step, that the plastic envelope cannot withstand.

The need to handle a large quantity of dosemeters has led to shorter set-up times. Thus, the annealing cycle for the unsealed chips has been modified from the "400°C, 1h and 80°C, 24 h" routine to the "400°C, 1h and 100°C, 2 h" one. The theory shows that the latter may worsen the reproducibility of repeated measurements and introduce sensitivity variations between chips of the same batch (1,2).

Objective of this paper is to compare the performances of three detector groups, two of them annealed by the two above-mentioned routines, while the third one is left without annealing.

METHODS

The batch of 90 individually coded TLD-100 chips has been divided in three 30 chip groups, A, B, C, each one further split to prepare a calibration set of six subgroups of five chips each. Five of these subgroups are irradiated with a Co-60 source in the range of kerma in air from 600 μGy to 10 mGy, under secondary particles equilibrium conditions. The sixth subgroup is read without irradiation, to evaluate the system background signal.

The reader is the manual Harshaw B+C type, always kept on. The applied HV to the PM tube is left unchanged. The operator acts carefully to obtain a good readouts reproducibility. The features of the reading cycle are:

- heating rate: 14°C s^{-1}
- initial temperature of the readings: 50°C
- integration interval: 120°C - 250°C
- at least five initialization cycles of heating and cooling with a dummy chip before reading the chips of the calibration sets.

The group A is annealed by the "400°C, 1h and 80°C, 24 h" routine and the group B by the "400°C, 1h and 100°C, 2 h" routine. The group C is not annealed, like the chips encapsulated in plastic.

The thermal features of the two ovens used for the annealing procedures have been investigated (3). They appear reproducibile. Within the heating volume, the temperature varies ±1°C in the 80°C or 100°C oven. It varies between a range of ±3°C for the 400°C oven. The value of the annealing temperature is read with a mini thermocouple fixed on the basis of the ceramic container of the chips.

Before the beginning of the test, the sensitivity S_i of each chip of each of the three groups is evaluated as the ratio of the mean readout of the chips of the group, exposed to the same value of irradiation, and the readout of the single chip.

The groups are then submitted to the same irradiation values and read through the same reading cycle, as already stated. The procedure is iterated 9 times. The sensitivity evaluation is then repeated as before the beginning of the test.

DATA PROCESSING AND RESULTS

Fig. 1 shows the S_i sensitivity values of every group as measured after the test as function of the values before the test for the same chip. The group B shows the minimum spread around the straight line with slope 1, while the unannealed group C shows the widest spread.

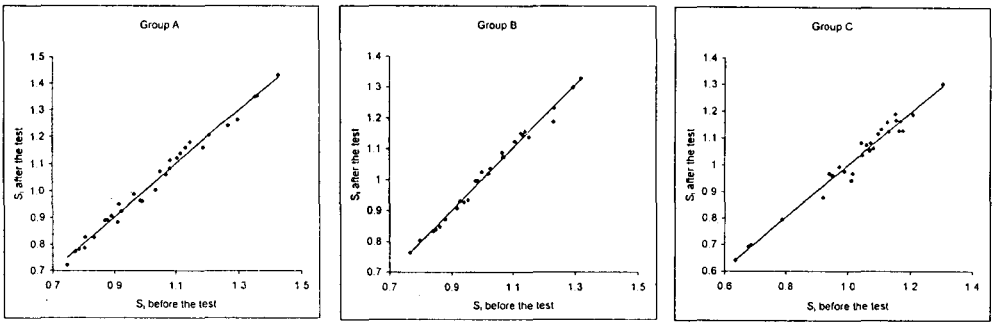

Fig. 1: S_i sensitivity values of the three groups

Fig. 2 shows the best fit obtained through a calibration set. Each dot is the average of subgroup's five detector readouts, sensitivity corrected, minus the average of sensitivity-corrected readouts of the unirradiated subgroup. The calibration factor is the reciprocal of the line' s slope. The calibration factors of the three groups are listed in Table 1 together with the related mean value and standard deviation. The mean value of the calibration factors of the group C unannealed chips is different from the mean values for the two other groups. The standard deviation shows the highest value.

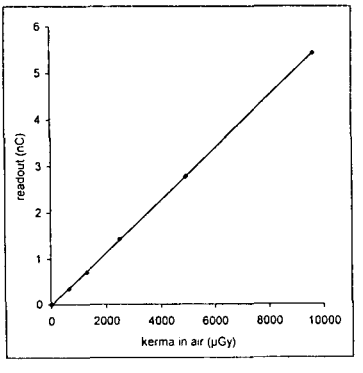

Fig. 2: Best fit of a calibration set

Table 1: Calibration factors (mGy/nC)

N. of test	Group A	Group B	Group C
1	1.80	1.74	1.18
2	1.85	1.88	1.38
3	1.87	1.90	1.39
4	1.91	1.89	1.28
5	1.89	1.90	1.31
6	1.86	1.81	1.31
7	1.91	1.92	1.28
8	1.89	1.89	1.33
9	1.96	1.95	1.52
mean	1.88	1.88	1.33
SD	0.045 (2.4%)	0.067 (3.6%)	0.094 (7.1%)

The mean values of the percent standard deviation (%SD) of each chip over 9 readouts for the same irradiation value are listed in Table 2. The Group B shows the lowest values with a low spread.

Table 2: Mean values of the %SD of each chip over 9 readouts for the same irradiation value

kerma in air (μGy)	Group A	Group B	Group C
9640	7.9	3.9	7.9
4940	3.5	2.6	5.3
2510	3.7	3.3	5.3
1300	3.1	3.3	11.3
690	5.4	3.8	7.3

CONCLUSIONS

The results show that a dosimetry service, adopting one of the two mentioned annealing routines will obtain better results than without anneling, if all other procedures are left unchanged.

Assuming the Table 2 results as an index of the variability of the reader and detector system, the group B evidences better results than group A.

Tests will continue to look for additional proof of such an evidence. Additional tests will also ascertain the reliability of the same detector, with or without annealing, until it lose its features.

ACKNOWLEDGEMENTS

Thanks to Mr. G. Minchillo for this assistance in the measurements.

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PRELIMINARY RESULTS ON SINTERED $CaSO_4$ PELLETS FOR X RADIATION DOSIMETRY USING THE TSEE TECHNIQUE

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INTRODUCTION

The increasing use of radioactive sources in medicine, industry and research has led to a growth in the number of persons exposed to some kind of ionizing radiation, therefore the importance of an accurate dosimetry. Thermally Stimulated Exoelectron Emission (TSEE), consisting of low energy electrons that are emitted from surfaces of many insulating solids at temperatures below those at which thermionic emission occurs, is a technique that can be used to measure weakly penetrating radiations, such as alpha and beta particles and low energy X-rays. The principles of exoelectron emission are very similar to those of thermoluminescence. Due to an exposure to ionizing radiation, electron traps in the energy band gap of an insulator are occupied. During the heating, the trapped electrons are released and are emitted from the surface of the crystal, if their energy exceeds the electron affinity (1,2,3). The emitted electrons are called exoelectrons and the curve of this emission as a function of temperature is called TSEE glow curve (1,3). Exoelectron dosimetry differs from thermoluminescent dosimetry in that particles (electrons) rather than photons provide the indication of dose. Due to the short range of exoelectrons, the sensitive layer of the dosemeter is thin enough that emission can be considered as a surface phenomenon (2). Usually, exoelectrons are measured with detectors counters, such as windowless Geiger-Müllers, ionization chambers or proportional counters (4). Sometimes, exoelectrons emission studies have to be performed in high vacuum conditions; in this case an electron multiplier is employed. Normally, this equipment is used preferentially for experimental research than for practical dosimetric purposes. In all devices, the samples are linearly heated up to a certain temperature. In this work the properties of pure $CaSO_4$ sintered pellets and others with 10% of graphite were investigated, using the TSEE technique, in order to verify their usefulness for dosimetric purposes.

MATERIALS AND METHODS

Powdered $CaSO_4$ crystals and chemically pure graphite powder were used to produce sintered pellets of pure $CaSO_4$ and others with 10% of graphite (6mm diameter and 0.8mm thickness). The $CaSO_4$ sintered pellets were tested in X radiation beams of a Rigaku-Denki generator, Model Geigerflex with a Philips tube Model PW/2184/00 (Tungsten target and Beryllium window - 60 kV). Prior to each irradiation, the samples were thermally treated at 300 °C for 15 min. The readout of the samples was made in a 2π windowless proportional counter with hemispherical volume and with P-10 gas flow (10% Methane + 90% Argon). The diameter of the gold anode wire is 50 μm and the operating high voltage is 2.0 kV. The samples are inserted into the counter and are fixed on a heater plate (Monel) ; they are linearly heated at a rate of 5.0 °C/s. The temperature control for linear heating is carried out by a temperature programmer (TP-2000, Theall Engineering Company), that provides rates between 0.1 and 5.0 °C/s, from room temperature up to about 400 °C. The glow curves were recorded in a multichannel scaler (EG&G - Ortec ACE-MCS SN 363 plug-in card).

RESULTS

Figure 1 shows the TSEE glow curve for a $CaSO_4$ sintered pellet irradiated with 10 Gy of X radiation (0.91 mmAl of HVL, 50 kV). The main glow peak appears at about 120 °C. Although the glow curve obtained with $CaSO_4$ + 10% of graphite is not shown, it is similar to that obtained with pure $CaSO_4$. The reproducibility of the TSEE response of the $CaSO_4$ sintered pellets was obtained measuring them 10 times after

repeated standard annealing and irradiation procedure. The standard deviation after ten readout cycles was lower than 5.0% for pure $CaSO_4$ and 11.0% for $CaSO_4$ + 10% of graphite sintered pellets.

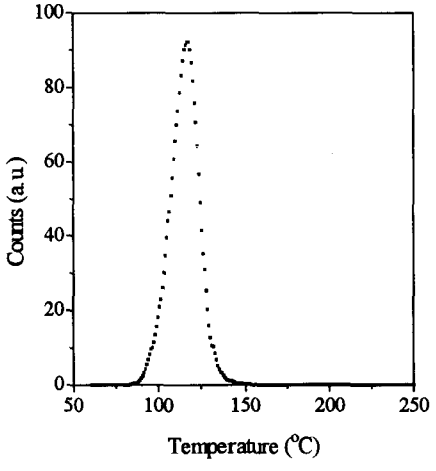

Figure 1. Typical TSEE glow curve for a pure $CaSO_4$ sintered pellet (10Gy).

The TSEE response of both kinds of materials as a function of the absorbed dose of X radiation was measured and the results are shown in Fig.2. The standard deviation of these measurements was always less than 10%. Although the observed sublinear TSEE behavior, the response increment in function of the absorbed dose show that these materials may be used for X radiation dosimetry in the range of 50 mGy up to 100 Gy.

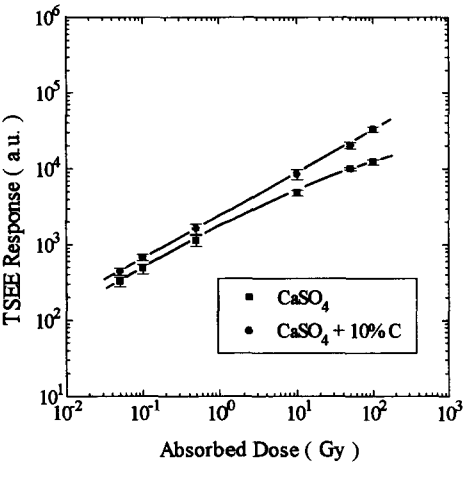

Figure 2 . The TSEE response for $CaSO_4$ sintered pellets.

The lowest detectable value was determined studying the variation of the signal obtained by the reading of non irradiated pellets. It was taken as being equal to three standard deviations from the mean zero dose reading of the pellets. The lowest detectable value was 0.5 mGy. The performance of the $CaSO_4$ and $CaSO_4$ + 10% of graphite sintered pellets was studied in relation to its energy dependence for X-rays. The

TSEE response was measured from samples exposed to 10 Gy in X radiation beams of 25, 30, 40,45 and 50 kV in air. The maximum energy dependence was reached for 19 keV of effective energy, as shown in Fig.3.

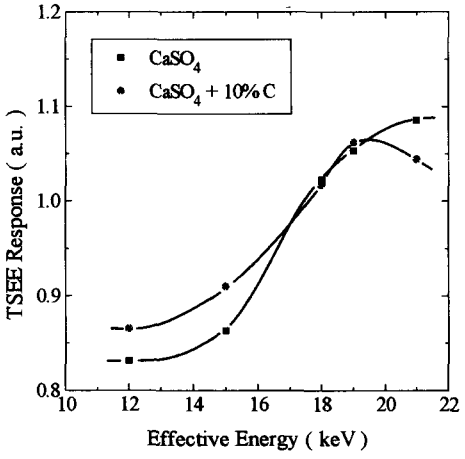

Figure 3. Energy dependence of $CaSO_4$ sintered pelllets.

CONCLUSION

The preliminary results on some dosimetric characteristics as reproducibility, glow curves ,calibration curves and energy dependence of the calcium sulphate sintered pellets studied in this work indicate that these materials may be useful for X radiation dosimetry between 0.5 mGy and 100 Gy.

ACKNOWLEDGEMENTS

The authors acknowledge the technical assistance of Mr. Marcos Xavier and the partial financial support of Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq).

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LIF THERMOLUMINESCENT DOSEMETER - TO MEASURE THE WHOLE GLOW CURVE , PEAK 5 ALONE , OR PEAK (4+5) ?

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INTRODUCTION

One of the most important characteristics of the integrating thermoluminescent (TL) detectors used as iosining radiation dosemeters is the stability of the stored TL signal. The most glaring disadvantage of the LiF (TLD-100) is its extremely complicated glow curve structure which had led to great difficulty in the arriving of universally accepted characterization of its fading properties (1,2,3,4,5,6). The routine measurement of this phosphor includes its heating to a temperature of 250-300°C. The main glow peak (and the most important for dosimetric applications) is the peak 5, vastly overlaping with peak 4.

One of the possible reasons of the great variability (especially the high fading rates for the short storage intervals) is the inclusion of the rapidly fading peaks 2 and 3 in the TL signal, when the measurement of the signal is performed by integrating the total glow curve (without peak separation). Another reason could be the application of different kind of annealings: pre-irradiation annealing of 400°C/1h and different low temperature pre- and/or post-irradiation annnealins(3). At the other extreme, the comparative low fading rates are presumably based on some combination of the peaks (4+5) or peak 5 alone (the main peaks used for routine dosimetry).

Another important parameter is the change of the sensitivity of peak 4, 5 and/or (4+5) as a function of the glow curve heating rate, as well as the cooling after the pre-irradiation annealing. It has been demonstrated that peaks 4 and 5 grow significantly, while peaks 2 and 3 decay following an intermediate temperature annealing from 150°C to 190°C (7,8).

In the present work I shall present the sensitivities of peaks 4, 5 and (4+5), separated by the computerized glow curve deconvolution (CGCD), as a function of the heating rate (from 100 to 1000°C), as well as the fading of these peaks, irradiated by gammas, alphas and thermal neutrons.

MATERIALS AND METHOD

TLD-100 (Bicron/Harshaw) hot pressed chips (3mm X 3mm X 0.9mm) were used in these investigations. The annealed materials were heated in air to 400°C/1h immediately prior to every irradiation followed by a natural "air cool" at approximately 75°C/min to room temperature. All dosimeters were calibrated individually and the annealed materials were read prior to the irradiation and calibration, so as to standardize their response. The TL glow curves were analysed using a computerized glow curve deconvolution technique (CGCD).

RESULTS AND DISCUSSION

1.Glow curve heating rate.

In table 1 the TL sensitivities of peak 4, 5 and (4+5) are tabulated for both 400°C annealed and unannealed TLD-100 as a function of glow curve heating rate from 100

to 1000°C/min. In these measurements no 100°C pre- or post- irradiation annealing was applied.

TABLE 1 : TL sensitivity as a function of the heating rate.

Heating rate	Annealed			Unannealed		
(°C/min)	4	5	(4+5)	4	5	(4+5)
100	1.78	7.09	8.87	1.92	7.14	9.06
275	2.19	8.38	10.60	2.06	7.20	9.26
430	1.93	7.42	9.35	1.45	7.57	9.02
580	2.53	6.77	9.30	2.69	6.73	9.42
800	2.62	6.43	9.05	2.78	6.43	9.21
1000	2.49	6.13	8.62	3.00	6.59	9.60
Average sens.	2.26	7.04	9.29	2.32	6.94	9.26
S.D. (%)	14	10.4	6.7	24	5.7	2.1

The followings can be concluded:

A. There is no significant difference in the TL sensitivity between the 400°C annealed and unannealed materials.

B. The integrated sensitivity of peaks (4+5) is not a function of the glow curve heating rate.

C. As the heating rate increases above 500°C/min a tendency appears, in both annealed and unannealed materials, for peak 4 to increase and peak 5 to decrease. This indicates that peak 4 and 5 are difficult to resolve accurately and consistently and this difficulty is greatly exacerbated at high heating rates as the two peaks merge more.

2. The fading of peaks 4 and 5.

The individualy calibrated TLD-100 chips were irradiated by Co-60 gamma ray source to a dose of 50 mGy, stored at dark room temperature for a period of 30 days and evaluated. The same chips were irradiated again to the same dose and evaluated immediately. The measurements were performed with annealed and unannealed materials. The same experiments were performed for alpha and thermal neutron irradiation of peaks 4, 5 and (4+5) and the fading of the whole glow curve are presented in table 2.

TABLE 2 : The changing in the sensitivity of different peaks in TLD-100.

Irradiation	Peak 4	Peak 5	Peak (4+5)	Whole glow curve
alpha (anneald)	-23.0	-5.2	-10.0	-19.9
alpha (unanneal)	-38.0	-5.2	-17.0	-27.6
neut. (annealed)	-5.6	-10.9	-9.9	-10.0
neut. (unanneal)	+43.8	-4.9	+3.3	+2.8
gamma (annealed)	+11.3	-3.9	+0.7	-10.8
gamma (unanneal)	+32.0	-2.9	+6.3	-24.5
over all rad. (both types)		-5.5±2.6	-4.4±8.3	-18.8±12.8

From the results of table 2 it can be concluded:

A. We have received a growth of peak 4 after a period of 30 days for irradiation of both annealed and unannealed gamma rays and thermal neutrons for unannealed chips. Similar results were reported by Johnson (9) and Booth (10).

B. The fading of the whole glow curve is very different for different types of irradiation (gamma rays, alphas or thermal neutrons) or different materials (annealed or unannealed).

C. No evidence of LET dependence of the whole glow curve fading was found (for example, for unannealed materials, the results of the fading alpha and gamma rays were similar, and different from the fading of thermal neutrons; for the annealed one, the results of the fading of gamma were almost identical to the fading of thermal neutrons and different from the fading of alpha).

D. A well behaved fading rate (-5.5±2.6) was obtained for the peak 5 after 30 days, for both types of materials and different irradiations.

CONCLUSIONS

In the dosimetric applications involving long storage of irradiation periods (environmental or personnel dosimetry) where fading corrections are important, the use of peak 5 alone (separated from peak 4 by using CGCD) is recomended. In other applications requiring maximum precision (clinical dosimetry), the use of the area of peak (4+5) is optimal, again using CGCD to separate the peaks (4+5) from peaks 2, 3 and 7. In all applications heating rates bellow 500°C/min are recomended to allow adequate separation to peaks 4 and 5 from their high temperature and low temperature satellites. The common practice of integrating the TL signal to include part of peaks 2, 3, 6 and 7 in dosimetric measurements is one of the main reasons underlying the non-universality in many of the reported dosimetric characteristics of TLD-100.

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TL DOSIMETRIC METHOD FOR ENERGY DISCRIMINATION IN EXTREMITY IRRADIATION MONITORING

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ABSTRACT

Radioisotopes employing in medical laboratory, nuclear medicine techniques for imaging and therapy, radioscopic investigations in surgery need extremity irradiation monitoring of workers.
Frequently, gamma - emitting radioisotopes differing for energy range are manipolated by the same person.
Extremity dose is therefore due to different radiation contribution.

INTRODUCTION

A TL dosimetric method is presented for EM energy discrimination in extremity irradiation monitoring.
The decision limit and the detection limit are defined.
The response of the dosimeter is indagated for EM energy ranging from E = 31 KeV to E = 1250 KeV, both in single and mixed fields.
The dosimetric system accuracy, tested with standard certificated radiations, is indagated like ratio between measured dose and the corresponding true one.

MATERIALS AND METHODS

The dosimeter consists of a watherproof nylon ring having density of 1.1 gr. cm^{-3} and window of 1 mm. Two detectors of different thickness and a 1 mm. Al foil between them are stacked in the case of the ring.The detectors are LiF chips, the inner having thickness of 0.9 mm., the outer 0.4 mm. The case window, the thin chip and the Al foil form the right buildup for the inner detector, from which measured doses are obtained.
Dosimetric system calibration is carried out with irradiations performed in the ENEA - Bologna secondary standard laboratory for ionizing radiations.
The chacteristic curve, obtained with a Co^{60} collimated source, is determined in the 0.4 - 48 mGy range.
The sensitivity curve is obtained with ISO - X reference raditions for calibrating dosemeters, having the following effective energy: 31 - 58 - 104 - 134 - 169 - 202 KeV. A Cs^{137} irradiation with the same value of air kerma is also used.
Dosimeter response to mixed fields is tested by several combinations of the above radiations.
The decision limit and the detection limit are defined at 95 % level of confidence, like stated by international raccomandations.
The reader is a Harshaw TLD System 4000. It provides the amount of the TL light like nCoulomb quantities.
The light emitted from the thick detector contained in the ring is used to defined the characteristic curve to obtain the dose measurement referred to the reference radiation (Co^{60}).
The ratio between the thin detector reading and the thick detector one is calculated to identify the energy of each investigated radiation field.
The energy correction factor to apply to the dose expression obtained from the characteristic curve, is derived like ratio between the thick detectors readings, the first relating to a stated energy radiation, the second to the reference one, each irradiated with the same dose.

RESULTS

The decision limit, defined as the dose level above which the measured doses with a 95 % probability can be regarded as true doses which do not belong to the background population, is 24 μGy. The detection limit, that is the dose level for which there is only a low risk (5 %) that a true dose will not be detected and that a background dose will be reported as a true dose , is 40 μGy.

To test the dosimetric system accuracy, unknown irradiation with standard certificated radiations are used to verify the condition stated by UNI raccomandations, that is $L_i \leq X_V / X_A \leq L_s$

where $L_i = 0.7 \times (1 - 2 X_o / (X_o - X_A))$, $L_s = 1.5 \times (1 + X_o / (2 X_o + X_A))$, X_V = measured dose, X_A = certified dose by standard laboratory, X_o = lower limit of dose range stated by standard laboratory (5×10^{-4} Gy).

Data analysis from tab. 1 shows a mean value of 0.849 for the ratio between the measured dose and the corresponding certified one (SD = 0.091).

Besides, each measured dose, except for the one referred to dosemeter 28, is included in the stated limits of accuracy.

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Table 1.

dosimeter	E (KeV)	X_A (mGy)	X_B (mGy)	X_B / X_A	L_i	L_s
1	37	0.802	0.728	0.9077	0.5448	1.6497
2	37	0.802	0.707	0.8815	0.5448	1.6497
3	37	1.305	1.295	0.9923	0.6004	1.5997
4	37	1.305	1.195	0.9157	0.6004	1.5997
5	37	2.164	1.921	0.8877	0.6382	1.5635
6	37	2.164	1.922	0.8882	0.6382	1.5635
7	37	3.529	3.245	0.9195	0.6614	1.5402
8	37	3.529	2.999	0.8498	0.6614	1.5402
9	37	8.915	7.918	0.8882	0.6845	1.5165
10	37	8.915	7.507	0.8421	0.6845	1.5165
11	79	0.895	0.808	0.9028	0.5593	1.6370
12	79	0.895	0.909	1.0156	0.5593	1.6370
13	79	2.642	2.341	0.8861	0.6489	1.5528
14	79	2.642	2.400	0.9084	0.6489	1.5528
15	79	4.385	3.999	0.9120	0.6688	1.5327
16	79	4.385	4.353	0.9927	0.6688	1.5327
17	79	15.410	13.201	0.8567	0.6910	1.5096
18	79	15.410	12.980	0.8423	0.6910	1.5096
19	79	26.520	22.469	0.8472	0.6947	1.5056
20	79	26.52	21.704	0.8184	0.6947	1.5056
21	1250	3.000	2.076	0.692	0.6548	1.5469
22	1250	3.000	2.275	0.7583	0.6548	1.5469
23	1250	5.000	3.591	0.7182	0.6725	1.5288
24	1250	5.000	3.526	0.7052	0.6725	1.5288
25	1250	10.000	7.355	0.7355	0.6861	1.5147
26	1250	10.000	7.136	0.7136	0.6861	1.5147
27	1250	50.000	36.059	0.7212	0.6972	1.5030
28	1250	50.000	34.609	0.6922	0.6972	1.5030
29	1250	100.000	72.112	0.7211	0.6986	1.5015
30	1250	100.000	69.956	0.6996	0.6986	1.5015
31	37 + 79	3.424	3.315	0.9682	0.6603	1.5414
32	37 + 79	3.424	3.470	1.0134	0.6603	1.5414
33	37 + 79	10.541	9.622	0.9128	0.6868	1.5140
34	37 + 79	10.541	9.600	0.9107	0.6868	1.5140
35	37 + 79	10.101	9.960	0.9860	0.6863	1.5146
36	37 + 79	10.101	9.863	0.9764	0.6863	1.5146
37	37 + 1250	6.774	5.569	0.8221	0.6796	1.5215
38	37 + 1250	6.774	5.460	0.8060	0.6796	1.5215
39	37 + 1250	9.119	7.862	0.8622	0.6848	1.5161
40	37 + 1250	9.119	7.812	0.8567	0.6848	1.5161
41	37 + 1250	12.992	11.226	0.8641	0.6893	1.5114
42	37 + 1250	12.992	9.973	0.7676	0.6893	1.5114
43	79 + 1250	8.393	6.85	0.8162	0.6835	1.5175
44	79 + 1250	8.393	6.405	0.7631	0.6835	1.5175
45	79 + 1250	8.412	7.224	0.8588	0.6836	1.5174
46	79 + 1250	8.412	7.253	0.8622	0.6836	1.5174
47	79 + 1250	19.190	14.553	0.7584	0.6927	1.5077
48	79 + 1250	19.190	16.061	0.8369	0.6927	1.5077

Perfectly Tissue-equivalent TLD Phosphor $Li_2B_4O_7(Cu,Pb)$

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The Lithium-Borate TLD phosphor shows a small sub-response below 80 KeV(1), while other commercially available tissue-equivalent phosphors such as Lithium Fluoride or Beryllium Oxide show an over-response. This indicates the possibility of a perfectly tissue-equivalent Lithium Borate by adding the proper amount of high Z elements.

We investigated several high Z materials including Ag,Sn,Zn,Cr and Pb as well as the method to add these materials to commercial $Li_2B_4O_7(Cu)$. Lead Oxide was found to be effective in adjusting the energy response of $Li_2B_4O_7(Cu)$ without damaging its original TL characteristics. In preparing samples, we first obtained $Li_2B_4O_7(Cu)$ by a sintering method(2), then the sintered phosphor was mixed with PbO and the mixture was heat treated for Pb ions to diffuse.

Figure 1 shows the glow curves of obtained $Li_2B_4O_7(Cu,Pb)$. The shape of the glow curve does not change by adding PbO, however the TL intensity gradually decreases as the amount of PbO increases. Figure 2 shows the distribution of Li. Cu and Pb ions on the surface of a $Li_2B_4O_7(Cu,Pb)$ grain observed by the secondary ion mass spectroscopy. The Pb ions distribute similarly as Cu ions, indicating that Pb ions have replaced Cu ions.

Using the new phosphor, we prepared dosimeters of the commercial structure(3). The $Li_2B_4O_7(Cu,Pb)$ grains of approx. 90μm in diameter were fixed on a plastic film to form a single layer. Figure 3 shows the energy response of the dosimeters placed on phantom. The response below 0.1 MeV increases as the amount of Pb added. The dosimeter using $Li_2B_4O_7(Cu)$ + Pb(0.28%) shows an energy response within ± 3% of the 1 cm deep dose equivalent conversion coefficient curve in the energy range 25KeV and 1.25MeV.

The excellent tissue-equivalence of the phosphor promises better accuracy in medical dosimetry as well as in personal dosimetry.

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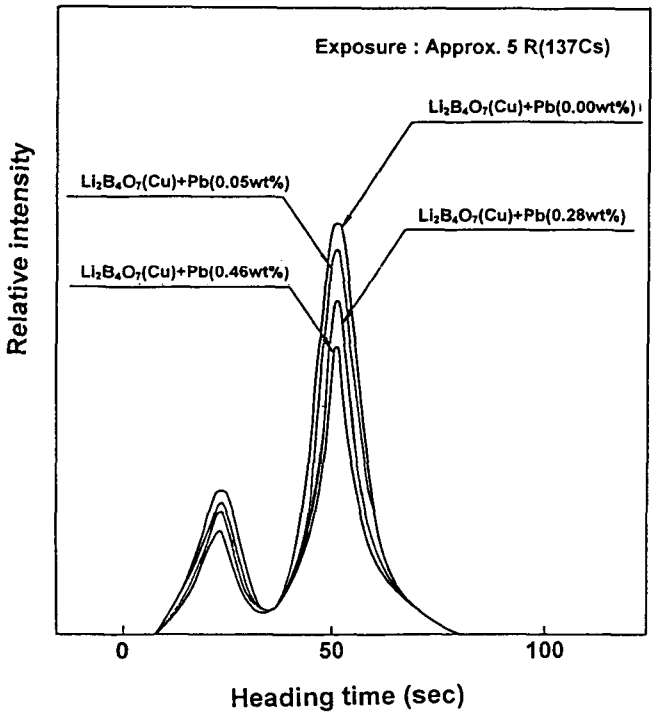

Fig.1 Glow curves of $Li_2B_4O_7$(Cu) added with Pb

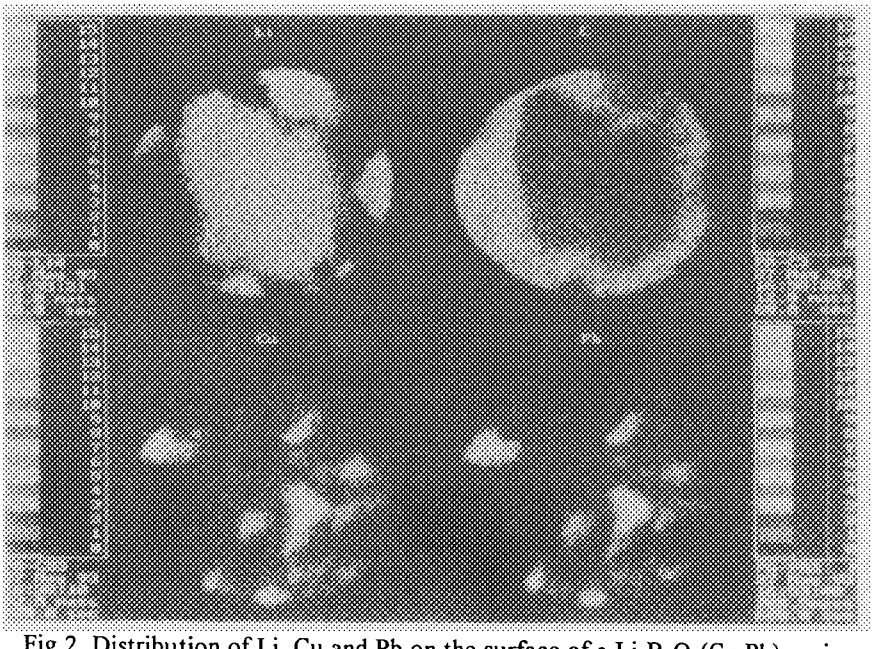

Fig.2 Distribution of Li, Cu and Pb on the surface of a $Li_2B_4O_7$(Cu,Pb) grain observed by secondary ion mass spectroscopy

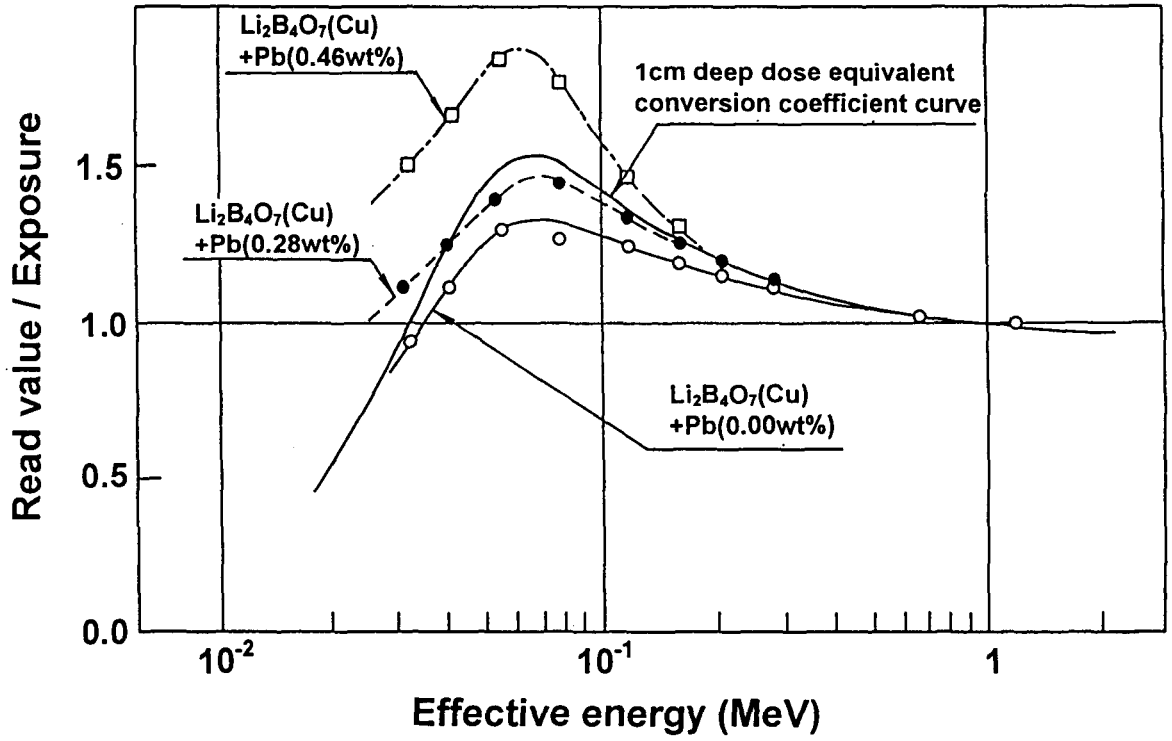

Fig.3 Energy response of $Li_2B_4O_7$(Cu,Pb) TLD located on phantom

THE INFLUENCE OF TIME - TEMPERATURE ON THE BEHAVIOR OF TLD-100 AS PERSONAL DOSEMETERS

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INTRODUCTION

Since 1965, the survey of individual exposure for the people working in the medical field is carried out within the Institute of Hygiene and Public Health, by a central service belonging to the Romanian Ministry of Health. During this period, 6500 persons per month were monitored, using film-badges. The relative low sensitivity of film dosemeters lead to a decrease of confidence in this old system. Furthermore, the film badges do not have appropriate filters for measuring the personal equivalent doses Hp(10) and Hp(0.07), as are the new recommendations of ICRP and IAEA.

These facts made clear there is a necessity of changing this type of monitoring system with another one, having a higher sensitivity and allowing the measurement of personal equivalent doses Hp(10) and Hp(0.07).

The decision was to switch to a TL dosimetry system, namely the Harshaw 6600 System [1]. The performances of this new system have to be checked by the TL Dosimetry Group, at the Institute of Hygiene and Public Health, before being officially accepted as the new individual dosimetry system.

The paper presents the results of some of the test performed on the TLD-100 chips, concerning the influence of the heating rate and pre-heat temperature and duration (the time-temperature profiles) and the fading of the detectors.

MATERIALS AND METHOD

In order to check the performances of the TLD-100 chips, we have used the Harshaw 6600 System, which is the new system meant to be used for individual dosimetry.

The Harshaw 6600 TLD System consists of a TLD reader and a computer system. The reader uses hot nitrogen, or hot dry air to read the TLD cards. The TLD cards we are using (Fig. 1) are bar-coded cards, containing two LiF: Mg, Ti (TLD-100) included in Teflon foil.

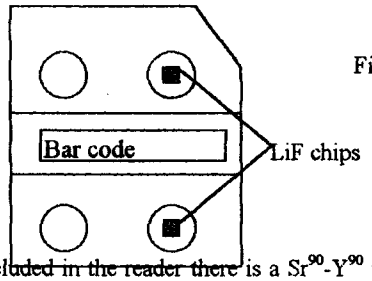

Fig. 1: Schematic of a TLD card used by the Harshaw 6600 System.

Included in the reader there is a Sr^{90}-Y^{90} irradiation source, that we used to irradiate the cards. The source was calibrated at the Secondary Standard Dosimetry Laboratory, within the IISP. A

computer system, with the TLDREMS software (also provided by Harshaw) is used to control the reader. The reproducibility of the working parameters (photomultiplier noise, high voltage, reference light, etc.) is checked at least daily, in this way assuring the reproducibility of the experiment.

The software allows the selection of different time-temperature profiles for every TL chip independently. Selection can be made on the preheat temperature and duration of preheat, on the temperature rate, on the maximum temperature, on the acquire time and on the annealing temperature and duration. Using this option from TLDREMS, we have defined six time-temperature profiles for our experiment.

Table 1: TTPs used in the experiment.

TTP1	TTP2	TTP3	TTP4	TTP5	TTP6
No preheat Temperature rate: 10°C/s Maximum temperature: 300°C Acquisition time: 13 1/3 s No annealing	No preheat Temperature rate: 25°C/s Maximum temperature: 300°C Acquisition time: 36 2/3 s No annealing	No preheat Temperature rate: 30°C/s Maximum temperature: 300°C Acquisition time: 13 1/3 s No annealing	Preheat temperature. 100°C Time of preheat: 30s Temperature rate: 25°C/s Maximum temperature: 300°C Acquisition time: 13 1/3 s No annealing	Preheat temperature: 100°C Time of preheat: 60s Temperature rate: 25°C/s Maximum temperature: 300°C Acquisition time: 13 1/3 s No annealing	Preheat temperature: 100°C Time of preheat: 120s Temperature rate: 25°C/s Maximum temperature: 300°C Acquisition time: 13 1/3 s No annealing

RESULTS AND DISCUSSION

The first three TTPs were use to determine the influence of the temperature rate on the use of the TLD cards as personal dosemeters. In our case, the peaks of interest are peaks number 3, 4, 5. (Fig. 2), the others peaks having a too low temperature to be used in this purpose, which presumes long-term (at least one month) survey.

It is well known that the Randall - Wilkins model predicts a dependence of the peak maximum position by the heating rate given by [2]:

(1) $\beta = (sk/E)T_M^2 \exp(-E/kT_M)$,

where s is the attempt to escape frequency, k is the Botzmann constant, E is the trap activation energy and T_M is the position of the peak maximum. This expression indicates that rather large changes in heating rate result in small changes in peak maximum position. The measurements we have performed indicated that the position of the peak changes with 10% +/- 3% when the heating rate changes from 10°C/s to 25°C/s, and with 5% +/- 1% when the heating rate changes from 25°C/s to 30°C/s. All measurements were performed on a batch of 50 cards.

The next three TTPs were used to determine if we can skip the period of minimum 1 week, that needs to separate the time of irradiation from the time of measurement, when the calibration of the system is performed. This time interval is needed to allow the fading of the low

temperature peak, not useful for personal dosimetry, and to realize calibration conditions as close as possible to actual personal dose measurements. We have used a batch of 50 cards irradiated with the Sr^{90}-Y^{90}, and read out immediately after irradiation. The glow curves were compared with the reading of other 50 TLD cards, performed 2 weeks after irradiation.

The results indicated that, in order to obtain similar readings with the readings performed after 2 weeks, a longer annealing time is needed, than the times we have used on the reader. For example, peak number 2 had an intensity about 5 times smaller when annealed at 100°C, for 120s, than when read out immediately after irradiation. However, the same peak had an intensity about 10 times lower when read out two weeks after irradiation, which means the annealing should be done for about 1 hour, at 100°C.

A batch of 100 cards was irradiated and read out at different times after the irradiation (immediately, one day after, two days, then at two days interval until one month was achieved). The fading of the TL signal is presented in Fig. 2.

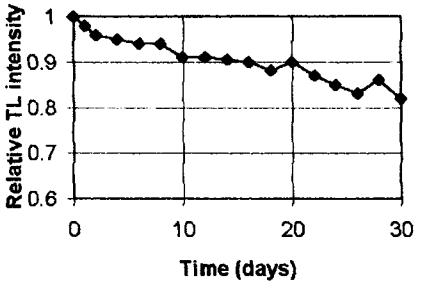

Fig. 2: Fading of TL chips used by Harshaw 6600

The relative intensity was calculated by respect to the read out immediately after the irradiation. As it can be seen, the fading is very low (less than 20% for the total area of the glow curve). Furthermore, the fading of the peaks of interest for personal dosimetry is even less important (only 3% for peaks 3, 4 and 5).

CONCLUSIONS

The measurements presented here allowed us to conclude that Harshaw 6600 is a very reliable system for personnel dosimetry.

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GRAPHITE MIXED $CaSO_4$:Dy FOR BETA DOSE MEASUREMENT

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INTRODUCTION

Dosimetry of weakly penetrating radiation, beta and photon radiation below 15 keV, has got more attention only during the last 20 years when a variety of methods for skin dosimetry were developed. One of the main difficulty in beta dosimetry is the energy independent dose determination. Dosemeters have been developed may be characterized as: mechanically thin TL detectors, surface sensitive elements, multi-element constructions and application of special evaluation regimes and/or glow curve analysis (1). Surface sensitive dosemeters include: TSEE detectors; reduction of the light transparency of thick detectors by adding graphite or other materials; creating new TL traps near the detector surface by diffusion of boron into LiF or Li_2 or $Li_2B_4O_7$. Good results have been achieved with carbon loaded LiF, $Li_2B_4O_7$ and MgB_4O_7 (2,3,4,5). Detectors loaded with carbon in a range of 11-20 % C have an effective thickness of a few $mg \cdot cm^{-2}$ therefore with an appropriate window thickness, 1-5 $mg \cdot cm^{-2}$, nearly energy independent beta dose measurement is possible even from $E_{\beta eff}$ 70 keV.

Lower sensitivity of thinner effective sensitive layer may be compensated by using high sensitivity TL material such as LiF; MCP (6) Present work presents results on graphite-mixed $CaSO_4$:Dy.

MATERIALS AND METHODS

Sintered $CaSO_4$:Dy pellets were prepared of 63-125 μm grains (7). These detectors have very good dosimetric properties: the main dosimetric peak exhibits at about 220°C, linear dose range is 10 Gy, no need for annealing at low doses, batch uniformity is letter than 2.3 %, reproducibility is better than 2 %. Environmental factors do not affect the detectors, no fading of the main dosimetric peak during a five months period is observed. The wavelengths of emission peaks are 478 nm and 550 nm, the relative TL response normalized to TLD-100 is 12-49 depending on the TL reader used for read-out. Detectors of $CaSO_4$:Dy were produced with different carbon loading, 5-10 % C. TL materials such as LiF:Mg,Cu,P IPN, Krakow, LiF (TLD-100), cold pressed LiF (0.85 % C) IKI, Budapest were used for comparison. BeO, Battelle Institute, Frankfurt and cold pressed LiF (C) were used for TSEE measurements.

Irradiations were carried out with newly designed nearly tissue equivalent beta sources. Beta dose-rates were measured with extrapolation chamber and flat ionization chambers. Irradiations were carried out at a 'contact' geometry, source to detectors distance was 7 mm, and 6.5 $mg \cdot cm^{-2}$ thick Mylar foil covered the detectors.

Irradiated TL detectors were analyzed by a Daybreak TL reader utilizing photon counting and the applied heating rate was $3°C \cdot s^{-1}$. TSEE measurements were carried out with reader developed at Fontenay-aux Roses, France having a multi needle counter.

RESULTS AND DISCUSSION

Detectors type $CaSO_4$ (C) used in measurements were selected from a larger batch and their sensitivities did not differ more than 3 % from the mean value.

Relative TL responses of pure, graphite-mixed $CaSO_4$:Dy detectors are given in Table 1 and in Fig. 1 and for the other TL detectors in Fig. 2.

Table 2. Relative TL responses of $CaSO_4$:Dy (C) detectors normalized to ^{90}Sr-^{90}Y irradiation

Detector	$CaSO_4$ pure	$CaSO_4$ 5 % C	8 % C	10 % C
^{147}Pm	10.5	36	48.5	49.7
^{204}Tl	58.3	69.6	72.5	81.1
^{90}Sr-^{90}Y	100	100	100	100
^{106}Ru-^{106}Rh	140	98	86	86

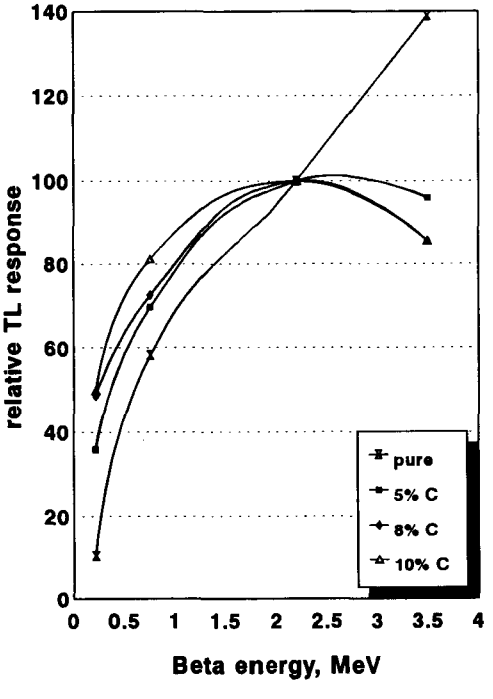

Fig.1. Energy dependence of TL responses of $CaSO_4$:Dy (C) detectors

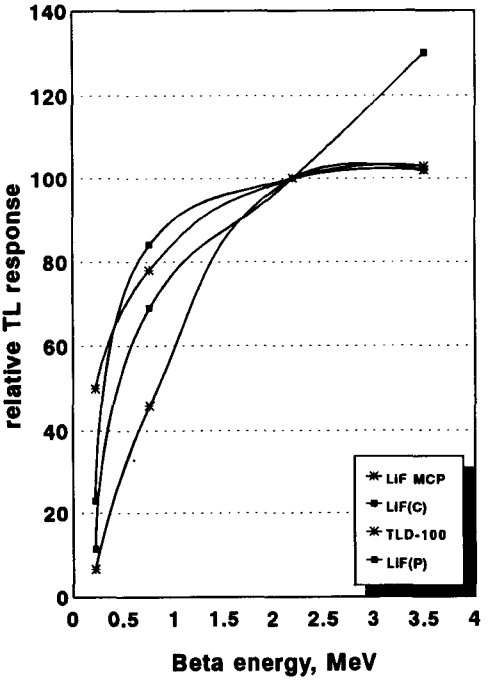

Fig.2. Energy dependence of TL responses of different LiF detectors

Sensitivities of different dosemeters were determined, see Table 2.

Table 2. Sensitivities of TL and TSEE methods using different materials

Method	Dosemeter	Lower detection limit, μSv
TSEE	BeO	2.5
	LiF (0.85 % C)	3000
TL	LiF; Mg,Cu,P	15
	LiF (0.85 % C)	300
	LiF (p) Krakow*	80
	TLD-100*	150
	$CaSO_4$:Dy pure*	10
	$CaSO_4$:Dy (5 % C)	80
	$CaSO_4$:Dy (8 % C)	200
	$CaSO_4$:Dy (10 % C)	250

Lower detection limits is set to 3 σ of the second reading; integration: the main dosimetric peak only.
*Sensitivities are valid only for higher $E_\beta \geq 0.5$ MeV radiations due to strong energy dependence of TL responses.

Results of $CaSO_4$:Dy (C) detector measurements have shown: this detectors are suitable for beta dosimetry, their sensitivity is high enough, energy dependence of TL responses are nearly flat for a wide energy range if an entrance window about 3 $mg \cdot cm^{-2}$ thick is used. The general excellent dosimetric properties of investigated $CaSO_4$:Dy (C) detectors make them a good candidate for beta dosimetry especially for pure beta radiation field measurement where their overresponses for low gamma radiation does not appear.

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IRPA9
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PAPER TITLE CHARACTERIZATION OF THE NEW HARSHAW 4500 TLD READER

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

The Harshaw Model 4500 Manual TLD Reader is a newly designed instrument that is unique in its ability to read all forms of Harshaw thermoluminescent (TL) material: cards, Chipstrates™, chips, rods, microcubes, disks, and powder. The unit uses a sliding, dual-tube PMT Assembly to simultaneously read two positions of a TLD card or two Chipstrates, or to read a single unmounted dosimeter. Mounted dosimeters are heated with a hot gas medium to a maximum of 400°C. Unmounted dosimeters are heated by conduction on a planchet heated by electrical resistance to a maximum of 600°C. In both cases, the temperature is held precisely to a linear time-temperature profile by an electronic, closed loop feedback system. This paper reports the complete characterization of the reader in both heating modes, using IEC standards as a guide. Superior results were achieved and are reported in the following categories: detection threshold, reproducibility, ambient light sensitivity, climatic sensitivity, residual signal, system stability, linearity, test light stability, and power leakage.

A NEW APPROACH TO SCREENING ABNORMAL GLOW CURVES

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ABSTRACT

A simple computer program specifically developed to electronically screen abnormal LiF:Mg,Ti glow curves in routine dosimetry is presented. The method identifies glow curves with abnormal patterns by examining a few key features from which information about background and thermoluminescent (TL) signals can be extracted. The underlying criteria for glow curve acceptance/rejection are discussed. By analyzing the records of Quality Control (QC) cards that are in the same group as the field cards, the dosimeter- and reader-dictated parameters (such as the peak location) are automatically determined, thus eliminating the need for any prior knowledge of those conditions required for reference determination. The program performance was evaluated by using actual glow curve data provided by users of commercially available readers and dosimeters in routinely high-throughput operations. The initial tests showed a near 100% success rate in identifying abnormal glow curves (no false positive results) and a minimal rate of rejection of normal glow curves. Results have shown that this method is applicable for dose levels down to 50 μSv.

INTRODUCTION

As radiation monitoring and protection becomes an increasingly important issue, the growth in the scale of routine dosimetry is inevitable. A well established and widely employed dosimetry technique is thermoluminescence dosimetry (TLD) using LiF as the host material[1,2]. Dose information is generally extracted from the integral area under the glow curve or part of the glow curve. Errors in such measurements are inevitable due mainly to the existence of background and to the complex nature of the fast-fading glow peaks. Much effort has been devoted to improving the accuracy of dose measurement by subtracting background signal and by eliminating contributions from the fast-fading glow peaks[3-5].

In routine dosimetry where accuracy is not a critical issue, dosimetric information is approximated by the total area under the glow curve. The validity of such practice is ensured by carefully examining each individual glow curve and eliminating spurious readings before they can be used in dose algorithms[6]. Currently glow curve screening is done manually. This is not only time consuming, but the pass/fail decision making is highly subjective. The existing analysis programs cannot help change this situation because many features of these methods are not compatible with the practical requirements of routine dosimetry.

For the now-apparent reasons we have developed a simple program for routine personnel dosimetry to electronically analyze the shape of LiF:Mg,Ti glow curves. Rather than separating individual glow peaks, this method automatically recognizes irregular glow curves by identifying a few key glow curve features and testing them against a set of empirical criteria. Emphasis is placed on complete rejection of faulty glow curves and minimal rejection of good glow curves. Some specific faulty features in glow curve shape are:

1) saturation of readout
2) spikes caused by electrical interference
3) lack of a glow peak
4) glow peak occurs at unexpected temperature
5) distortion caused by non-radiation induced signal
6) high IR tail or high residual signal
7) end of acquisition before completion of glow curve.

GLOW CURVE STANDARDS

The intrinsic glow curve characteristics — the location and width of the main glow peak — are adversely affected by a host of factors such as the chip thickness, electronics of the card-reading instrument, and the heating profile used for readout. If all cards in a group are of the same type and are read in a relatively short time frame (say, one day) by the same reader, we can reasonably expect these glow curves to exhibit close similarity in shape. The established values of these parameters are defined as 'glow curve standards'.

Because the standards are largely determined by factors that are different from user to user, it is impossible to arrive at a universal set of parameters that are applicable to all users. This difficulty, however, can be overcome by utilizing the Quality Control (QC) cards that were interspersed in groups of field cards during card reading. The values of the intrinsic parameters (with reasonable tolerances) for field glow curves

can be predicted from the corresponding aspects of the QC curves. This approach enables the determination of glow curve standards without any prior knowledge of the aforementioned conditions.

GLOW CURVE ASSESSMENT STEPS

The glow curve screening program proceeds through the following logical decision tree to determine which glow curves are likely to have faulty features and need visual inspection.

1. Saturation Test. Check for readout saturation by testing for a flat top in the TL spectrum.

2. Curve smoothing. Outlier points must first be removed from the initial spectrum. An outlier is a discrete point on the curve which, judging from its amplitude, is likely to arise from instrument malfunction rather than from statistical fluctuation. Specifically, an outlier y_i satisfies the condition:

$$\frac{|y_i - y^0{}_i|}{y_{max}} > 0.1$$

where

$$y^0{}_i = 0.5 * (y_{i-1} + y_{i+1})$$

is the corresponding average of the immediate neighboring channels. An outlier is removed by replacing y_i with $y^0{}_i$. The weighted averaging method is used to filter out glow curve noise caused by the dosimeter and by instrument electronics. The weighting factors are chosen such that a respectable smoothing is achieved without altering the original features of the glow curve.

3. Glow curve noise level. The noise level of a glow curve is characterized by the number of outlier points and by the extent of the TL scatter. The latter is estimated by comparing the spectra before and after smoothing. If the estimated noise level exceeds a pre-set threshold, the glow curve is rejected.

4. Peak finding and Peak 5 identification. Peaks are automatically found by searching for local maximum, which is defined as the highest points in a region of ±10 channels from a putative apex. The main dosimetric peak — peak 5 — is identified as the one closest to the expected peak location (as defined by the standard) and lies within a certain tolerance window.

5. Estimate various regions of interest. The glow curve from an irradiated dosimeter upon linear heating can be divided into three regions, where background and dosimetric information can be extracted.

   Region I: Low temperature region (peaks 1 and 2). In routine personnel dosimetry, peaks 1 and 2 are generally invisible due to long fade time. Consequently, this region should be dominated by the photomultiplier tube dark current. Excess signal in this region indicates the possibility of non-radiation induced signal, in which case the glow curve should be rejected.

   Region II: Main dosimetric peak region. This is the region of dosimetric interest and usually includes peaks 3, 4 and 5.

   Region III: High temperature region. This region is dominated by the temperature-dependent background, due mainly to infrared radiation (IR) by hot TL element and detector parts. The relative size of the IR radiation to the actual dosimetric signal is estimated by the ratio of the area under the curve for the last few channels over the area of Region II. An internal upper limit for this ratio has been set to reject those glow curves with high IR tails.

6. Determine the inflection point on the high-temperature side of Peak 5. This point is found either by searching for a local minimum immediately to the right of Peak 5, or by utilizing the curve slope information. If this point cannot be found, the glow curve should be rejected as incomplete.

7. Baseline fitting and background subtraction. For a complete glow curve, the baseline is fitted to an exponential function:

$$y = \exp(a + bx),$$

where x is the channel number, y is the background level at channel x, a and b are parameters to be

extracted from the fit. The sampling points used in the fit are the first N channels with constant background and the points in Region III. Figure 1 shows a smoothed glow curve and a superimposed background curve. The background signal can be calculated by integrating the area under the fitted background curve. If the ratio of background signal to the total signal is greater than the limit pre-specified by the user, the glow curve will be rejected.

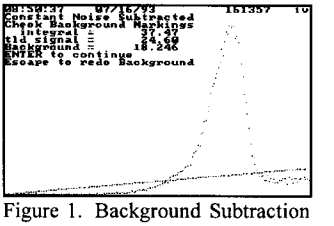

Figure 1. Background Subtraction

8. Estimate the width of the main glow peak. The glow peak width is measured by its full width at half maximum (FWHM) height. A maximum acceptable value of FWHM is determined based on the information obtained from the QC card records in the same group. A very wide glow curve structure usually indicates poor heat transfer and hence should be rejected.

PROGRAM OUTPUT

If a glow curve is rejected by the algorithm as a result of a failure from the evaluation criteria, an online message is displayed stating the dosimeter ID and reason for rejection. Upon completion of the program a summary report is created including the user-specified values of glow curve acceptance criteria, the number of dosimeters processed and a list of all glow curves rejected along with reasons for rejection.

PERFORMANCE EVALUATION

The method described herein has been tested with some 2000 glow curves provided by users of Harshaw/Bicron Model 8800 Automatic TLD Readers. The current algorithms demonstrated a success rate of nearly 100% in rejecting problematic glow curves. While having certain irregularities, a small fraction of the rejected glow curves may be deemed by some users as acceptable for dose evaluation. Figures 2 through 4 show some of the glow curves rejected by this program.

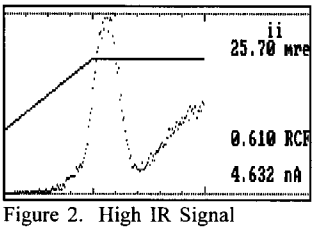

Figure 2. High IR Signal

Naturally this method is more effective for high doses than for low doses. Nonetheless, it is effective down to the level of 50 μSv. Bearing in mind that the purpose of this program is to avoid erroneous evaluation and reporting of radiation received by the dosimeter wearer, the significance of this program is of less concern in analyzing glow curves that reflect only background radiation.

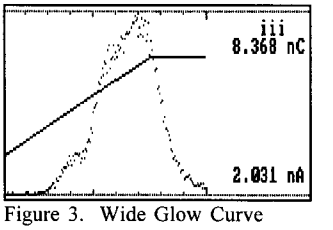

Figure 3. Wide Glow Curve

CONCLUSION

A simple computer program has been developed to screen abnormal LiF:Mg,Ti glow curves in routine dosimetry. Extremely encouraging results were obtained from the initial tests. More careful and extensive tests are suggested to thoroughly evaluate the current algorithms. Continuous efforts are anticipated to improve the program's functionality. It is worth pointing out that although this method was developed specifically for analyzing LiF glow curves, it can be modified with reasonable ease to extend its scope to include glow curves from other types of TL materials.

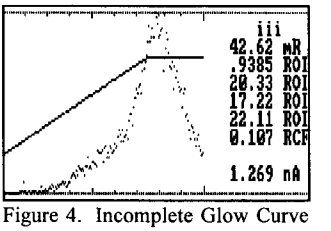

Figure 4. Incomplete Glow Curve

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PAPER TITLE THE INFLUENCE OF TL DOSIMETRY SYSTEMS ON SOME HIGH SENSITIVE SOLID TL DETECTORS

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

ABSTRACT (See instructions overleaf)

The expansion of the number of TL materials currently available that exhibit ultra high sensitivity, such as LiF:Mg,Cu,P (GR-200A),$CaSO_4$:Dy (TLD-2000H), $CaSO_4$:Tm (TLD-2001H), and Al_2O_3:C (TLD-500K), make it important to study and compare their individual TL properties under the optimal measurement procedures.

The behaviour of hypersensitive thermoluminescent phosphors in form of solid TL detectors has been studied under two basically types of TL Dosimetry Systems based on different heating methods, planchet , and hot nitrogen gas heating. Also, the different types of TLD systems may differ markedly in their spectral light detection responses, i.e. the spectral response of photomultiplier depends on the composition of the photocathode, on the special transmission of the tube window, as well as on optical filter built in. Since the spectral emission of the various TL materials spans from 300nm to 600nm, the investigation of the influence of the measurement conditions on the main characteristics of the investigated phosphors is also analyzed.

This study shows that the main characteristics of examined TL materials including sensitivity, reproducibility, linearity, minimum detectability repeatibility, residual dose, ambient light sensitivity, stability of glow curve shape, fading, dependence on the heating rate,etc., support the importance of achieving compatibility between the TL material and the type of Thermoluminescence Dosimetry System used.

A new laser thermoluminescence dosimetry system for the determination of the personal dose equivalent $H_p(10)$ and the irradiation conditions

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Introduction

For the purpose of radiation protection additional information about the irradiation conditions may be important if a personal dose above a specified threshold is detected. The objective of the development of a new personal dosemeter therefore is to determine the personal dose equivalent with the necessary accuracy and, also, to provide infomation about the irradiation conditions. To fulfil these requirements, a new laser-heated TL dosemeter (TLD) system, commercially available and produced by International Sensor Technology Inc. has been modified. A new algorithm based on linear programming (1, 2) is applied to improve the dose determination. A TL-sheet is inserted into the dosemeter badge to allow information to be obtained about the irradiation conditions.

Laser TLD system

The unique feature of the dosemeter (Figure 1) used in the laser TLD system is a $CaSO_4$:Tm imaging sheet (image area: 9 mm × 15 mm). The sheet determines the direction of incident radiation by the "shadow" projected by two metal plugs (ø: 1,5 mm, length: 3,8 mm, lead alloy), one arranged above and the other below the sheet. Four $CaSO_4$:Tm elements (3,5 mm × 3,5 mm) (3), each with different filtering, are used to calculate the personal dose equivalent $H_p(10)$, including an estimate of the mean energy of the incident photon radiation. The following four filters are used: plastic material 1,17 mm and 3,30 mm thick, tin 1,40 mm thick and stainless steel 0,25 mm thick.

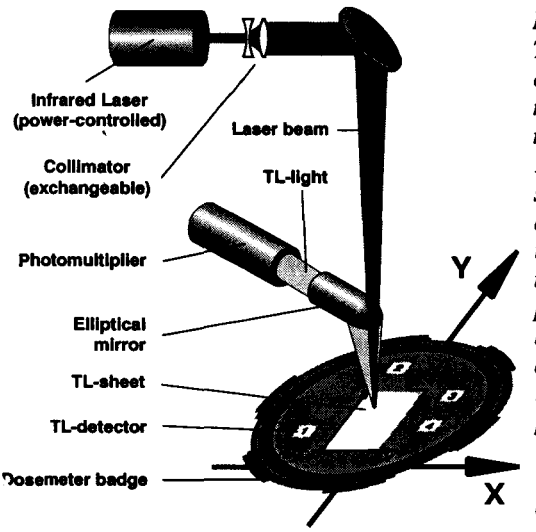

Figure 1. Schematic diagram of the laser TLD system. For evaluation of the TL elements and the sheet, the dosemeter is moved in an x-y plane to place each element in the beam of a power-controlled 10 W CW CO_2 laser (4, 5). A shutter (not shown) is opened for element reading and annealing only. Two types of beam-shaping optics (collimators) are automatically moved into the laser beam. The first type produces a 4 mm × 4 mm uniform heating beam used to read the single elements and anneal both the single elements and the imaging sheet (4, 5). The second type produces a small spot for reading the imaging element with a high spatial resolution (0,3 mm in both the x and the y direction).

Determination of the personal dose equivalent $H_p(10)$

All TL-elements of the dosemeter are calibrated using gamma radiation of ^{137}Cs. The dosemeter is read to determine the personal dose equivalent $H_p(10)$ over an energy range from 17 keV to 1250 keV and over an angular range from 0° to 60° (Figure 2). For dose determinations, information about irradiation conditions is not necessary, because an algorithm based on linear programming is used (1, 2). The algorithm determines the lowest (H_{min}) and the highest (H_{max}) dose values compatible with the signals of the four single elements. The geometric mean of the extreme dose values, H_{min} and H_{max}, is taken as the dose value M_p of the dosemeter. The response matrix (1, 2) used for the calculations consists of the four single element responses to X-ray radiation of the ISO narrow spectrum series (N-30 to N-300) and to gamma rays of ^{137}Cs and ^{60}Co and angles of 0°, 30°, 45° and 60° on an ISO water slab phantom. The maximum difference between the irradiated dose and the dose measured for $H_p(10)$ by this dosemeter determined with linear programming is less than ± 16 % for mixed photon fields in an energy range of 26 keV to 1250 keV and an angular range of 0° to 60° (2).

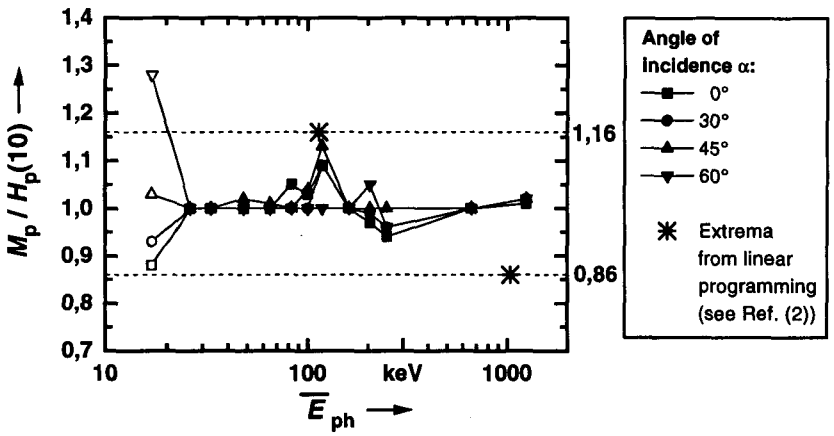

Figure 2. *Dosemeter response to $H_p(10)$ using a linear programming algorithm to calculate the dose based on the responses of the four single elements. The radiation energy range is obtained with X-rays of the ISO narrow spectrum series (N-20 to N-300) and gamma rays of ^{137}Cs and ^{60}Co. The open symbols denote irradiation conditions outside the range defined by the response matrix. The dashed lines indicate a deviation of the response of + 16% / – 14 % from unity. The small angular dependence is a result of specially designed filters. $\overline{E}_{ph}$ is the mean photon energy.*

Direction of incident radiation

The direction of radiation incidence is determined by scanning a $CaSO_4$ imaging element when the measured personal dose equivalent exceeds a specified threshold. The collimator in the laser beam path will scan the sheet with a 0,3 mm spatial resolution (both in x and y direction). The sheet is annealed by scanning with the large (4 mm × 4 mm) heating beam. The imaging sheet has sufficient sensitivity to provide incident radiation direction information for X-ray (30 kV - 300 kV tube potential) doses higher than or equal to 0,2 mSv.

The direction of incident radiation is inferred from the "shadow" projected by the two lead alloy plugs (one arranged above and the other below the sheet) onto the sheet. If the incident radiation is perpendicular to the sheet, a circular shadow is produced on the image. Non-perpendicular irradiation produces an oval shadow (Figure 3). The two plugs are staggered to distinguish radiation incident from the front or back of the dosemeter.

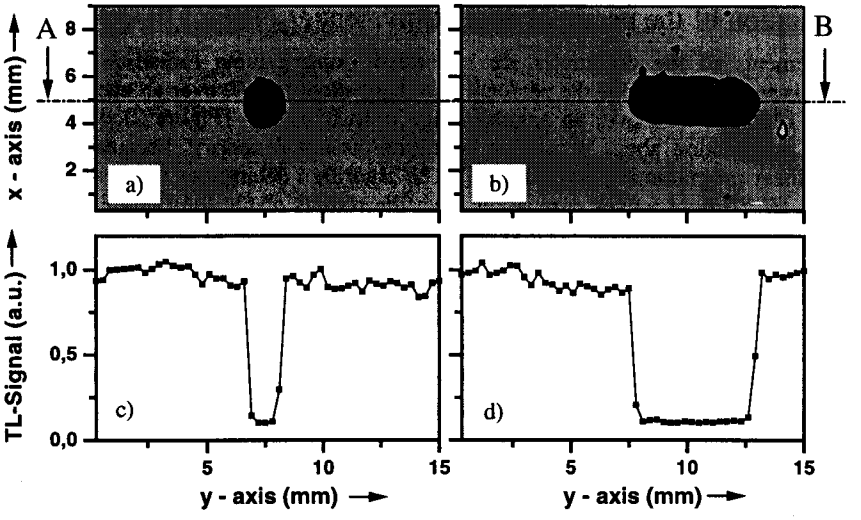

Figure 3. *Contour (a, b) and cross section (c, d) plots of the dose distribution of a $CaSO_4$ imaging sheet (9 mm × 15 mm) after irradiation with 1 mSv 60 kV X-rays. The contour plots show that the shadow of the metal plug is circular (a) and oval (b), when irradiation takes place perpendicular or at 45°, respectively, to the imaging sheet. The cross section plots (c, d) show the dose distribution along the line AB as shown in figures a and b.*

Conclusions

The laser-heated TLD system has a unique 2-dimensional dose image based on TLD, which provides information about the direction of incident irradiation. The system also allows the personal dose equivalent $H_p(10)$ to be determined with sufficient accuracy for a wide range of the photon energy and the angle of incidence. Besides the features of the system shown here, it also allows the personal dose equivalent H_X to be determined and the mean energy of the incident photon field(s) to be estimated.

Acknowledgement

The authors thank P. Kragh for making his dose calculation software "LINOP" (6) available to them. This work was supported by the Minister of the Environment, Nature Conservation and Reactor Safety of the Federal Republic of Germany. The views and opinions expressed herein are those of the authors and do not necessarily represent those of the Minister.

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* formerly: International Sensor Technology, Inc.

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APPLICATION OF RADIATION-INDUCED DOPING OF POLYANILINE FILM TO MEASURE INTEGRATED GAMMA-RAY DOSE

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INTRODUCTION

For safety operation of high energy accelerators, radiation dose measurement for accelerator components is very important since severe damages of electric parts, such as a breakdown of insulation and change of electric properties, take place if they receive high radiation doses. A dosimeter, which can measure integrated radiation doses up to at least 1 M Gy, is needed for this purpose. There are a couple of simple devices applicable to the measurement for a wide range of integrated doses; the dosimeters using conducting polymers are promising candidates for the measurements in high radiation fields.

Conducting polymers, represented by polyacetylene (1, 2) and polyaniline (3), are known to exhibit drastic increases in their electrical conductivities from insulator to conductor by doping. The device consisting of a conducting polymer and a source material of a dopant can be used as a dosimeter (4, 5). A dopant gas is produced through the radiation decomposition of the source material, and the polymer is doped with the released gas. In this dosimeter, radiation doses can be easily measured as an increase in conductivity of the polymers with a simple resistance meter. We applied this method to polyaniline systems, and the basic characteristics of the dosimeter using polyvinylchrolide powder (PVC) as the source material were reported in the previous paper (6).

In this study, the new system of a polyaniline and SF_6 gas was adopted as a dosimeter, and the conductivity change by radiation-induced doping was examined extensively. The relation between the integrated radiation dose and the conductivity change is described.

EXPERIMENTAL

1) Samples The polyaniline used in this work was of the emeraldine base form which was chemically synthesized from a 1 M perchloric acid solution of aniline. The polyaniline film was prepared by gently drying the solution of polyaniline on a glass plate placed horizontally. *N*-methyl-2-pyrrolidone was used as the solvent, and was evaporated completely at 80 °C under vacuum for 1 day.

The polyaniline system consisted of a polyaniline strip (0.5 × 4 cm, *ca.* 40 μm thick) and SF_6 gas (purity: 99.99%), which were sealed in a Pyrex glass tube (volume: *ca.* 2.5 cm^3). The pressure of SF_6 gas was 0.1 or 1 atom.

2) Irradiation The samples were irradiated with ^{60}Co γ-rays at room temperature. This irradiation was carried out at the ^{60}Co irradiation facilities of Japan Atomic Energy Research Institute. During the irradiation, the radiation-induced doping proceeds in the glass tube.

3) Measurement of Electrical Conductivity The resistance of the polyaniline film was measured with an ultra-high resistance meter (Advantest Co., Model R8340A) in a pure nitrogen atmosphere at room temperature. After γ-ray irradiation, the glass ampoule of the sample was open, and the resistance of the irradiated strip was measured.

4) Determination of Fluorine in the Polyaniline In order to examine the relationship between the conductivity change and the fluorine content of the strip, the content was determined by neutron activation analysis. A piece of the strip after the radiation-induced doping was irradiated by neutrons in the PN-3 pneumatic tube of the JRR3 reactor in Japan Atomic Energy Research Institute. The repeated irradiation technique was applied to the samples with low content of fluorine to obtain high precision of determination. The intensity of the 1633-keV γ-ray of ^{20}F (half life: 11 sec) formed by the (n, γ) reaction of ^{19}F was measured with a Ge semiconductor detector system. The fluorine content was determined from its γ-ray intensity, compared with reference samples.

RESULTS AND DISCUSSION

The volume conductivity of the undoped polyaniline film prepared was 1×10^{-13} S/cm and the surface conductivity was 1×10^{-13} S. The film prepared (the emeraldine base form) is very stable to radiation. The conductivity of the film decreased by only one half of its original conductivity even for ^{60}Co γ-ray irradiation up to 3.6 M Gy under vacuum. This change is negligibly small compared to an increase in conductivity by doping mentioned below.

Figure 1 shows the conductivity change of irradiated polyaniline strips with different pressure of SF_6 gas (0.1 atm and 1 atom). The abscissa represents the absorbed dose for the polyaniline film, and the ordinate shows the conductivity ratio of irradiated to non-irradiated polyaniline. The conductivity increasesd almost linearly on a log–log plot along with an increase in the dose over a wide region of about 10^3 to 1 M Gy; the maximun conductivity reached 10^8 times its original value above 1 M Gy. As shown in Fig. 1, the threshold dose can be controlled by changing the pressure of SF_6 gas.

The surface of the polyaniline film was doped more homogeneously in this polyaniline–SF_6 system than in the polyaniline–PVC system (6) where a heavy dopant gas (HCl) is produced by radiolysis in the solid source material. This indicates that gases are much suitable for the source material of the dopant.

Figure 2 shows a relationship between the content of fluorine (μg/cm^2) introduced on the surface of the polyaniline film through the radiation-induced doping and its relative conductivity. The content was obtained by subtracting the fluorine content originally contained in the undoped polyaniline film from the total content in the doped polyaniline obtained by the neutron activation analysis. The original content was determined to be 1.4 μg per 1 cm^2 of the film with the same method as the doped polyaniline films.

The conductivity change through doping is very influenced by species of the dopants and their amounts on the surface of conducting polymers. The concentration of the dopant gas depends on the G-value of the dopant gas liberated from the source material, the amount of the source material and the integrated radiation dose.

The linear increase range in this SF_6 system (Fig. 1) was narrower than that of the polyaniline–PVC system (6); however, the increase in the SF_6 system continued to *ca.* 1 M Gy, while that reached saturation at 0.1 M Gy in the PVC system. This SF_6 system is, therfore, considered to be more suitable for a dosimeter which is used in high radiation fields. A prototype dosimeter

with electrodes will be discussed.

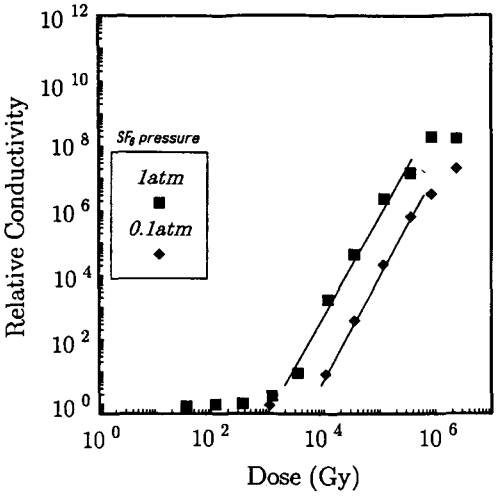

Fig.1 Relationship between the dose and conductivity change of the polyaniline film

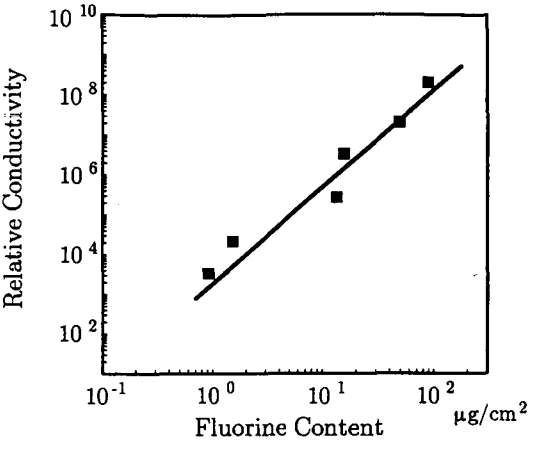

Fig.2 Relationship between the fluorine content of the polyaniline film and the conductivity

ACKNOWLEDGEMENTS

The authors wish to express their appreciation to Dr. Toshihiro Ohnishi and Mr. Shuji Doi of Sumitomo Chemical Co. for supplying the polyaniline powder. They are also grateful to the members of the Inter-University Laboratory for the Common Use of Japan Atomic Energy Research Institute Facilities for their assistance in γ-irradiation.

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THE FEATURES OF RADIATION DAMAGES IN L-ALANINE CRYSTALS

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INTRODUCTION

The method of the ESR alanine dosimetry has appeared the most convenient one for measurement of radiation dose in the range $1\text{-}10^6$ Gy (1). Its peculiarities are the wide dose range, the high accuracy, the absence fading at room temperature, the possibility of many times repeated measurements as dosimeter accumulates dose, the simplicity of measurements.

Because of this performance ESR alanine dosimetry technique can be applied to continuous monitoring radiation doses absorbed by materials on nuclear power stations as well as of dose fields and restoration doses after an accident situation. In order to determine accurately the absorbed dose in an accident on background of accumulated dose for previous period, it is necessary to the utmost increase the accuracy of dosimetry system.

For this reason it is necessary to know how the properties of free radicals which formings in irradiated L-alanine are displayed in signal ESR.

PROCEDURE

With the purpose to detect the structure of the free radicals the ESR spectra the L-alanine and L-alanine-d_3 single crystals were studied. The samples were grown by slow evaporation of the saturated aqueous solution. For obtain the L-alanine-d_3 the three-divisible recrystallisation in heavy water had been used. The samples were irradiated with ^{60}Co at room temperature and in liquid nitrogen. The irradiation doses were 10 kGy and dose rate was 8,3 Gy/s. To increase the resolution of the ESR spectra hyperfine structure the second derivative for the absorption curve was registered. The measurements were conducted in X-range at temperatures 77-430 K.

RESULTS AND DISCUSSION

The primary anion-radical is formed when an undamaged molecule carboxylic group trap an electron that expulsion by radiation from other molecule of initial substance

$$CH_3CH(NH_3)^+COO^- + e^- \Rightarrow CH_3CH(NH_3)^+CO^-O^-.$$

At once after it the amino proton of the adjacent molecule passes on the carboxylic group of anion-radical. The free radical $\dot{C}H_3CH(NH_3)^+CO^-OH$ (I) is formed and it's observable experimentally(2).

The charge of damage is localised on adjacent molecule. At T>110K the C-N bond in radical I is broken off, then the intermediate radical CH_3CHCOO^- (II) and molecule $NH_3$are formed(3,4). At last, at T>220 radical II is transformed to stable radical $CH_3CHCOOH$ (III) (5,6). Its ESR signal is used to determine an absorbed dose.

The structures of radicals I-III are good investigated and known. The radical III is different from the radical II by space structure, orientation in crystal lattice and the hindered internal rotation of methyl group. The width of ESR lines, relaxation time and consequently, signal strength, on which the absorbed dose is defined, depend on the internal rotation frequency. The most drastic dependence is observed in area of frequencies $\nu_c \cong \delta\omega$; where $\delta\omega$- is distance between the exchanging lines of spectrum in the frequency units.

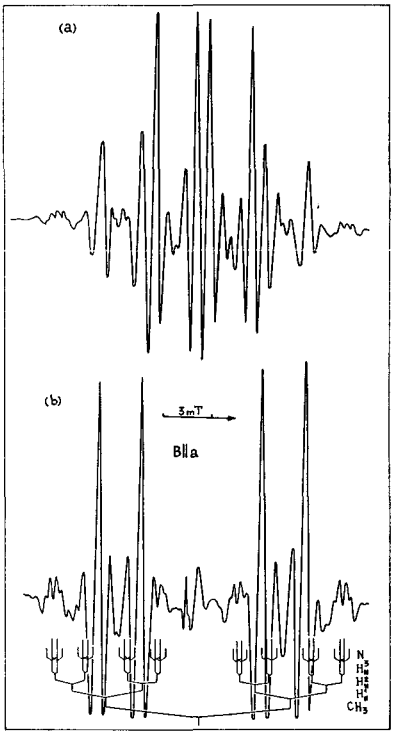

Fig. The spectra ESR γ-irradiated L-alanine single crystals: at T=293K (a); at T=160 K (b)

For internal rotation of methyl group of the radical III the frequency ν_c is reached at T≅160K. At room temperature the rotation frequency $\nu >> \nu_c$, the ESR spectrum this radical is completely averaged and the internal rotation of methyl group not appreciable influence on the ESR signal strength.

However, in 1987 the new type of movements of the radical III was described as conformational transitions[7,8]. Has appeared that the ESR spectrum, which registered in range 250-430K is means between spectra of two conformers.

It is essentially, that the frequency of these movements is reached ν_c at room temperatures. Thus, the measured ESR signals at different temperature in range 18-28 °C will have be the different intensity.

From this it follows that the phenomenon of the conformational transitions of radical III must take into account to increase the accuracy of the ESR alanine dosimetry. It is interesting, that the space structure and orientation in unit cell of the one conformers of the radical III coincides with structure and orientation of the intermediate radical II. Photo stable radical $CH_3CHCOOH$ has the same structure and orientation also. It is formed under action of light with the wave length λ≤365nm on irradiated crystal of L-alanine. The stability of this radical is the same that the radical III. So, the radical CH_3CHCOO^- may have dissimilar stability depending on treatment of the crystal.

To clarify the cause of this fact the simulation of transform the anion radiation damage from primary to stable state was performed. The calculations were carried out by AM1 and INDO. As the model of the radiation damage was used cluster $[CH_3CH(NH_3)^+COO^- + 3*(NH_3)^+CH_2COO^-]$. The central molecule that is rearranged to radical, is connected with three adjustment molecules by hydrogen bonds on carboxylic group. For simplification of computations the methyl groups of these molecules are replaced by the hydrogen atoms.

At transition I ⇒ II the C– N distance is increased from 1.47 to 2.85A. The energy E≈110 kJ/mole is thus allocated. As a result the radical $CH_3CHCOOH$ and NH_3 molecule are separated. The carboxylic proton is discharged from chain O–H...N and then is seized in a nearest trap. The oxygen atom of the radical is connected with the nitrogen atom of the molecule and the radiation damage energy is decreased. It explain the parallel displacement of the radical II in comparison with initial molecule as well as the absence of proton in the carboxylic group of the radical and in the amino group of the appropriate molecule[4].

At T≥220K the proton trapped on nearest environment comes back on the carboxylic group of the radical. It results in breakage of connection O...N. The radical II displaces back in situation which characteristic for radical I and the initial molecule. However for maintaining the equilibrium distance between molecule NH_3 and C_α atom the radical is turned in unit cell. As a result the distance between the ammonia and the methyl group of

the radical is decrease. This is tended to occurrence of barrier for internal rotation of methyl group. Then the reduction of energy radiation damage and stabilisation of radicals III is accounted for by this interaction. The appearance of stable radical CH_3CHCOO^- under action UV on irradiated L-alanine may be explain by moving the proton from chain O–H...N to remote traps and the return transition does not happen at room temperature.

The primary radical for cation radiation branch is σ-radical with delocalized unpaired electron density. The most the probable structure is the neutral radical $[CH_3CH(NH_3)COO]$ (V) [10]. Positive charge is transferred ammonia proton to adjacent molecule[11]. At T≈110 K this radical decay and the second type of the stable radical in L-alanine is formed. Its ESR spectra are observed on background spectra of radical III. In order for identify this radical the angular and temperature dependencies of ESR spectra of irradiated L-alanine and L-alanine-d_3 crystals were investigated. Earlier[12,13], the CH_3CHNH_2 structure had been assumed for it. However an analysis of our experimental data follows that the unpaired p_z electron interacts with three protons of a methyl group and three ammonia protons through superconjugation and with nitrogen through configuration mechanism. Interaction with α-proton was not found. It was concluded[14], that the radical structure is $CH_3C(NH_3)^+COO^-$ (VI). It is essential that characteristic frequency of the methyl group rotation in radical VI is reached at the same temperature that for radical III.

CONCLUSIONS

The $CH_3C(NH_3)^+COO^-$ (VI) radical is stabilised in irradiated L-alanine in addition to $CH_3CHCOOH$ (III) radical. Their concentrations are the same approximately. Both radicals exhibit internal motions and characteristic frequency ν_c of the conformational transition of the radical III as long as the amino proton's exchanges in the radical VI are reached at room temperatures. These phenomena should be considered in order to increase the accuracy of ESR alanine dosimetry.

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Extremity Dosimeter by $Li_2B_4O_7(Cu)$ and its Response to β Rays

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An ideal dosimeter to evaluate skin dose should be a $5mg/cm^2$ detector with a $5mg/cm^2$ window. To realize the TL detector using $Li_2B_4O_7(Cu)$ of density $2.2g/cm^3$, the thickness of the phosphor layer must be as low as 23μm. However, a thin TLD such as this would lack the sensitivity necessary to practical use.

In designing an extremity dosimeter composed of a single layer of phosphor grains, we first investigated β ray response of samples prepared using various combinations of grain size and window thickness. The $Li_2B_4O_7(Cu)$ grains of three grades in size, 38 to 45μm, 45 to 53μm and 53 to 75μm, were fixed on plastic films to form single layers. These samples were exposed to ^{90}Sr-Y, ^{204}Tl and ^{252}Pm β rays through polyethylene windows of thickness 1.2 mg/cm^2, 3.6 mg/cm^2 and 6.0 mg/cm^2. Figure 1 shows the β ray energy response of these samples indicating that the window thickness more affects the energy response than grain size. Even the largest grains (53 to 75μm) showed a flat response when the window thickness is 3.6 mg/cm^2. The lowest detectable dose of the sample using this grain size was 0.2 mSv.

Figure 2 shows the structure of the extremity dosimeter using 53 to 75μm grains. A circular phosphor layer of diameter 2.5mm was formed on a black polyimide film of 30mm x 5.5mm x 0.07mm, with the phosphor guarded by a hard frame of 9mm x 5.5mm x 0.6mm. A dosimeter number is given by both barcode and numerical characters. The dosimeter is worn on the fingers with a thin plastic cover with thickness approx. 3.6 mg/cm^2.

Figure 3 shows the conceptual view of the main part of the extremity dosimeter reader. A magazine guides the dosimeters first to barcode scanner then to the dosimeter reading position. The infrared ray beam from the lamp heats the dosimeter, and the photomultiplier detects the TL by a photon counting method.

The requirements to extremity dosimeter is somewhat contradictory. For example, a dosimeter must be small to be worn on finger, whereas it must be big enough to be treatable by human hands. The dosimeter above, which is a combination of small detector guarded by hard frame and a larger flexible tail where visible numbers are printed, could be one solution.

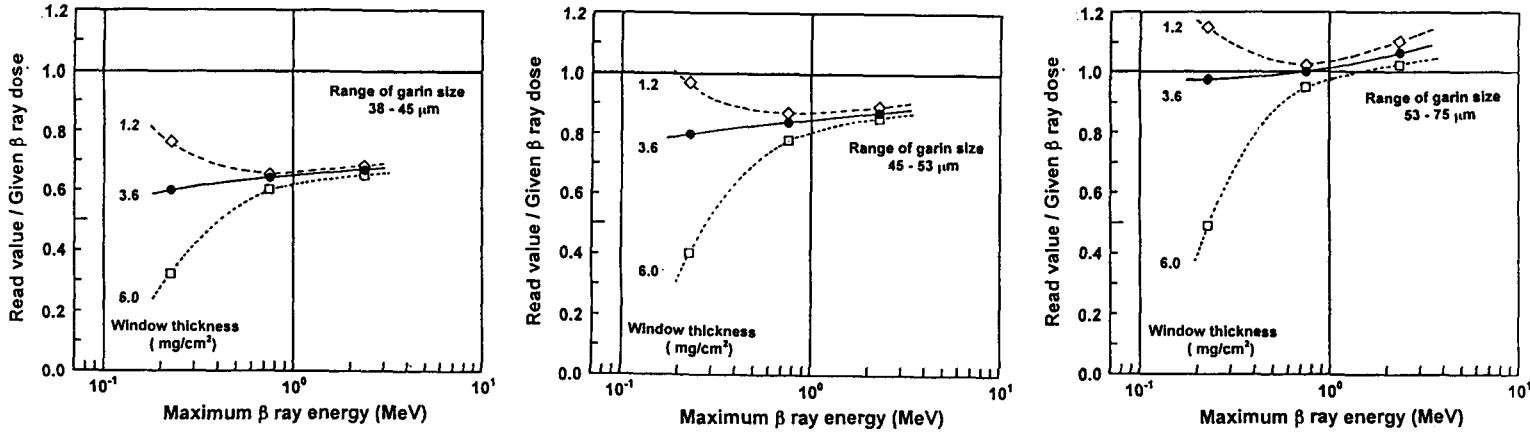

Fig.1 Response of the single layer of the $Li_2B_4O_7(Cu)$ grains covered with polyethylene film of various thickness.
Radiation source: ^{90}Sr-Y, ^{204}Tl, ^{252}Pm

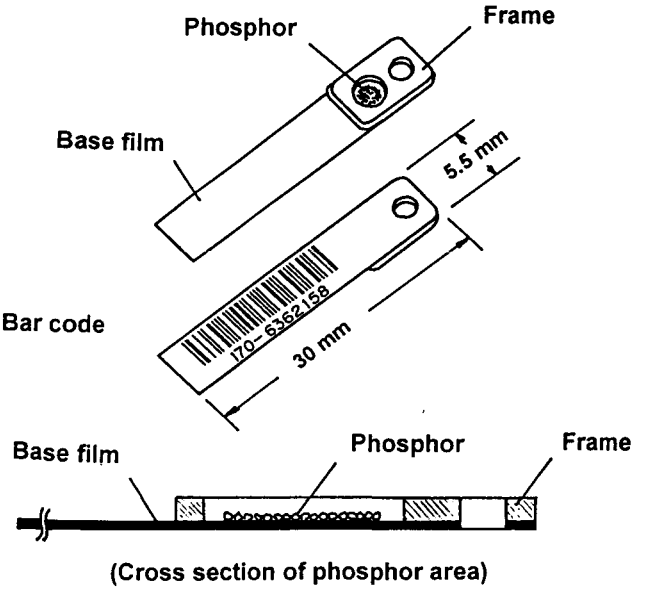

Fig.2 Structure of extremity dosimeter

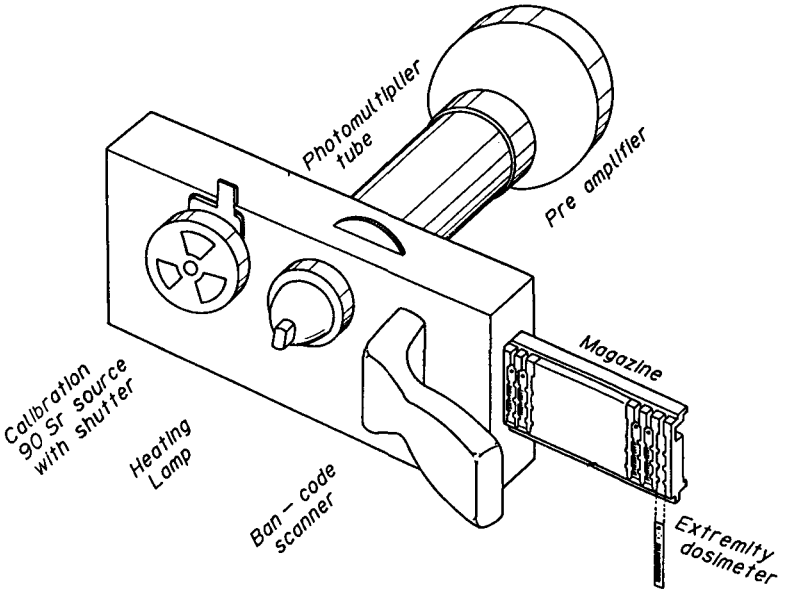

Fig.3 Conceptual view of the dosimeter reading portion of the automatic reader

DOSIMETRY OF ELECTRON AND GAMMA RADIATION WITH DL-ALANINE

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INTRODUCTION

A dosimetric method based on the quantitative determination of stabilised free radicals in irradiated crystalline DL-alanine by electron spin resonance (ESR) spectroscopy was proposed as early in 1962[1]. Since then, alanine dosimeters owing to their unique properties have been investigated by many authors and used in dosimetry of various types of radiation, namely gamma rays, electron and neutrons.

Alanine is a simple aminoacid, on irradiation at room temperature predominantly free paramagnetic radicals of the type CH_3 - CH - COOH are produced.

This paper reports the application of powder DL-alanine/ESR dosimeter for measurement of absorbed dose of gamma radiation from ^{60}Co sources and reactor nucleus and electron beams from accelerator. The obtained results give useful information about the instrumental care necessary to obtain the needed overall accuracy in determination of absorbed dose.

MATERIALS AND METHODS

Commercially available purified crystalline amino acid DL-alanine (Merck) has been investigated for use in radiation detection. The chemical composition of alanine (CH_3 - $CHNH_2$ - COOH) is close to that of tissue, thus providing good similarity to biological systems in absorption of radiation energy[2].

The microcrystalline powder is transferred to suitable sized quartz tubes for X band ESR spectrometry.

The irradiations were performed using a panoramic ^{60}Co gamma ray source, electrons emitted from an accelerator (Dynamitron II) for industrial purposes with samples sealed in glassy ampoules. The samples irradiated at the IPEN Reactor - IEA-R1 were sealed in poliethylene film and covered with Cd foil. All doses given further are expressed as dose to water, as determined by of Fricke dosimetry for gamma ray irradiations.

The ESR spectra were all recorded at room temperature using a JEOL spectrometer (model: JES-ME 3X) with a microwave cavity (TE_{011} mode) operating a frequency of 9400 MHz (X band) and an constant microwave power of 0.1 mW. The magnetic field setting was 334 mT, the field scan range was 25 mT, the scan time was 0.3 s, and the magnetic field modulation amplitude and frequency were 0.5 mT and 100 kHz, respectively, for all dose - response measurements. The gain was always adjusted so that an optimum ESR signal was obtained. The ESR spectrometer sensitivity was checked by introducing into the cavity a DPPH sample (2,2- diphenyl - 1- picrylhydrazyl) contained in a sealed quartz tube.

The ESR spectra was recorded as the first derivative of the paramagnetic absorption spectra where the response of the alanine was expressed as the maximum peak to peak amplitude of the ESR spectra and measurement taken at different days were normalised using a Mn^{2+} reference sample.

RESULTS AND DISCUSSION

The Figure 1 shows the first derivative (ESR spectrum) of the microwave absorption signal of an alanine dosimeter irradiated with ^{60}Co gamma rays at 1 kGy.

The reproducibility of ESR signal height depends on the short and long term stability of the instrument parameters, as well as on the reproducibility of the position of sample inside the microwave cavity. Short-term stability was investigated by carrying out repeated recordings of an irradiated sample. Each parameter was changed after a recording had been completed and then adjusted again, keeping the sample inside the cavity all time. Variations in the shape of ESR spectra of DL -Alanine irradiated with gamma ray and electrons measured at different microwave power were studied.

Post irradiation stability of the DL-alanine response as a function of absorbed dose was investigated for absorbed doses from 1 to 60 kGy over a storage period of 180 days at room temperature (about 15-28 ^{0}C) and indicates sufficient longevity.

Figure 1. ESR spectrum of alanine irradiated with ^{60}Co gamma rays at 1 kGy[3].

The same signal-dose relationship was obtained for both gamma rays (1.25 MeV) and electron beams (1.14 MeV). No dependence of the response on dose rate has been found in the range from 0.21 to 1.16 kGy/h.

In the dose range of 10^2 to 10^5 Gy, the dose-response signal showed a linear relationship and the precision was less than 2%. At dose levels greater than 10^5 Gy the dose response curve begins to saturate, it is shown in Figure 2.

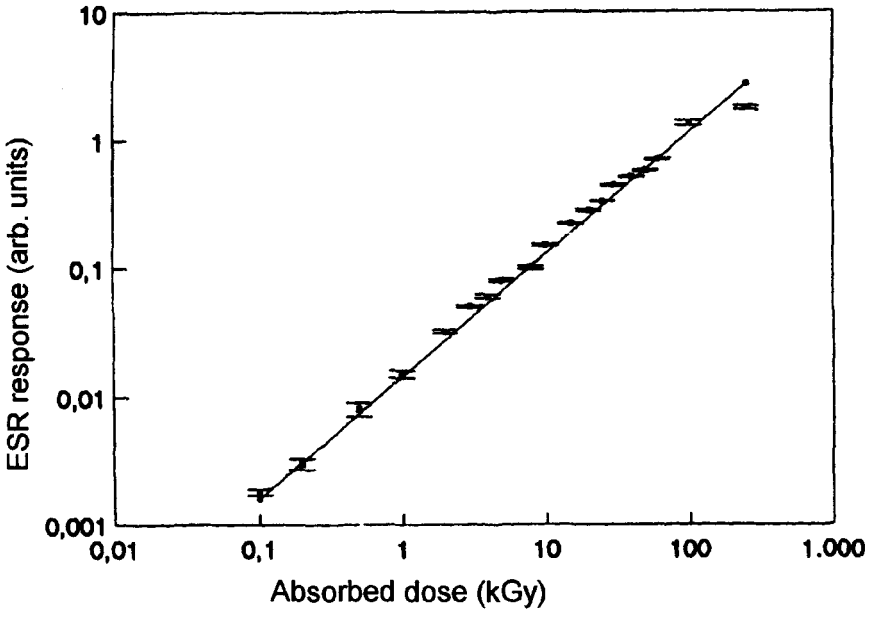

Figure 2 .Dose-response curve of DL-alanine irradiated by ^{60}Co gamma rays[3].

CONCLUSION

The obtained results in this work using a non-destructive readout technique demonstrate that the alanine dosimeter has the long-term stability of radiation induced response, accuracy and reproducibility to be suitable for application in high-level dosimetry. The dosimetric information obtained from the measurement of the gamma absorbed dose in reactor nucleus, are in general accordance with the results determined by thermoluminescence.

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NEUTRON DOSIMETRY IN FRENCH NUCLEAR POWER PLANTS. PROBLEMS AND THEIR SOLUTIONS IN 1995

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Background and development of the problem.

Exposure to neutron radiation in the nuclear industry is normally limited to a small number of workers essentially EDF employees operating in specific areas. Operational collective dose due to neutron exposure is almost negligible compared to the rest of the external doses (less than 2 % in the collective dose equivalent). But this risk represents a significant fraction of the annual dose equivalent of those exposed.

Suggest specifications for individual dosemeters which would idealy meet both technical and practical requirements.

Present situation of Neutron Monitoring in the French Nuclear Power Plants [2]

Exposure of the workers to neutron is mainly related to interventions in the containment building. Fields of radiation in the reactor environment are well known to present energy spectra which can be divided in 2 components :

- The high energy neutrons in the range 50 keV - 2 McV with a maximum of intensity between 200 keV et 600 keV. The dose equivalent rates due to this component lie typically from 0.05 to 500 $mSvh^{-1}$. High energy neutron beams mainly occur locally due to less efficacy of the shieldings and thus are subject to strong variations in space.

- The neutrons from thermal to 10 eV with a maximum of intensity between 0.05 and 0.3 eV. The dose equivalent rates slightly vary around 0.05 $mSv.h^{-1}$. Due to multiple scattering thermal neutrons constitute almost isotropic fields.

Interventions inside the containment building are limited both in frequency (less than one per month in each plant) and duration (about 1 hour). Gamma dose rates in the containment building are in the same order of magnitude than neutron dose rates mentioned above. Collective dose received per intervention is about 0.6 mSv (neutron : 0.26 mSv and gamma 0.34 mSv) distributed between 3 persons on average.

Discharge of irradiated fuel elements from the reactor and receipt of MOX also contribute to exposure of the employees. Neutron energy spectra present a maximum in the region 200 keV - 2 MeV. Dose rates typically measured are in the order of $0.1 mSv.h^{-1}$. Individual doses received during each operation vary between 0.1 and 1 mSv. On average 5 interventions are made each year on a reactor.

Exposure to neutrons at EDF is characterised by doses received during operations of short duration. Workers generally move from place to place where radiation fields can change rapidly in intensity as well as in energy spectrum and angular distribution. This explains why individual dosimetry is particularly critical for this category of workers.

None of the passive techniques existing for individual monitoring was considered to be convenient owing to their limitations, i. e. high energy response dependence, poor lower limit of detection for neutrons or high sensitivity to photons. Thus, till now individual dose assessments are extrapolated from ambiant monitoring. While the reactor is functioning at reduced power, ambient measurements with rem-counters are carried out in the different locations where workers will then operate. In some cases spectrometry systems are used to measure fluence spectra in working areas. Neutron dose rates are extrapolated at the nominal power of the reactor.

At last, individual doses are calculated taking into account main residence duration for each worker in the different locations. Further measurements with rem-counters and electronic gamma devices are made by the personnel during their interventions. This allows to consolidate individual dose assessments based on the ambient monitoring and also to prevent any risk of over exposure.

Results and conclusion

At present, the problem of personal dosimetry for people working in neutron radiation fields is still far from solved to a level comparable with gamma radiation. None of the passive personal detectors developed prior to 1985 are suited to neutron dosimetry in nuclear power plants, either because of poor spectral response or because of inadequate sensitivity.

Before the advent of bubble detectors, the only form of personal dosimetry was based on ambient dosimetry, in accordance with approved procedures applied uniformly in all power plants. The dose equivalent is therefore obtained by multiplying the dose equivalent rate by the time spent in each work area. The dose equivalent rate is monitored using the rem meters carried by the personnel and is checked against the mapping values recorded during the startup period.

This form of dosimetry, the accuracy of which is relatively rudimentary, still compares well with the forecasts based on neutron and gamma dose mapping. It has the advantage of being reliable albeit difficult to implement.

In 1992, this situation led EDF to set up a task group made up of representatives of the various French operators concerned with neutron dosimetry (CEA*, COGEMA*, EDF* and DGA*) and the CEA IPSN, with a dual objective :

- in the short term: to establish procedures for the use of Canadian BTI bubble detectors, which have just undergone successful certification testing [3].

- in the medium term (1996 - ...) : to adopt an advanced, direct reading dosimeter incorporating dose equivalent and equivalent dose rate alarms, in accordance with joint specifications drawn up by the operators.

The main difficulty in developing an electronic neutron dosimeter lies in achieving correct discrimination of the spurious signals generated by gamma photons. This will require a great deal of on-going effort and scientific discipline.

At present, the direct reading electronic dosimeter is still at the basic research stage and is unlikely to come on to the market in the next 5 years. This delay is due to two main reasons: the limited size of the market, some 3000 units in France for the nuclear industry as a whole; and the lack of knowledge of the problems specific to this industry. The existence of joint specifications should enable research organisations to better manage their research efforts. Furthermore, there appears to be renewed interest in personal neutron dosimeters and several laboratories throughout the world are developing potentially successful techniques.

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MEASUREMENT OF REACTION RATE DISTRIBUTIONS IN PHANTOM IRRADIATED BY INTERMEDIATE ENERGY NEUTRONS FOR EFFECTIVE DOSE EVALUATION

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ABSTRACT

Reaction rate distributions in a plastic phantom irradiated by 41- and 64-MeV quasi-monoenergetic neutrons were measured with a ^{238}U fission counter and solid state nuclear track detectors. Measured distributions were compared with those calculated by using the HERMES code system.

INTRODUCTION

From the view point of radiation protection in the radiation shielding design of high energy accelerators, it is important to establish an evaluation method of effective dose due to high and intermediate energy radiations. The HERMES code system (1) has been extended to calculate the effective dose (2). The accuracy of the code system, however, has not been evaluated for intermediate energy neutrons by experiments. Therefore, experiments were carried out for 41- and 64-MeV quasi-monoenergetic neutrons generated by the $^{7}Li(p,n)$ reactions with 43- and 68-MeV protons at an AVF cyclotron facility, TIARA (Takasaki Ion Accelerators for Advanced Radiation Application) facility, of Japan Atomic Energy Research Institute.

EXPERIMENTS

Figure 1 shows a cross sectional view of the experimental arrangement. Quasi-monoenergetic source neutrons ("41 MeV" and "64 MeV") of about 41- and 64-MeV were produced in 3.6 mm and 5.2 mm thick ^{7}Li-targets by 43- and 68-MeV protons, respectively. Source neutron spectra above 6- and 7-MeV were measured with a 5"ϕ x 5" BC501A liquid scintillation detector for the "41 MeV" and the "64 MeV". Absolute fluences of source neutrons in the monoenergetic peak were also measured with the recoil-proton counter telescope. Source neutrons emitted in the forward direction were introduced to an iron collimator of 11 cm diameter, while proton beam penetrating through the ^{7}Li-target was bent down by a clearing magnet to a Faraday cup. The phantom of 30 cm cube made of an acrylic resin was placed at the exit of the collimator. The intensity of source neutrons was monitored with a proton beam Faraday cup and two fission counters placed near the ^{7}Li-target, of which the efficiencies were calibrated by the recoil-proton counter telescope. Reaction rate distributions along the center line in the plastic phantom were measured with a ^{238}U fission counter (Centronic FC4A) and SSNTDs (Nagase Landauer Ltd. MR-3) which had been calibrated with a ^{252}Cf source.

EXPERIMENTAL RESULTS, ANALYSIS AND DISCUSSIONS

Energy spectra of transported neutrons were calculated with the HERMES code system. In the code system, neutron energy spectra above 19.6 MeV were calculated by the HETC-KFA2 code and those below 19.6 MeV

were calculated by the MORSE-CG code with the DLC-119/HILO86 multi-group cross section library (3). Figure 2 shows the calculated energy spectra at the depths of 3 cm and 25 cm in the phantom for the "41 MeV". Error bars of the calculations in Fig. 2 show the fractional standard deviations. The relative energy spectra of source neutrons below 6- and 7-MeV were assumed to be the values at 6- and 7-MeV for the "41 MeV" and the "64 MeV", respectively. Reaction rate distribution of the ^{238}U fission counter was obtained from the neutron spectra and fission cross sections (4, 5). Calculated and measured distributions of the reaction rate of the fission counter are compared in Fig. 3. The calculated distributions are in good agreement with the experimental ones at the depth up to 15 cm in the phantom, while overestimates about 25% at the depth of 25 cm. The neutron energy spectra for whole regions were also calculated by the MORSE-CG code with the HILO86 library. The calculated distributions with these energy spectra are also in good agreement with the experimental ones even at the depth of 25 cm.

Reaction rate distributions of SSNTDs were also obtained from the neutron spectra and an energy response calculated by a newly-developed Monte Carlo code (Y. Nakane et al.) in which the energy range of incident neutrons had been expanded to intermediate energy. Calculated and measured distributions of reaction rates of SSNTDs are shown in Fig. 4. The calculated distributions for the "41 MeV" are in good agreement with the measured ones within 15% at the depth between 10 and 25 cm in the phantom, while those for the "64 MeV" are about 50% higher than the measured ones at the depth after 20 cm. These disagreements probably come from the energy response of SSNTDs for the incident energy of around 64 MeV. At the depth up to 5 cm, the calculated distributions for both the "41 MeV" and the "64 MeV" are lower than the measured ones. The contribution of the incident neutron spectra below 6- and 7- MeV to reaction rate distributions is estimated about 30% for the SSNTDs at the depth up to 5 cm. This discrepancy can be attributed to the assumption of source neutron spectra below 6- and 7-MeV. For the fission counter, that contribution is estimated to be in less than 4%.

SUMMARY

Reaction rate distributions of the fission counter and the SSNTDs in the plastic phantom using 41- and 64-MeV quasi-monoenergetic neutrons were measured and compared with those obtained from fission cross sections, an energy response and energy spectra calculated with the HERMES code system. Calculated distributions of the reaction rate of the fission counter obtained with the energy spectra by the MORSE-CG code and the HILO86 library are in good agreement with the experimental ones for the "41 MeV" and the "64 MeV". Calculated distributions of the reaction rate of the SSNTDs for the "41 MeV" are in good agreement with the measured ones within 15% at the depth between 10 and 25 cm in the phantom, while those for the "64 MeV" are about 50% higher than the measured ones at the depth after 20 cm.

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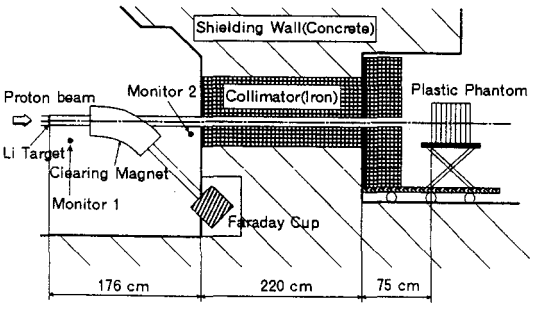

Figure 1. Cross sectional view of the quasi-monoenergetic neutron source facility.

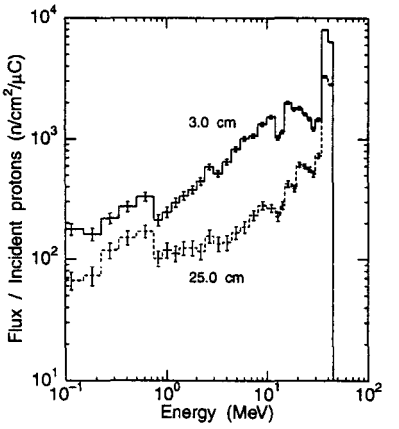

Figure 2. Calculated energy spectra of transported neutrons at the depths of 3 cm and 25 cm in the phantom for the 41 MeV quasi-monoenergetic neutrons.

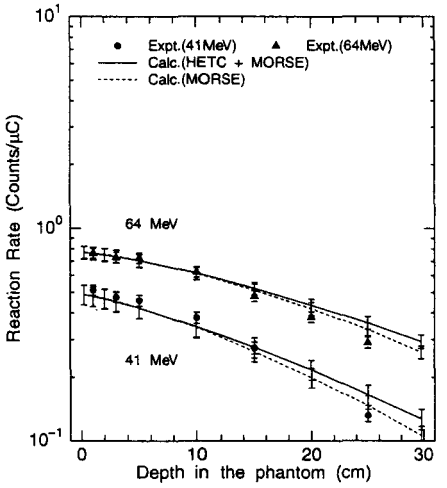

Figure 3. Calculated and measured distributions of the reaction rate of the ^{238}U fission counter.

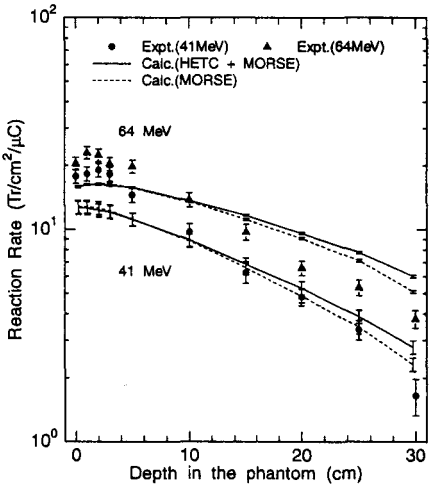

Figure 4. Calculated and measured distributions of the reaction rate of the SSNTDs.

Real Time Neutron Dosimeter Response Calculations

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Abstract

The response of a real time neutron dosimeter using a thin LiF target sandwiched between two parallel surface barrier semiconductor detectors is studied for different neutron distributions and different angles of incidence. Calculations of the response function defined for a simultaneous detection by the two detectors of the particles emitted when the reaction $^6Li(n,t)\alpha$ occurs in the target are fulfiled by geometrical considerations of the reaction kinematics and the differential cross section variations. Finally, the efficiency of the studied detection system is analysed for dosimetric uses.

Introduction

A good neutron dosimeter will present good sensitivity and a known neutron energy response. These conditions are generaly fulfiled by neutron spectrometers, reason for which many neutron dosimeters derive directly from neutron spectrometer designs. Recently [1], a neutron spectrometer using a thin LiF target and two surface barrier detectors has been studied, designed, and applied to determine neutron energy distributions at different irradiation sites in a nuclear reactor. This design is easily modified to serve as a neutron dosimeter.

Dosimeter description

Two surface barrier detectors which active surfaces face each other are used to detect charged particles from ^{6}Li(n, t)α reactions occuring in a thin LiF layer sandwiched between the two diods. The response of a neutron dosimeter based on the use of such a detection assembly is strongly dependent on the kinematics of the ^{6}Li(n, t)α reaction and the geometrical considerations of the emission and the detection of the charged particles. In this work, the attention is focused on the detection assembly in order to calculate the response function of the neutron dosimeter.

Response function calculations

Neutron fluxes are generally contaminated by γ. In order to realize a neutron dosimeter only suitable for neutrons in a mixed $n - \gamma$ field, a coincidence detection mode is used. In the first approach, it seems that such a detection assembly will have a poor detection efficency

for intermediate and fast neutrons. Nevertheless, it has been shown in another work [1], that the charged particles issued from the reaction $^6Li(n,t)\alpha$ are emitted in an angle greater than 140^0 for neutron energies as important as the fission neutron energies ($17 \sim 20\,\mathrm{MeV}$).

The total number of the particles detected with the coincidence condition is :

$$N(E_n) = N_0\ \Phi(E_n)\ G(E_n) \tag{1}$$

where the geomerical function $G(E_n)$ is defined as :

$$G(E_n) = \int_0^e dx \int_0^R r dr \int_0^{2\pi} d\phi \int_0^{\theta_{max}} \int_{\beta_{min}(\theta,\phi)}^{\beta_{max}(\theta,\phi)} \frac{d\sigma(\cos\theta)}{d\cos(\theta)}\ P\ R(\beta)\, d\beta\, d\cos(\theta) \tag{2}$$

whith :

N_0 : target nuclei number, $\Phi(E_n)$: neutron flux of an energy E_n, e : ^6LiF target thickness, R : detector active surface radius, P : coincidence detection condition : P = 1 if the detection coincidence is fulfiled, P = 0 if the detection coincidence is not fulfiled, and $R(\beta)$: the ratio of the detected particles emitted within a solid angle $d\Omega = d\beta\, d\cos(\theta)$ to the total emitted particles.

Results of calculations of 78 neutron energy groups are reported in Fig.1.a. were the response functions for normal and lateral incidences, i.e. 0^0 and 90^0. It can be seen that geometrical function follows 1/E form for neutrons below 1 keV and the dosimeter angular dependence is only sensitive to fast neutrons.

The neutron energy response of the dosimeter is calculated using several well known neutron distributions. Calculations are fulfiled taking into account the energy detection resolution of diods (fwhm $\simeq$ 15 – 20keV). This conduct to express the total number of the detected particles of a neutron incident energy E_n as :

$$N(E_n) = N_0 \int_{E_n-\Delta E_n}^{E_n-\Delta E_n} \Phi(E)\ R(E_n - E)\ dE \tag{3}$$

with:

$$R(E_n - E) = \frac{1}{\sqrt{2\pi}\,\sigma} G(E)\ \exp(E_n - E)^2/2\sigma^2 \tag{4}$$

Results of the neutron dosimeter responses for : 1/E distribution, fission distribution and reactor distribution (moderated 1/E plus fission) are reported on figures 1.b.c. and d.

Conclusion

For neutron energies below 1 keV, the response of the real time neutron dosimeter has a 1/E form independently of the angle of incidence and the neutron distribution. Whereas, for fast neutrons it is found that the dosimeter is sensitive to angle and energy variations.

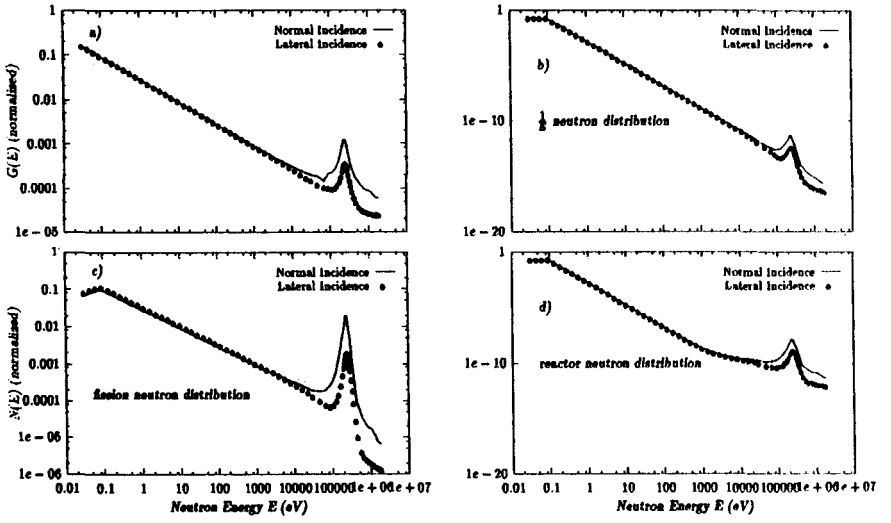

Fig1. a) The geometrical function. Neutron response functions for : b) 1/E distribution. c) fission distribution. d) reactor distribution.

Acknowlegements

The authors are grateful to Mr. D. E. Cherouati for his helpful remarks.

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IRPA9
1996 International Congress on
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PAPER TITLE Progress in the characteristics of realistic neutron spectra facility (RNSF)

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MAJOR SCIENTIFIC TOPIC NUMBER .5... (see page 7)

ABSTRACT (See instructions overleaf)

For the 5 last years, the concept of realistic neutron spectra simulating practical conditions encountered at workplaces has been investigated by several laboratories involved in radiation protection dosimetry and collaborating in the frame of a Joint CEC Contract and Eurados Working Committees. At a further step, the ISO/WG2 is also involved in preparing standards which define the characteristics of reference radiations and calibration procedures. As a consequence, the radiation field of a RNSF has to be accurately specified in terms of both angle and energy distributions. This is particularly important for the calibration of individual dosemeters which the angular response has to be taken into account for. Calibration facilities mimic as closely as possible the features of an aligned and expanded field in order to comply with fluence-to-dose equivalent conversion coefficients of ICRU operational quantities. This paper intends to report on techniques which have been developed at the SDOS/Cadarache RNSF to satisfy with those requirements and to implement the appropriate calibration procedures assuring traceability to primary references. This research work relies on a computational part to optimise the modifications of the facility and is complemented by an experimental validation involving irradiations of several types of dosimeters distributed by routine dosimetry services.

NEUTRON SPECTROMETRY FOR RADIATION PROTECTION AT THE JINR

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The major basic facilities of JINR are the Nuclotron - a superconducting accelerator of relativistic nuclei, the U-400, U-400M and U-200 heavy ion cyclotrons, the Synchrophasotron - a 10 GeV proton and light nuclei accelerator, the Phasotron - a 680 MeV proton accelerator, the IBR-2 fast neutron pulsed reactor and the IBR-30 neutron pulsed booster multiplier. These nuclear facilities generate the different types of radiation fields with wide energy range and complex composition. The main radiation component of the fields behind thick biological shielding is neutron with energy from thermal up to several hundreds of MeV. The energy spectra of these neutrons vary widely, depending on the type and design of the facility and its shielding. Development of broad energy range neutron spectrometry techniques for measurements in mixed scattered radiation fields is the one of the prior topics in radiation protection. There are no experimental techniques to cover an entire neutron energy range, but the problem can be solved by the combining the measurements with a set of different spectrometers. The key information for efficient accelerator shielding design is true determination of the source term. Double differential (on angle and energy) distributions of neutrons produced by primary particles in thick targets which imitate the construction parts of accelerators are the crucial data to define the source term. There is a lack of such experimental information available especially for heavy ions. This presentation is intended to give the flavor of our work on the problems, its objectives, range of interests and methods.

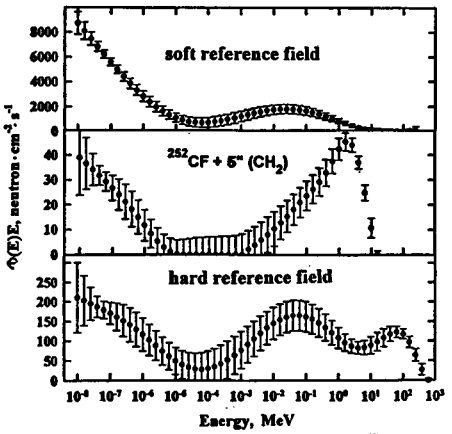
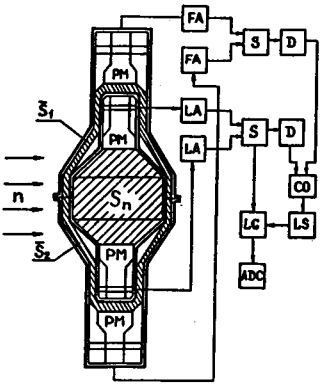


Fig.1. The neutron reference fields of JINR.

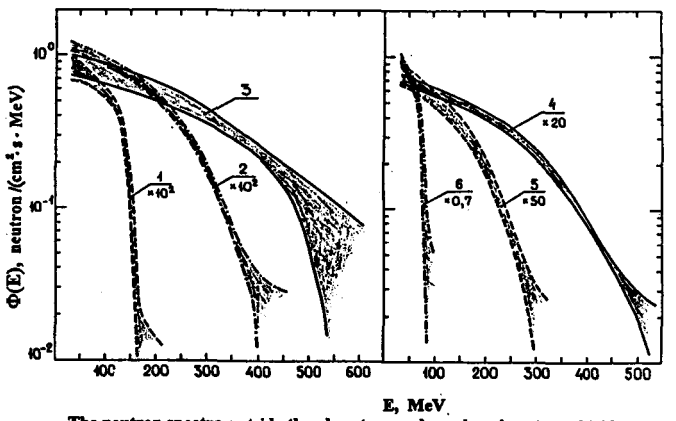
The JINR has significant experience in the implementation of neutron detectors having near 1/v response with different moderators to detect wide-range energy neutrons. The spectra of the JINR neutron reference fields, design for neutron metrology purposes, are shown in Fig.1 for example /1/. The soft reference field is formed by scattered neutrons into the 660 MeV phasotron labyrinth.The next neutron reference fields are produced using a ^{252}Cf source within polyethylene moderators. The hard neutron reference field is formed behind thick concrete shield of the 660 MeV phasotron irradiated by the secondary radiation from the accelerator chamber and the target station. The spectra have been measured by multisphere neutron spectrometer with LiJ(Eu) detector (4 mm diameter, 4 mm height) and unfolded by the statistical regularization method. The 2,3,5,10,12 inches polyethylene moderators, "bare" and "bare" + cadmium data has been included in the system of equations. The response functions were taken from the /2/.

Widely used multisphere spectrometer has a wide energy range, low sensitivity for photons and charged particles and isotropic response, but it yet does not give enough information content at high energies.The estimations of the hadron fluences with energies above 20 MeV by using ^{12}C(n,2n)^{11}C activation method and other suitable a priori information about neutron spectra are often applied to improve the spectra shapes



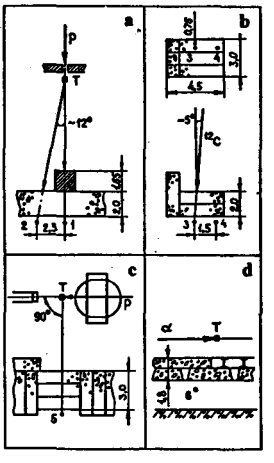
The construction of the high - energy neutron spectrometer and the block diagram of the electronic system.

S_n - neutron detector; $\bar{S}_1$, $\bar{S}_2$ - veto counters detectors; PM - photomultipliers; FA - fast amplifiers; LA - linear amplifiers, S - linear summators; D - discriminators; CO - coincidence unit; LS - logic shape unit; LG - linear gate; ADC - analog - digital converter unit.



The neutron spectra outside the phasotron and synchrophasotron shields. The figures 1 to 6 correspond to the numbers of the measurement points.

at high energy region. Nevertheless, there seemed to be a need of a new type of dosimeter and spectrometer for high energy neutrons. To meet this need in JINR was developed a new high-sensitive technique /3-5/ for measuring the energy spectra, fluence and dose equivalent The construction of the spectrometer is shown schematicaly in Fig.2. The device consists of the neutron counter with near spherical plastic scintillator as a neutron detector (volume is about 1.6 l) coupled with twq PMT. This neutron counter is surrounded by a composite veto scintillator for charged particles rejection equipped with two PMT. The principle of operation of the device is based on the dependence of the readings of the neutron counter on the differential energy neutron flux at various thresholds of signal registration. The calculation of the plastic scintillator neutron detection efficiency vs. threshold was performed using the Monte Carlo code /6/. The influence of the veto counter on the neutron efficiency was taken into account and led to a considerable decrease in the efficiency at neutron energies above approximately 100 Mev. The problem of energy spectrum unfolding from apparatus spectra in this case is similar to the unfolding of multisphere spectrometer spectrum, but instead of counting efficiencies for moderators, one uses scintillator efficiencies at different thresholds. Neutron spectra behind various shields of phasotron and synchrophasotron have been measured by the prototype of the spectrometer. Fig.3 illustrates some high energy neutron spectra in different geometries. The location of the measurement points and shielding configurations are given in Fig.4 (a,b,c,d). The high sensitivity of the spectrometer permits to measure the cosmic neutron spectrum near the Earth's surface as well.



The location of measurement points (solid circles) and shielding configurations. All sizes are given in meters. In (b) the front surface of the shielding is also shown.

The next problem of neutron spectrometry dealt with measurement of the angular-energy distribution of neutron yields from targets irradiated by the particles. The energy spectra of neutrons with energy more than 10 MeV produced in thick copper target by 3.65 Gev/nucleon carbon ions were obtained by TOF method /6/. For this aim the TOF neutron spectrometer with triggering by ions passing through the counter's telescope before the target was designed. The Fig.5 shows the setting and block diagram of the spectrometer. The signal "Start" was formed by the second telescope counter with selection of ^{12}C ions energy losses in the detector and rejection two or more events (ions) in each bunch. The signal "Stop" was formed by the neutron counter on base of plastic scintillator with known detection efficiency in case of the absence of veto counters signals rejecting the charged particles. The difference method by the shadow bar for subtracting the background was used at each measurement. The spectra of neutrons with energy over than 10 MeV from the target (diameter 10 cm, length 13 cm) under two angles are shown in Fig.6. The calculation method for description of hadron double differential yields in thick targets, bombarded by relativistic ions have also been developed on base a thermodynamic "firestreak" model and the solution of the transport equation. The verification of this program have been done by the experiments and gave a satisfactory result.

The experimental check of the calculated response function for the multisphere neutron spectrometer at energies over than tens MeV and for the high-energy spectrometer (especially at high thresholds) is a very crucial problem of these methods. We are planing now to carry out the experiments on verification of the response function on base of the medical facility of the JINR phasotron. This facility permits to have a wide and narrow neutron beams with variable energies from 100 to 500 MeV produced by protons irradiating the light targets (Li, Be) under 0 degree. The other task is the combination of multisphere and high-energy spectrometers data to unfolding of

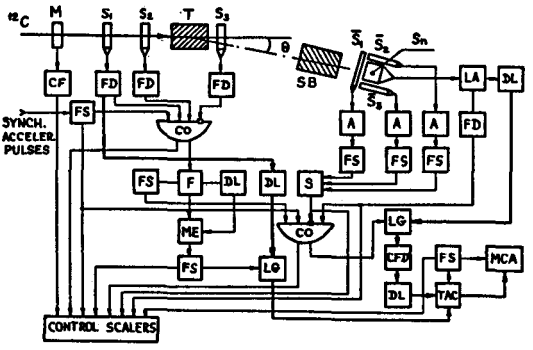


Fig.5 The experimental arrangement and block diagram of the TOF spectrometer. LC - linear gate, CFD - constant fraction discriminator, TAC - time-to-amplitude convertor, FS - fast logic shaper, S - logic summator, ME - multiple events unit, F - fanout unit, M - beam monitor chamber, T - target, CF - current-to-frequency convertor, SB - shadow bar, A - fast amplifier, LA - linear amplifier, DL - delay unit, FD - fast discriminator, CO -coincidence unit, MSA -multichannel analyzer.

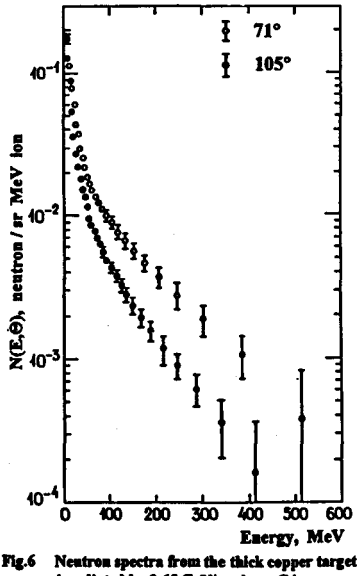


Fig.6 Neutron spectra from the thick copper target irradiated by 3.65 GeV/nucleon C ions.

neutron spectra in entire energy range from thermal up to several hundreds of MeV. In the field of determination of the source term we are planing to study a neutron double differential distributions from thick targets irradiated by heavy ions with energies of tens MeV/nucleon on base of the U-400 JINR facility.

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Dose Characteristics of the IHEP Neutron Reference Fields

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INTRODUCTION

Interpretation of neutron detector readings in terms of dose equivalent is an important problem. Since response of a neutron detector has usually an energy dependence which is not adequately simulating the energy dependence of dose equivalent, its using as a dosimeter in real neutron fields of nuclear-physics device after a simple calibration in a neutron field of radionuclide source (such as ^{252}Cf) leads to considerable errors. There are some different approaches to solve this problem. One of them [1], [2] proposing the dosimeter's calibration in workplace fields is available for personnel and area dosimeters. Another method [3] is the calibration at neutron field having the spectrum simulating the real neutron field one. The method of superposition of calibration fields [4] to obtain weighted neutron response is a following step in this direction. All of these approaches one way or another are concerned with application of neutron reference fields for these goals.

THE IHEP NEUTRON REFERENCE FIELDS

Requirements to neutron field to be "reference" are the following. First of all it would have a neutron spectrum which is simulating typical workplace neutron spectra. Its main characteristics such as neutron spectrum, dose equivalent rate, angular distribution, etc. must be well established. It would be reproducible and enough intensive to provide possibility of quick calibration.

At the Institute for High Energy Physics (IHEP), the workplaces concerned with personnel irradiation are concentrated mostly at the experimental hall of IHEP 70-GeV proton synchrotron. The radiation fields behind the biological shielding of the accelerator have rather component composition where neutrons in a wide energy range from thermal energies up to primary proton beam's energy of 70 GeV are dominated. Numerous investigations of real radiation fields behind the accelerator's shielding [2] ,[5] ,[6] ,[7] , show that they can be separated into three main groups: 1) high energy fields behind the upper shielding (320--350 cm of concrete); 2) low energy neutron fields behind the side shielding (600 cm of concrete); 3) radiation fields with large contribution of charged particle nearby the charged particle transport beams. The IHEP neutron reference fields include the high energy reference field and four radionuclide reference neutron fields [10] formed by filtered and direct radiation of ^{252}Cf and ^{239}Pu-Be radionuclide sources: ^{239}Pu-Be in free air (PU), ^{252}Cf in free air (CF), ^{252}Cf at the centre of polyethylene sphere $\otimes$ 30 cm (CH) and ^{252}Cf at the centre of iron sphere $\otimes$30 cm (FE). The radionuclide reference neutron fields are reproduced at the middle of a special calibration room having dimensions of 5.4 m $\times$ 13 m and 4 m height. The distance between a radionuclide source creating the field and a reference point is the same for all the four fields (0.75 m) as well as the distance from a floor (1.65 m). The high energy reference field (HEF) is reproduced at 1 m height outside the upper shielding in the experimental hall of the IHEP 70-GeV proton accelerator. Inside a volume of 0.5 $\times$0.6 $\times$0.3 m^3 with the centre at the reference point the intensity of radiation varies in the range less than 10%.

MEASUREMENT TECHNIQUES

The dosimeters and spectrometers being used in measurements are listed in the Table 1. The detailed description of measurement procedures may be found in the original the main spectral papers. Briefly, the tissue-equivalent proportional counter TEPC allow one to obtain dose equivalent and absorbed dose of radiation of any kind by simultaneous measuring the charge produced in a cavity and the event spectrum. The analog component remmeter ACR includes three high-pressure ionization chambers (IC): argon-filled IC with aluminium wall (γ and charged particles absorbed dose), tissue-equivalent IC (total absorbed dose) and ^{3}He-filled IC in a 10"-diameter polyethylene moderator (neutron dose equivalent). Other instruments are: multysphere spectrometer (MS) with ^{6}LiI(Eu) detector (the method of linear combination of six spheres readings), thermoluminescent proportional counter SLETCP with aluminum wall.

Table 1. The instruments used in dose equivalent measurements in the IHEP reference fields.

Instrument	Refs.	Measuring components.	Scopes, limitations.	Errors	Certification
TEPC	[8]	Photons and charged particles, neutrons.	E_n > 0.3 MeV	9%	VNIIFTRI, State Standard
ACB	[9]	Photons and charged particles, neutrons.	$H_n/H\gamma$ <100, $H_n/H\gamma$ >0.1	18%	VNIIFTRI, State Standard
MS	[10]	neutron	--	15%	VNIIM, State Standard
TLD	[11]	Photons and charged	--	15%	VNIIM, State Standard
SLETCP	---	Photons and charged	--	10 %	----

RESULTS AND DISCUSSION

The results of neutron and photon ambient dose equivalent measurements for the IHEP Radionuclide reference neutron fields are given in Table 2. They are relate to one neutron of a radionuclide source generating a field, for convenience of further applications. The results are the expert estimation of the values usually taken as an average between TEPC, ACR and MS results except the cases when the error obtained after treatment of an instrument readings exceeds the permissible level for the dosimeter here we use the data for IKS and SLETCP.

Table 2. Measured ambient dose equivalents (expert estimation) and some important integral characteristics of the IHEP reference fields.

Reference.	$H^*(10)$, fSv/neutr., neutrons.	$H^*(10)$, fSv/neutr., γ + ch.part.	$h^*(10)$, pSv.cm^2	E_n, MeV.	$N_{10°}/N_{5°}$ rel.units.
PU	4.88±9%	0.183±20%	322.	3.32	1.32
CF	4.47±12%	0.220±15%	284.	1.83	1.01
CH	0.631±10%	0.250±10%	193.	1.36	0.73
FE	3.38±9%	0.023±15%	216.	0.76	0.61
HEF	4.0±13%, μ Sv/cycle	0.8±15%, μ Sv/cycle	282.	49.	0.86

The neutron spectra of the IHEP reference fields have been measured in[10] by the IHEP multisphere spectrometer [4]. Cadmium-covered polyethylene spheres of 2",3",5",8",10" and 12" diameter with ^{6}LiI(Eu) ⊗ 10× 10mm^3 thermal detector have been used in measurements. For spectrum measurements in high energy neutron fields a 18"-diameter sphere and activation carbon-based scintillation detector $^{12}C(x,xn)^{11}C$ were used. The neutron spectra of radionuclide reference neutron fields are shown at Fig.2 in comparison with accelerator low energy spectra outside the side shield. It could be noticed that the radionuclide fields do not closely simulate the real ones, but nevertheless have more spread neutron spectrum like real ones and rather similar shape of a separate part of a real spectrum which could be interesting from a point of view of a detector response. Then, in Table 3 the components of a neutron spectrum as well as the main spectral characteristics are given for the reference fields in comparison with the same values for the real neutron fields behind the IHEP accelerator shielding, measured in [5] and [7]. The real neutron spectra are divided on several groups according to spectrum hardness. As it is seen from the Table, the range of ambient dose equivalent conversion factors (193--322 pSv. cm^2) and the mean neutron energies in spectrum (0.76--3.32 MeV) as well as the ratio between a separate parts of neutron spectra cover a most part of a real neutron spectra behind the side shielding of the IHEP accelerator . The main advantage of the radionuclide neutron reference fields is a constant spectrum and component composition, good reproducibility and intensity (30--200 nSv/s) which we cannot reach in real fields behind the accelerator shielding.

The neutron spectrum at the high energy reference field is measured by the IHEP multisphere spectrometer with using apriori information. Comparison of neutron spectrum at the IHEP high energy reference field with theoretical spectrum behind shielding calculated by FLUKA [13] and ROZ6H [14] computer codes presence of two peaks - in evaporation region and near 100 MeV is confirmed by all authors. Such behavior of high-energy neutron spectrum appears also in the cosmic neutron spectrum measured in [15].

Table 3. Comparison of the main characteristics of neutron spectra of the reference and real workplace fields.

Reference Fields

Type	Spectra	Part of F_n, rel. unit., intermed.	Part of F_n, rel. unit., fast	Part of F_n, rel. unit.,>20 MeV	$h^*(10)$, $pSv.cm^2$	E_n, MeV
	PU	0.11	0.89	0.	322.	3.32
	CF	0.13	0.87	0.	284.	1.82
	CH	0.41	0.59	0.	193.	1.36
	FE	0.20	0.80	0.	216.	0.76
	HEF	0.20	0.36	0.44	282.	49.1

Measured workplace neutron fields

Soft, iron	MS[5]	0.79- 0.84	0.16- 0.21	0.	41.-52.	0.11-0.141
	PDSN[7]	0.51- 0.73	0.24 -0.49	0. -- 0.0014	75. 137.	0.6 -5.9
Side shieldig	MS[5]	0.45 -0.67	0.28- 0.47	0.013- 0.04	106-157	0.7 -3.5
	PDSN[7]	0/37- 0.60	0.37- 0.57	0.054- 0.16	156-222	5.7- 47
Top shielding	MS[5]	0.20	0.36	0.44	282.	49.
	ROZ6H[4]	0.22	0.39	0.38	253.	71.

CONCLUSION

Investigations of dosimetric characteristics of the IHEP reference neutron fields have been fulfilled by different dosimeters specially developing for measurements in mixed radiation fields. The ambient dose equivalent data obtained with an accuracy of 10-20%. It is shown that the main spectral characteristics of this set of reference fields are cover a most part of a real neutron spectra behind the side and the upper shielding of the IHEP accelerator. It allows one to using these reference fields for routine dosimeter calibration purposes, dosimeter intercomparisons and other applications as cosmic spectrum simulations.

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ELECTRONIC NEUTRON SENSOR BASED ON COINCIDENCE DETECTION

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INTRODUCTION

The last Symposium on neutron dosimetry which took place in Paris in November 1995 have shown again that it doesn't exist any individual active neutron dosemeter. The state of art on electronic device (1), the needs of the nuclear power industry in individual neutron monitoring (2, 3) and the new trends of researchs (4) were presented. They confirm the relevance of our studies in progress in the C2M team of the University of Limoges. The aim of this work is to realize an individual electronic neutron dosemeter.

The device in the progress of being development will operate either as a dosemeter or as ratemeter giving $H_p(10)$ and $\dot{H}_p(10)$ either as a spectrometer permitting to caracterize the primary neutron beam.

PRINCIPLE OF THE ACTIVE MONITOR

The reactions used are given : - for Boron 10 there are two mechanism

a - ${}^{10}_{5}B + {}^{1}_{0}n \rightarrow {}^{7}_{3}Li + {}^{4}_{2}\alpha$ with Q-value = 2.79 MeV (6%) (E_{li} = 1.01 MeV, E_α = 1.78 MeV)

b - ${}^{10}_{5}B + {}^{1}_{0}n \rightarrow {}^{7}_{3}Li^{*} + {}^{4}_{2}\alpha$ with Q-value = 2.31 MeV (94%) (E_{li} = 0.84 MeV, E_α = 1.47 MeV)

$\downarrow$

$Li + \gamma + 0.478$ MeV

- for Lithium 6

a - ${}^{6}_{3}Li + {}^{1}_{0}n \rightarrow {}^{3}_{1}H + {}^{4}_{2}\alpha$ with Q-value = 4.78 MeV (100%) (E_H = 2.73 MeV, E_α = 2.05 MeV)

The monitor has two parts : the sensor and the electronic for the signal processing. The sensor (fig.1) is composed of two facing diodes which enclose a reactive layer of ^{10}B or ^{6}Li. These elements have a high cross section to low energy neutron. Several parameters influence the response of the device :

1 - The area of the reactive layer is limited by the area of usual wafers available for microelectronic chips. The increasing of layer thickness leads to an absorption of particles from the nuclear reaction that occurs inside the layer, and in order to use the device as a spectrometer that last-mentioned must be thin as possible. However, it's not compatible with the need for high sensitivity necessary in dosimetry.

2 - The thickness of the dead layer of the detecting diodes must be minimised since it always acts as an energy absorber for the recoils.

3 - The thickness of the depleted layer must be adapted to the range of the most energetic particles crossing the various layers.

4 - The bulk of the wafer should not have any influence on the response of the sensor.

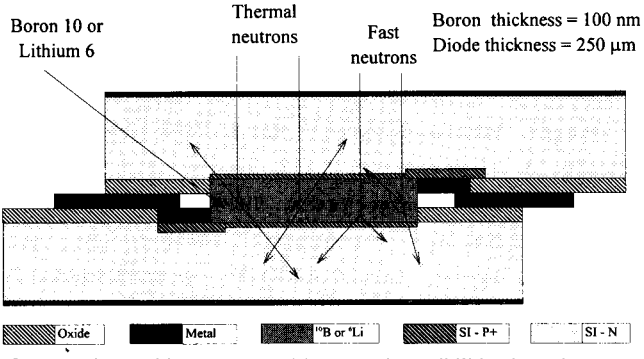

Fig.1 : Schematic view of the detector

Signal processing consists of an usual coïncidence system (5). Several possibilities have been used experimentally to measure the spectra of each detector or, by summing the pulses from both detectors, to get the distribution of the total energy of the nuclear reaction particles.

DESCRIPTION OF THE DETECTION SYSTEM

In order to obtain dose equivalent, we need to know characteristics of neutron beams with different energy ranges because quality factors are function of neutron energy. In these conditions the aim of this work is to realize a rough spectrometer. Next explanations are presented in the case of ^{6}Li reactive layer ; relation between energy is : $E_\alpha + E_{Li} = Q + E_n$ with E_α energy of α particle, E_{Li} energy of ^{6}Li ion, Q value of the energy reaction and E_n energy of the incident neutron.

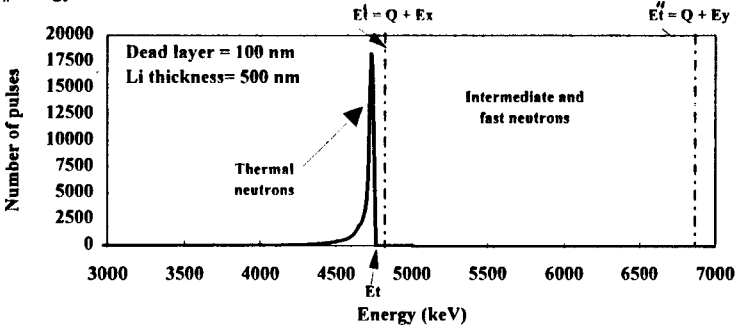

Fig.2 : Spectrum obtained by sommation of the two particles energy and thresholds

Working on spectra obtained by sommation of the two particles energies (fig.2), energy thresholds may be used so that we obtain :

1 - under the energy threshold, $E_T = Q$ (E_T = total energy = $E_\alpha + E_{Li}$) , pulses are due to thermal neutrons,

2 - between two energy thresholds $E_T^{'} = Q + E_x$ and $E_T^{''} = Q + E_y$ ($E_T^{''} > E_T^{'}$) lies spectrum of pulses due to neutrons with energy $E_x < E_n < E_y$. This method is interesting for intermediate neutrons and fast neutrons of low energy.

We have point out that it is necessary to satisfy two conditions :

- $E_x \geq \varepsilon$,where ε is the energy resolution of the device (sensor and electronic system). Electronic are in progress to optimize the resolution.

- $E_y \leq E_M$, where E_M is the maximal energy which allows to consider that in the nuclear reaction ^{6}Li (n, α) ^{3}H the value of angle θ (θ = angle between the two products of the reaction) leads to coïncidence detection.

Modelisation are in progress to define this value.

FIRST EXPERIMENTAL DEVICE AND PERSPECTIVE

In order to show the device feasibility, we used commercially available diodes (Hammamatsu) including a reactive layer (Boron 10) which was realized in the laboratory. The dead layer thickness is larger than 1 µm (which is too high), the depleted zone thickness is 100 µm and the boron 10 layer thickness is 100 nm. The device was tested thanks to an AmBe source with paraffin shielding. The registered spectrum is presented on figure 3. Two peaks appear for α particles and two other peaks for Li ions. This result is in agreement with the response obtained by simulation.

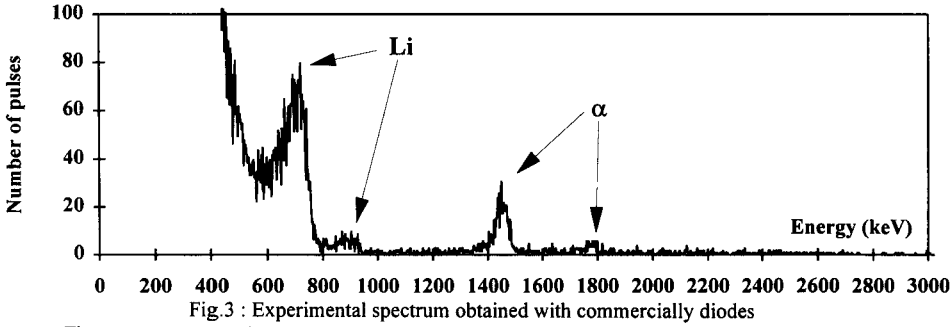

Fig.3 : Experimental spectrum obtained with commercially diodes

The sensor response has been modeled thanks to calculation codes developped in our laboratory and it appears that reaction with ^{6}Li reactive layer is more interesting because it doesn't present an excited state and the

energy of particles is higher than in ^{10}B case. But a technological difficultie exits for Li which is able to diffuse inside silicium bulk of the diode. So, it is impossible to use a ^{6}Li reactive layer.

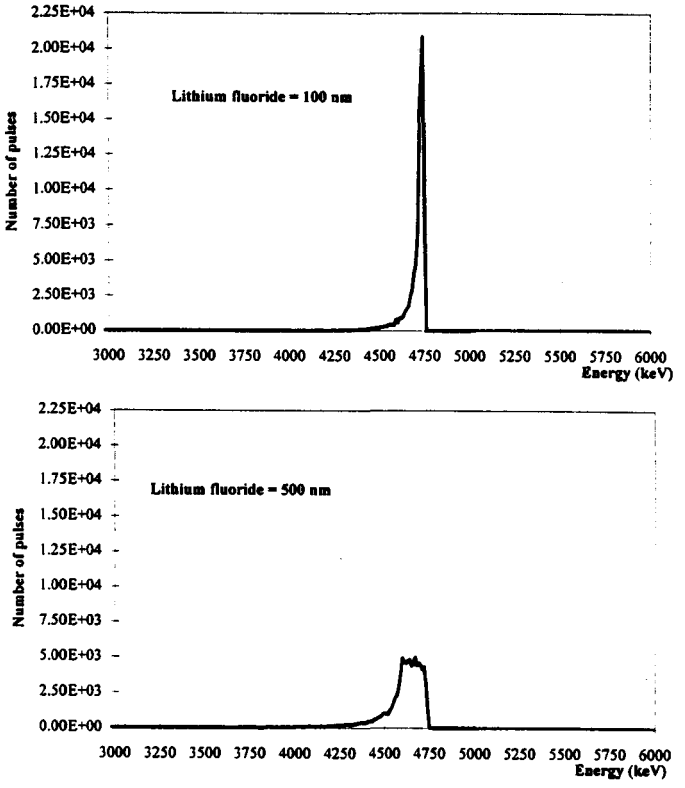

Fig.4 : Distribution of the total energy deposited in the sensor for 100 nm dead layer thickness and a LiF reactive layer

In these conditions, it is necessary to realize a reactive layer with a compound which contains ^{6}Li. We propose LiF. The modelization was made and the first results are given figure 4.

In conclusion, it appears that LiF will be used as reactive layer in the coïncidence device and technological realization of LiF layer is in progress.

ACKNOWLEDGEMENTS

This work was supported by the Commission of European Community (contract FI3PCT9 30072) and EDF - France (Comité de radioprotection). We are very grateful for these helps.

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DEVELOPMENT OF A DUAL TYPE IONIZATION CHAMBER SYSTEM FOR BURST n-X MIXED FIELDS

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INTRODUCTION

In the vicinity of high temperature plasma experiments for nuclear fusion studies, mixed radiation fields which are composed of X-rays and neutrons are expected to arise as burst-like pulses according to plasma shots. A typical operational condition is such that several-to-ten seconds plasma shots are given repeatedly in every ten minutes or so. It is therefore important from radiation protection point of view to have a monitoring sensor effective for real time and separate measurements of X and n components with a good dynamic range.

For that purpose we have designed and tested a dual type ionization chamber system to be used as area monitors around the coming Large Helical Device (LHD)[(1)] of NIFS. This might also be applicable to other case, for instance, radiation fields surrounding accelerator experiments. This system is composed of a pair of cylindrical vessels of the same size and shape, one containing ^{3}He gas sensitive to both n and X, and the other ^{4}He insensitive to neutrons. The ionization chambers have been examined with various kinds of radiation sources[(2),(3)]. In the present study, we have carried out further irradiation experiments and discussed about the results.

CHAMBER SYSTEM

As the radiation dose sensor we have adopted the ionization chamber scheme to obtain the response in the current form, since it is required to cover a rather wide range of neutron fluence levels, sometimes accompanied by much stronger X-ray components, without suffering from the pile-up counting failure. In order to distinguish neutron contributions from X-rays, a dual-chamber system is proposed.

A dual type ionization chamber system is consisting of two cells of the same geometric shape, volume and wall material. The system turns the difference of reaction between isotopes to account. One of them is filled with pressurised ^{3}He gas, being sensitive to both neutrons and X-rays, and other filled with ^{4}He gas being sensitive only to X-rays. It will be possible to find out the neutron contribution by subtraction of the signal.

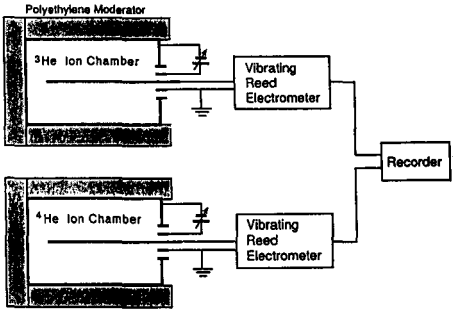

Figure 1. Chamber system.

Each chamber cell is a cylindrical vessel (80mmϕ x 200mm) of stainless steel (3mm thick) to give a gas volume of 1 liter. Measurable current range is 10^{-5}A $\sim 10^{-14}$A with a moderate time constant. Chamber cells with ^{3}He and ^{4}He gases of different pressure (1,2,3 atm) are prepared. Since we are interested in neutron energies up to a few MeV, a detachable polyethylene moderator (5 cm thick) is usually placed surrounding the chamber cells.

IRRADIATION EXPERIMENTS

(1) Responses to gamma rays ; The chamber characteristics irradiation against ^{60}Co gamma rays has been

carried out. The field intensity is varied by means of changing the source distance or source intensity. The response of ^{3}He and ^{4}He are quite similar to each other. Gas pressure dependence of the saturation current levels was given, there seems to hold good proportionality as expected. An apparent linear relation is attained to cover 8 orders of magnitude from 10^{-5} Gy/h to 10^3 Gy/h.

(2) Responses to thermal neutrons ; The neutron response of ^{3}He chamber has been surveyed by means of D_2O themarised neutron irradiation at the KUR fission reactor of Research Reactor Institute, Kyoto University. The chambers were placed about the center of the irradiation room, where is 1.5 m distant from Bi surface. The reactor power has been changed from 0.1 kW to 30 kW. The thermal neutron flux at this position is estimated by a different experiment as $0.96 \sim 1.88 \times 10^4$ n cm^{-2} s^{-1} at 1 kW operation, so that the observed ^{3}He current 0.35 nA should imply the sensitivity of the 1 atm ^{3}He chamber with moderator to be $0.019 \sim 0.036$ pA/(n cm^{-2} s^{-1}). Both output currents from ^{3}He and ^{4}He chamber are closely proportional to the reactor operating powers, or fission neutron yields, though the current ratio ^{3}He /^{4}He is high as $\sim 10^3$.

(3) Testing with mixed n-X fields ; Response to source distance characteristics have been examined by means of ^{252}Cf and ^{241}Am-Be whose neutron emission rate were 1.5, 3.6, 5.1×10^5 n/sec and 4.66×10^6 n/sec, respectively. The ^{3}He response reveal $r^{-1.8}$ or $r^{-1.7}$ decrease with distance r. This trend may be due to the increasing of thermalised neutron component or incidence of scattered neutron to the flank. For the 1 atm ^{3}He chamber with moderator, response currents against ^{252}Cf and ^{241}Am-Be are evaluated as 0.07 and 0.05 pA/(n cm^{-2} s^{-1}).

In addition, in order to estimate the effect of moderator, polyethylene blocks have been put on between ^{252}Cf source and ^{3}He ionization chamber. When 50 mm thick moderator was put on, the sensitivity of the chamber was recorded the highest value.

DISCUSSION

The ^{4}He response to several sources those are ^{60}Co, KUR(D_2O), ^{241}Am-Be are plotted in Fig.2, showing very good fitting to the line. This implies that the small amount of gamma rays coexisting with neutron can easily be separated with the dual-chamber.

Table 1 shows the sensitivity related with ^{3}He gas pressure and moderator. This response ratio means output current ratio to that of the 1 atm ^{3}He chamber with moderator.

Table 1. Sensitivity related with the ^{3}He gas pressure and moderator.

		Response ratio	
Neutron source	moderator	1 atm	3 atm
Thermal neutron	5 cm	1	1.87
	0 cm	6.26	13.1
^{252}Cf	5 cm	1	1.82
	0 cm	-	0.09
^{241}Am-Be	5 cm	1	1.82
	0 cm	0.04	0.11

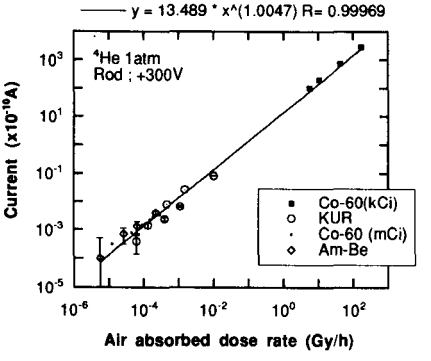

Figure 2. Responses to gamma rays, (1 atm ^{4}He chamber).

In order to compare the experimental result with estimation, ^{3}He reaction rate has been calculated by means of ANISN which is one dimensional discrete ordinates neutron transport code. The conditions of the calculation have been changed about type of neutron source, ^{3}He gas pressure, thickness of polyethylene moderator.

The 3 atm chamber generates 1.8 times current than the 1 atm one, as shown in Table 1. This result can not

be explained by one dimensional calculation. This is related with mean free path of neutron in the chamber. The shape of the chamber is cylinder. From two dimensional view, the chamber is circle. The thickness of that is the same as the diameter at the center, and becomes thinner when recede the center point. When neutrons go into the 1 atm chamber from one direction uniformly, 48 % of neutron collides with ^{3}He, the remainder go through it. On the other hand, 96 % of neutron make reaction with ^{3}He in the case of the 3 atm one. These calculation can explain the results of the experiment.

When gas pressure in the chamber is made higher, the sensitivity for X(gamma)-rays is increased correspondingly, but that for neutron is not risen so much. Even if the gas pressure is 7 atm, the sensitivity is ascended only 1.02 times as large as that of 3 atm. Therefore, the gas pressure in the chamber is rather considered for the sensitivity of X-rays.

The neutron signals from ^{3}He are due to a thermal capture process which is high sensitive for thermal neutron and low for fast neutron. It is important to select a suitable thickness of moderator. A polyethylene moderator gives reducing energy, shielding and reflecting for neutron. As shown in Table 1, moderator performs only as neutron shielding in the thermal region. The same value has been given by the calculation of ANISN. On the other hand, for the fast neutron, moderator slows down neutrons to thermal neutron. The result of calculation indicates that the 5.5 cm thick moderator gives the highest sensitivity for a few MeV neutron. The ^{252}Cf irradiation test shows the same consequence as this.

Fig. 3 shows the response characteristic of ^{3}He chamber against neutron energy. By the effect of the moderator, the sensitivity of ^{3}He chamber becomes quite uniform for neutron energies up to a few MeV.

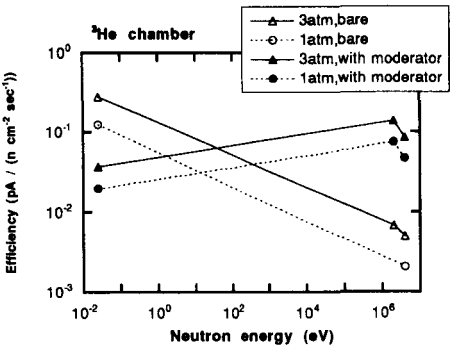

Figure 3. Response characteristic of ^{3}He chamber against neutron energy.

SUMMARY AND REMARKS

A dual type cylindrical ionization chamber system has been developed for dosimetry use of neutrons and X(gamma) rays. Several tests so far carried out in various field conditions seem to show good capability of the system.

Task and issues left for the future : (i) Tests in the burst-like pulse radiation fields (use of existing accelerators or plasma devices) using fast responsive electrometer (ii) n-X separation test in the field where n/X is small.

ACKNOWLEDGEMENT

The authors thank Prof. T. Tsujimoto and T. Yoshimoto (Research Reactor Institute, Kyoto University) for their kind arrangements in making the irradiation tests.

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DEVELOPMENT OF NEUTRON MONITORING SYSTEM IN n - γ MIXED FIELD WITH A TWIN-TYPE $^{10}BF_3$ - $^{11}BF_3$ IONIZATION CHAMBER

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INTRODUCTION

A neutron monitoring system that could measure the contribution of neutrons separately from that of γ-rays in a n-γ mixed field has been developed in order to evaluate such a mixed field as encountered around a thermonuclear fusion reactor, a particle accelerator and a nuclear fission reactor. The characteristics of the detectors have been pre-checked with an Am-Be neutron source and then the experiments have been carried out around a target of the electron linear accelerator installed at the Radiation Laboratory in Osaka University and also in the D_2O facility of the research reactor in Kyoto University. In the former case the radiation field consists of burst X-rays and fast neutrons, which could simulate the field around a proposed fusion reactors such as Helical or Tokamak. On the other hand, in the latter case the field consists of stationary thermal neutrons and γ-rays, and it would be useful for the detector system to be tested in simulating the field of environmental level.

The main purpose of the present experiments is to establish a detecting system of a wide dynamic range that could measure all kinds of n-γ mixed fields, that is, various fluence rates and mixed ratios which could be realized just near the radiation generator and also far from it, namely the environmental level.

EXPERIMENTAL APPARATUS

In order to detect neutrons separately from γ-rays it is essential to explore such materials possessed with different sensitivities for neutrons and γ-rays. Namely, one is sensitive to both neutrons and γ-rays, and the other is only to γ-rays. The difference between the two output signals could correspond to that of neutrons. In case of a gas counter ^3He and ^4He gases are usually adopted. In the present experiments, however, a twin-type ionization chamber filled with $^{10}BF_3$ and $^{11}BF_3$ gases in each chamber has been prepared. The chamber is of a disk shape, diameter and thickness of which are 16 cm and 6 cm, respectively, and a rod-type electrode is inserted to each chamber. The applied voltage has been so adjusted as could be operated within the ionization chamber region. In the $^{10}BF_3$ chamber the thermal neutrons would react with ^{10}B and the emitted α-particle and ^7Li-particles would ionize the $^{10}BF_3$ gases. This chamber, however, also acts as a usual ionization chamber for γ-rays, and hence the output signal is attributed to both

neutrons and γ-rays. On the other hand, as $^{11}BF_3$ chamber is insensitive to neutrons, the output would correspond only to γ-rays, that is, the exposure due to γ-rays could be measured without the influence of neutrons.

With this twin-type chamber the mixed field of high-fluence rate could be measured by the output current, while that of an environmental level by the counting method.

EXPERIMENTAL RESULTS AND DISCUSSIONS

In the beginning the experiments have been carried out with the Am-Be neutron source to obtain the fundamental characteristics of the twin-type chamber. In Fig. 1 are shown the output currents of both $^{10}BF_3$ and $^{11}BF_3$ chambers as a function of the distance from the source, where the dose equivalent rate due to γ-rays evaluated with an air-equivalent ionization chamber is also shown. It is found that both signals of $^{11}BF_3$ and ionization chamber are possessed with the same attenuation gradient, that is, they would satisfy the inverse square law of distance. The gradient of the output signal from $^{10}BF_3$ chamber which correspond to (n + γ) signals is a little larger than the other's. This is probably due to the scattered thermal neutrons coming from all directions around the chamber. As the difference of both signals could clearly be observed, the contribution due only to neutrons would be found out by taking subtraction between both signals.

In Fig. 2 are shown the experimental results with this chamber set around the target of the electron linear accelerator. The energy, the peak current and the repetition rate of the electron beam are 28 MeV, 1 A and 60 pps, respectively, and the material of the target is Ta. The angular dependence of the output current from both chambers set at 3 m from the target is shown in the figure. The signals of both chambers increase towards the beam direction, which corresponds to the fact that the intensity of the Bremsstrahlung X-rays is extremely intense in the forward direction of scattering. However, the subtraction of both signals comes to be flat, that is, the neutrons generated in the Ta-target by the X-n reaction reveal no angular dependence.

In Fig. 3(a) and (b) are shown the experimental results obtained with those chambers set inside the D_2O thermalized neutron irradiation facility of the fission reactor. It is found that both output currents due to thermal neutrons and γ-rays increase linearly with the reactor power and there exists difference of about two orders of magnitude between both signals. The linearity of the exposure due to γ-rays for the reactor power has also been ascertained with an air-equivalent ionization chamber.

CONCLUSION

The twin-type $^{10}BF_3$ -$^{11}BF_3$ ionization chamber has been operated in such various n-γ mixed fields as of Am-Be neutron source, as around a target of the electron linear accelerator and as inside the D_2O thermalized neutron irradiation facility of the research reactor. As a result it has been confirmed that both neutrons and γ-rays could be evaluated independently in the mixed field with satisfactory accuracy by taking subtraction of both output signals.

The present measuring system will be expected to be useful to evaluate the mixed field that would appear around various kinds of particle accelerators and also proposed large-scale devices for the thermonuclear fusion.

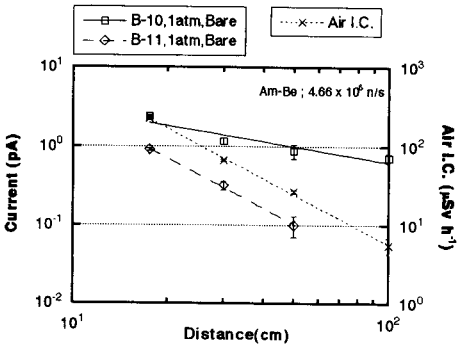


Fig. 1 Dependence of output currents of both $^{10}BF_3$ and $^{11}BF_3$ chambers on distance in the n- γ mixed field of Am-Be neutron source. The dose equivalent rate due to γ -rays measured with ionization chamber is also shown.

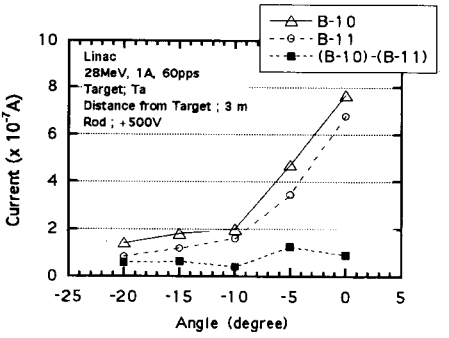
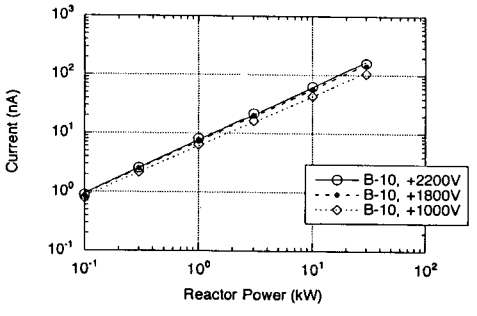
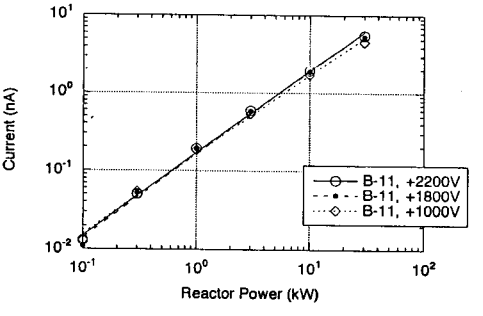


Fig. 2 Angular dependence of output currents of both $^{10}BF_3$ and $^{11}BF_3$ chambers operated around a target of an electron linear accelerator. The subtracted data correspond to the neutron component.



(a) $^{10}BF_3$ chamber



(b) $^{11}BF_3$ chamber

Fig. 3 Dependence of output currents of both $^{10}BF_3$ and $^{11}BF_3$ chambers operated inside the D_2O facility of the research reactor on reactor power.

SEMICONDUCTOR DOSIMETERS FOR SELECTIVE DETERMINATION OF THE COMPONENTS IN MIXED GAMMA AND NEUTRON FIELDS

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INTRODUCTION

In order to estimate doses received by people in a case of nuclear accident it is important to have convenient sensors of mixed radiations and to discriminate heavy particles from ionizing radiation, becouse their biological effects are different. Semiconductor devices are widely used for such purposes. In comparison with traditional sensors (ionization chambers, Geiger-Muller counters,TLD and track detectors), semiconductor dosimeters have certain advantages because of their high sensitivity, small sizes and weights and simple readout. They can work in both passive and active modes, can store information for a long time, and can operate in count integrating mode. They register different kinds of radiation over wide ranges of doses and energies and are useful for accident and for routine dosimetry as well.

Possible use of semiconductor devices for dosimetry were described previously (for instance,MOS transistor for gamma (1,2) and PIN diode for neurton (3,4) - irradiation).New gamma and neutron sensors have been produced in our Institute and their main dosimetric parameters have been published in (5,6).

The purpose of this work was to describe all functional opportunites of PIN and MOSFET dosimeters, designed in our Institute.

PRINCIPLES OF SEMICONDUCTOR DOSIMETERS AND EXPERIMENTS.

The non-ionizing loss induced by neutrons in the bulk of silicon structures generates atomic displacements leading to defects in the crystal. These act as generation-recombination centers. PIN structures have relatively small sensitivity to gamma-rays , which produce few displacements, so they can be used for neutron radiation measurement only. Extensive studies have been performed on the effects of neutrons on the behaviour of PIN structures. The variation of forward voltage under fixed carrent shows a simple dose dependence by the equation:

$$\Delta U_f = U_f - U_0 = kD \qquad (1)$$

where U_0 and U_f are the initial bias and the bias after irradiation, respectively; D is the neutron dose. Sensitivity k has a complicated dose dependence on minority carrier lifetime τ_0 , resistivity ρ and diode geometry and can be expressed as:

$$k = dU/dD = k_\tau \tau_0 f(d/L) \qquad (2)$$

were k_τ is the damage constant, d is the base thickness and L is the diffusion length.

The known PIN diodes for dosimetry (4) were made from p- and n-Si ($\rho \sim 10 \div 100\ \Omega$ cm) and have been studied at a high level of injection, where damage constant depends on lifetime only. The authors found a value of k = 100 mV/Gy and dose dependence was linear up to 12 Gy.

We have studied (both theoretically and experimentally) PIN diodes on the high-resistivity base under intermediate injection level (6). In this case one can vary three parameters (d, L, ρ) to achive the best neutron sensitivity.

Thus for PIN diodes on the base of n-type silicon we have received a higher sensitivity over a wider dose range in comparison with a case of high level of injection (for $\rho \sim 1k\Omega$ cm, d=0,12÷0.3cm, k=0.1÷1.0 V/Gy, dose range from 0.1 Gy to 100 Gy; for $\rho \sim 40k\Omega$ cm, d=0.5cm, k=5V/Gy, dose range (0.005 ÷3.0)Gy).

The discrimination coefficient to gamma irradiation in mixed field is ~1000.

An accuracy of neutron dose determination depends on a temperature instability coefficient (TIC). For PIN diodes on the base of high resistivity silicon TIC is smaller and decreases with neutron dose in contrary with diodes on the low-resistivity base.

PIN sensors described above have been tested in count integrating mode under ^{137} Cs and ^{139} Ce gamma irradiation and had readout cut-off ~40 keV. Lower limit for gamma dose rate was 0,7 msv/h. Radiation defects created by fast neutrons do not effect significally on count characteristics up to neutron dose 10 Gy.

Proposed PIN diodes may be usful to measure the gamma dose rate and the accident fast neutron dose at the same time.

The mechanism of radiation sensitivity of MOS structures have been studied in details by many authors (1,2). A positive charge is stored in the gate oxide and the radiation-induced charge on the $Si-SiO_2$ interface appears under ionizing radiation. This leads to the threshold voltage shift, ΔU_t, for MOSFET or flat-band shift, ΔU_{FB}, for MOS capacitors. The built- in charge in the oxide and, therefore the value ΔU_t, depends on the concentration of hole traps, their space and energy distribution and the gate bias during irradiation. This dose dependence can be expressed:

$$\Delta U_t = (\Delta U_t) \max [\, 1 - \exp (- \beta D_\gamma \,)], \qquad (3)$$

where sensitivity β depends on the oxide structure and is proportional to $N_t d^2_{ox}$, d_{ox} is the oxide thickness, N_t is the trap concentration. In order to get higher sensitivity one must increase d_{ox}.

Our previous measurements have been extended to n-channel MOSFET sensors on the base of p-Si substrate with resistivity 10 Ω·cm and oxide thickness 1 μm.

MOSFETs channel lengths have been varied from 10 to 50 μm.The dosimetric sensitivity to ^{137} Cs gamma irradiation was 150 mv/Gy in a passive mode and did not depend on channel length up to 40 μm; For standard radiotechnical transistor with 0.1μm oxide thickness β is three orders of magnitude less.In an active mode (with the positive gate bias V_g) the sensitivity is higher and varies as $\beta \sim V_g^{2/3}$.

Irradiation temperature (-60°÷60°C) does not effect on the sensitivity both in passive and active modes.

Studies of MOSFETs irradiated at the high dose rate (with pulse gamma rays source) has showed that sensitivity was practically independed on the dose rate up to 10 Gy/s.

Temperature instability coefficient depends on readout current. We have found experementally thermostability current 43 μA, when TIC=0.

A coefficient of discrimination to neutrons in mixed radiation field is ~100 and strongly depends on dosimeter encapsulation (7).

For dose measurement applications it is important to know the fading of devices.

Changes of voltage shifts with time for our MOSFET sensor are ~15% in two months storage, while the main change takes place during the first hour after irradiation (as in the case of PIN diodes).

The MOSFET dosimeters described above have been tested on the 3-d block of Tchernobyl atomic pover station after accident and have shown good agreement with results for standard TLD dosimeters (see table).

Gamma ray dose (rad)		
MOSFET dosimeters	Termoluminiscence dosimeters	Place of exposition
599	450±30	roof
365	370±20	roof
253	247±15	roof(Pb-3mm)
247	320±30	bulding, point 12
300	320±30	bulding, point 12
318	320±30	bulding, point 12
897	900	roof
394	353±13	bulding, point 12

Our next step was to combine PIN diode and MOS transistor in one sensor. For this purpose p-channel MOSFET with thick gate oxide and high resistivity base have been taken. Source-base and/or drain-base give P-N diodes with a linear response, changes in their current-voltage curves under irradiation could be used to measure neutron dose. Influence of gamma ray and neutron irradiation on dosimetric characteristics of such p-MOSFET has been investigated. The sensitivi to ^{137} Cs gamma rays was 100 mV/Gy , to fast neutrons of nuclear reactor ~ 30 mV/Gy. The last value may be increaced by using MOSFET with a thicker base.

CONCLUSION

Silicon dosimeters described above were developed for reliable selective determination of gamma and neutron radiation at accident , routine measurements and medical application:

1. PIN diodes on high resistivity base for fast neutron dose measurement, n-MOSFET transistors for gamma dosimetry.
2. PIN diodes to measure gamma dose rate in routine dosimetry and accident fast neutron dose at the same time.
3. Accident dosimeter for mixed radiation fields, using PIN diode together with MOSFET transistor.

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A MODERATOR TYPE NEUTRON DOSEMETER BASED ON SILICON DIODES

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ABSTRACT : A moderator type neutron dosemeter is described and its response to realistic neutron fields from C.E.N. Cadarache-France is investigated, as well as its sensitivity to the gamma rays generated along with the realistic neutron fields.

Changes in response induced by the use of a moderator are observed throughout the experiments, with respect to the energy distribution of the neutron fluence and the dose equivalent. The Silicon detector sensitivity to the gamma ray component of these fields is also discussed.

Compared to an electronical neutron dosemeter without a moderator, for the same active area of the Silicon detector, the actual design shows a response 10 to 40 times higher to realistic neutron fields and a sensitivity to gamma rays up to 10 times lower.

INTRODUCTION

Previous work in the field[1,2,3,4] showed that the main problems in designing a neutron dosemeter based on Silicon diodes consist in finding the means to provide high gamma ray discrimination and to improve the response in the intermediate neutron energy range. A new approach to these problems is made using 2 Silicon detectors of 19.6 mm^2 active area surrounded by a small-size moderator.

While moderator type neutron equivalent monitors using passive detectors[5] require full moderation of neutrons over the whole neutron energy range (from thermal to 14 MeV), the use of Silicon detectors led to the concept of a "selected" energy range for moderation, which consist in moderating neutrons up to a certain energy and detecting higher energy neutrons directly as fast neutrons, using the same detector.

This is possible by using a polyethylene foil radiator deposed with a ^{10}B layer, positioned on the active surface of the Silicon diode. This radiator allows the detection of thermal neutrons through the $^{10}B(n,\alpha)^{7}Li$ reaction and fast neutrons detection through protons issued by fast neutron elastic scattering on Hydrogen atoms in polyethylene.

This "selected moderation" of neutrons requires moderators smaller in size than those used for "full moderation", allowing their use for personal dosimetry.

The use of a small size moderator has also the advantage of increasing the potential detection area of the dosemeter and allows a higher low amplitude threshold in order to discriminate pulses due to the gamma ray background.

THE MODERATOR TYPE NEUTRON DOSEMETER BASED ON SILICON DIODES

The detection assembly (moderator + diodes) shown in Fig.1 consists of 2 implanted Silicon diodes, having an active area of 19.6 mm^2, one of the diodes being provided with a polyethylene radiator deposed with 2 μm layer of B_2O_3 (96% ^{10}B enriched).

The pulses from the diodes are fed through two acquisition chains and then treated by a differential method.

Each diode is covered with an Aluminium shielding (0.3 mm thick), and both diodes are included in a hemispherical moderator provided with a cylindrical opening (Fig.1) intended to provide a direct detection for fast neutrons at normal and near normal incidence.

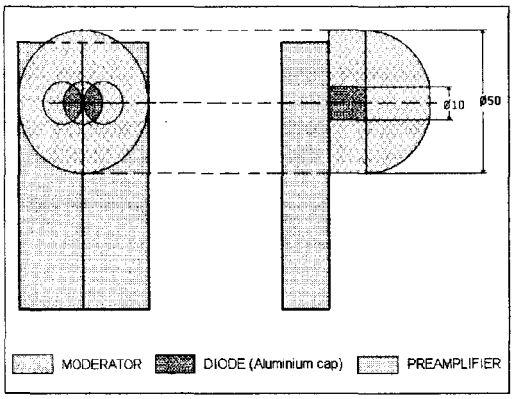

Fig.1. The detection assembly with moderator.

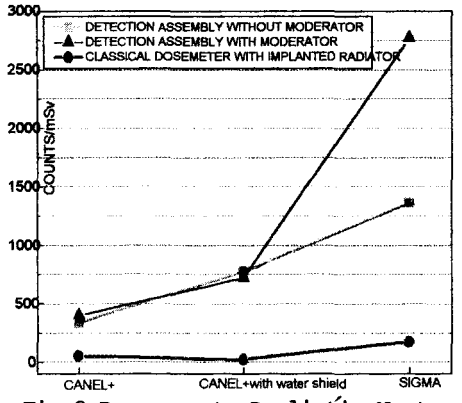

Fig.2 Response to Realistic Neutron Fields (Cadarache)

The low amplitude threshold of the system was at 100 keV and an informatic cutoff was imposed at 290 keV in order to discriminate gamma rays generated pulses up to a sensitivity of 10 counts/mSv for a ^{137}Cs gamma ray source.

THE IRRADIATION TESTS

Irradiation tests have been performed using the "Realistic Neutron Fields" from CEA/CEN Cadarache-France. The main parameters characterizing these fields are indicated in Table No.1, data being provided[6] by CEA/CEN - Cadarache and CEA/ IPSN/ DPHD Fontenay-aux-Roses. The following parameters are indicated: the spectral distribution of the neutron fluence Φ and neutron dose equivalent H for three neutron energy ranges : thermal (0.025 eV - 1 eV), intermediate (1 eV - 10 keV) and fast neutrons (10 keV- 14 MeV); the average neutron energy E(keV); and the effective energy E_H(keV), which is the dose equivalent average neutron energy.

Characteristics Neutron Field	Φ_{th} %	Φ_{int} %	Φ_{fast} %	E keV	H_{th} %	H_{int} %	H_{fast} %	E_{eff} keV
CANEL+	36	20	44	150	6	3	92	610
CANEL+ with water	43	26	31	80	10	7	83	68
SIGMA	97	1	2	72	46	4	50	1323

TABLE No.1.: Characteristic data for the Realistic Neutron Fields from Cadarache (France).

Irradiations of the detection assembly with and without moderator have been performed at normal incidence for each neutron field, with and without a PMMA phantom.

For each measurement, it was provided the value of the ambient dose equivalent H*(10) obtained from neutron fluence monitoring and the values of the corresponding fluence to dose conversion factors, as indicated by ICRP 60.

RESULTS

Dosemeter's response for the irradiation tests described above is shown in Fig. No.2, and is compared to the response of a differential detection system with two PIPS detectors, provided only with a radiator (and no moderator), developed at CEA/IPSN/DPHD/SDOS Fontenay-aux-Roses[2]. To enable comparison, the Silicon diodes active area has been normalized to 19.6 mm^2. In the diagram (Fig.No.2) the three type of fields (CANEL+, CANEL+ with water and SIGMA) were placed in an order reflecting their content of thermal neutrons. A statistics on the number of the pulses counted and an

investigation of their spectral distribution to observe the influence of the moderator to the dosemeter response, is shown in Fig.3 and Fig.4.

A part of this study is focused on the dosemeter response to the gamma rays generated along with the realistic neutron fields.

The response to gamma rays and the eventual direct neutron interactions with Silicon atoms in the diode were observed as the response on the diode without a radiator .

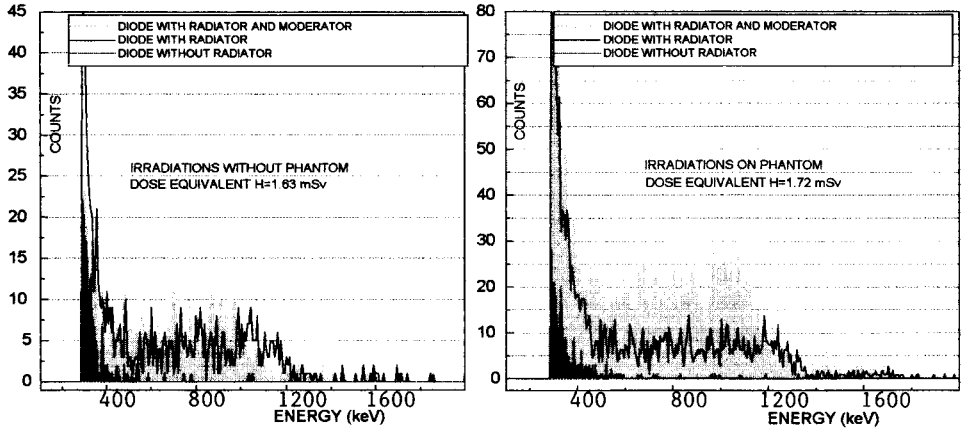

Fig.3 and Fig.4 : Cadarache irradiations on CANEL+with water shield. Pulse height spectra.

DISSCUTION AND CONCLUSIONS

For thermal neutrons and on phantom measurements it was obtained an over response when using the moderator, which is an expected result. A Cadmium shielding of a certain extent on the moderator's surface could overcome this inconvenient.

In the intermediate neutron energy range the sensitivity of the moderator system is higher than that obtained for the "classical system" using only the radiator. This is mainly due to the higher ^{10}B concentration in the radiator, but also to the presence of the moderator.

In conclusion, the moderator type neutron dosemeter investigated shows a high sensitivity for neutrons in the intermediate energy range where other detection systems using Silicon detectors proved to have a gap in response.

The small active area of the Silicon detectors, as well as the differential system used allows a very low gamma ray sensitivity of the detection system (10 counts/mSv), but it has also the disadvantage of providing a low sensitivity to fast neutrons.

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THE THERMOLUMINESCENCE CRITICALITY DOSIMETRY AT COGEMA

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1 - Introduction

Through its activities, and in particular in enrichment, fabrication and nuclear fuel reprocessing, COGEMA has to master the dosimetry in case of a criticality accident event.

The COGEBADGE®, equipped with a film and a 4 dosimeter (2LiF600, 2 LiF700) thermoluminescent card provides with satisfaction routine dosimetry for beta, gamma and neutron rays.

Other COGEBADGE® éléments (ebonite, gold (naked and cadmium shielded)) are also available for criticality dosimetry. This personal dosimetry system is completed by an area one, using the SNAC 2, a neutron spectrometer by target activation and counting.

The counting and spectrometry operations of the targets last for a long time in order to obtain the dosimetric quantities, neutron, kerma and photon dose.

So, COGEMA in collaboration with IPSN (National Safety and Protection Institute) and ETCA (Etablissement Central de l'Armement) have carried out a series of critical experiments, using the SILENE reactor (in VALDUC Nuclear center) especially devoted to this task.

In the SILENE cell, different screens in nature and size, have been disposed in order to simulate as closer as possible a criticality accident event. Two configurations have been used, unshielded and lead shielded Silene. The main goal has been to test the thermoluminescence dosimetry capacity in the field of the criticality personal dosimetry.

2 - Procedures and results

The reference values (neutron kerma and photon dose) have been given by the IPSN and ETCA.

A special procedure has been set off for the dosimeter treatment. It uses, in particular, a light signal attenuation filter (1/100) in the reading procedure.

Results

For the different configurations studied, giving rise to a ratio (neutron kerma/photon dose) ranging from 0.21 to 4.17, the results are as follow :

the neutron kerma response Rn (calculated/reference) = 1.03 ± 35 %
the photon dose response Rγ (calculated/reference) = 1 ± 15 %

3 - Conclusion

In regards to the above described experiences and results obtained, the thermoluminescence technics allows a first assessment of the dosimetric quantities in a rapid and simple way.

One hour after the arrival of the dosimeters in the laboratory, the dosimetric quantities, for about 50 persons, at the location of the dosimeter, can be obtained under the following conditions :

kn (Gy) within ± 35 %

Dγ (Gy) within ± 15 %

24 hours later, these above performances are improved by the results coming from the use of the different targets (ebonite and gold) of COGEBADGE® and of the criticality belt.

A STUDY ON NEUTRON AND GAMMA-RAY RESPONSES OF LABORATORY MADE LiF:Mg,Ti SINGLE CRYSTAL TLD

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ABSTRACT

The main objective of this study is to determine the neutron and gamma-ray sensitivities of laboratory made LiF:Mg,Ti thermoluminescence dosimeter (TLD) for practical application in neutron-gamma mixed field dosimetry.

INTRODUCTION

In recent years the application of TLD in neutron and gamma-ray mixed field has steadily increased to assess the dose burden of radiation workers. The dosimetry of neutrons in a mixed field is not as simple and precise as that of pure gamma-ray fields due to energy transfer processes and to variation in the reaction cross-sections with energy (1). In the presence of a mixed field (neutron+gamma), the separate evaluation of the neutron and gamma-ray contributions is essential. Thermoluminescence (TL) dosimeters are now extensively used in radiation dosimetry. Numerous investigations for the development of TL materials have been performed for mixed field radiation dosimetry (2-4). The most frequently used TLD in neutron and gamma-ray mixed field dosimetry is LiF:Mg,Ti. The isotopic constituents of the LiF:Mg,Ti family are: ^{6}Li 7.5% and ^{7}Li 92.5% for the TLD-100, ^{6}Li 0.01% and ^{7}Li 99.99% for the TLD-700 and ^{6}Li 95.62% and ^{7}Li 4.38% for the TLD-600. It is well known that TLD-700 is practically insensitive to neutrons and TLD-600 is sensitive to both neutrons (to a greater extent) and gamma-rays. TLD-100 is also sensitive to both neutrons (to some extent) and gamma-rays. The LiF:Mg,Ti (equivalent of commercially available TLD-100) single crystal of natural isotopic composition has been developed in the laboratory in order to use it for radiation dosimetry . The LiF:Mg, Ti dosimeters have been used for gamma-ray dosimetry (4-6). The objective of this study is to determine the neutron and gamma-ray sensitivities of laboratory made LiF:Mg,Ti single crystal for practical application in radiation monitoring in neutron and gamma-ray mixed field.

EXPERIMENTAL METHODS

The single crystal of LiF:Mg,Ti has been produced at the Atominstitut der Osterreichischen Universitaten,Vienna, Austria using the Czocharalaki technique (7). LiF:Mg,Ti single crystal, totally transparent, measures about 6mm x 6mm x 1mm in size. These TL dosimeters were submitted to the standard pre-annealing thermal treatments; heating at 400 ^{o}C for one hour followed by 2 hours heating at 100 ^{o}C. The photon irradiations were performed at the Secondary Standard Dosimetry Laboratory (SSDL), AERE, Savar with the standard radiation sources. Two gamma sources were used: ^{137}Cs and ^{60}Co, with gamma energies of 662 keV and 1250 keV$_{eff}$, respectively. The X-ray generator was used to produce normalized X-ray beams of 33, 48, 65, 83, 100 and 118 keV$_{eff}$. The collimated beams are horizontal, the dosimeters are positioned on the calibration bench at a distance of 1m by means of a laser beam. The

beams were calibrated with two secondary standard chambers NE2571 and NE2575. Thermal neutron irradiation was performed at the thermal column of the 250 kW TRIGA Mark-II research reactor at Vienna. The themal neutron flux was measured by using a gold foil activation technique. Fast neutron irradiation were performed in the ^{252}Cf (2.1 MeV), Am-Be (4.5 MeV) and neutron generator (14 MeV) neutron fields. The neutron doses were measured using a rem counter. After irradiation the TL responses were read by means of a laboratory made TLD DAT-II and a Harshaw TL reader Model 4000, using a linear heating rate of 10 °C/s from room temperature to 300 °C with a constant flow of nitrogen (about 3 l/min). No further treatment procedure was used in the period after irradiation and before evaluation, because all measurements of the irradiated TLDs were made at least 48 hrs after irradiation to allow decay of low temperature peaks. The responses to thermal neutrons were calculated after subtracting the gamma-ray contributions. A computer aided procedure was set up in order to store data and to display the glow curve. The LiF:Mg,Ti single crystal samples are totally transparent so that the emission of thermoluminescence light during readout is much larger compared to other similar lithium fluorides (TLD-100 and TLD-600) which are totally or partially opaque. The sensitivity was calculated by taking into account the integrated light from the most important peak only. The detailed procedures for calibration and messurement are described elsewhere (5).

RESULTS AND DISCUSSIONS

The shape of glow curves for gamma-rays from ^{60}Co source and neutron+gamma-ray mixed radiations from reactor are shown in Figs.1a & 1b. Two main peaks at 200 °C and 250 °C are evident in case of gamma radiation from ^{60}Co source (Fig. 1a). The peak at 200 °C has to be considered as the main peak whereas peak at 120 °C appears as a shoulder of the main peak on

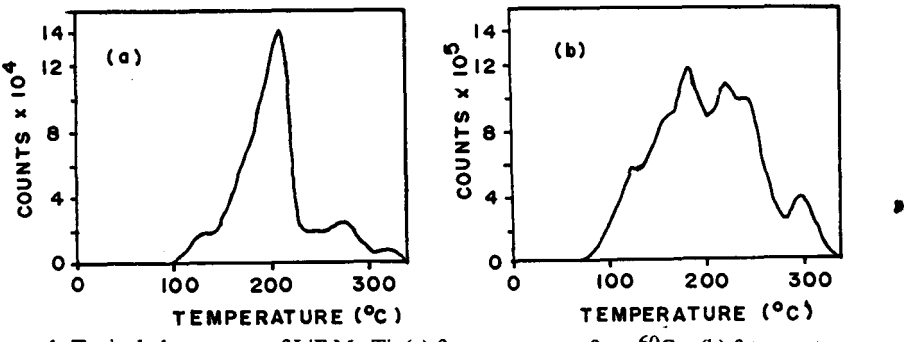

Figure 1. Typical glow curves of LiF:Mg,Ti: (a) for gamma-rays from ^{60}Co; (b) for neutron+ gamma-ray mixed fields from research reactor.

its low temperature side. From the Fig. 1b, it is found that the glow curve of laboratory made LiF:Mg,Ti TLD obtained from neutron+gamma-ray mixed radiations is quite different from those obtained from a gamma source (^{60}Co). In case of thermal neutron irradiation, a peak at 250 °C is produced more efficiently as compared to the 200 °C peak produced by the gamma-rays as well as the neutrons. Hence by following the difference between the two peak heights one can estimate the thermal neutron exposure from laboratory made single LiF:Mg, Ti dosimeter. Table 1 shows all the X and gamma-ray sources, the corresponding energies and the TLD sensitivities for each point of energy. The sensitivities were calculated by dividing their responses by the exact equivalent-dose delivered by each source. The relative gamma-ray sensitivity of LiF:Mg, Ti

compared to TLD-100 was found to be 0.97 which means that the dosimetric properties of laboratory made LiF:Mg,Ti TLD are comparable with those of TLD-100.

Table 2 shows all the neutron sources, the corresponding energies and the TLD sensitivities for each point of neutron energy. From Table 2, it is evident that the neutron sensitivities are highly dependent on neutron energies. The fast neutron sensitivity is less than that of thermal neutron sensitivity. The thermal neutron responses were reported interms of ^{60}Co equivalent roentgen (R) per 10^{10} n.cm^{-2} . The thermal neutron sensitivity varies from 325 to 389 R per 10^{10} n.cm^{-2} with an average value of 368 R per 10^{10} n.cm^{-2}. This value agrees well with the values quoted in the literature for TLD-100 (2). The accuracy in neutron and gamma-ray responses in the present study was within $\pm$10%.

Table 1. TLD sensitivities at various photon energies.

Source	Energy (keV)	Sensitivity (nC/mSv)
X-ray	33	8.45
X-ray	48	6.35
X-ray	65	5.98
X-ray	100	5.12
X-ray	118	4.78
^{137}Cs	662	4.21
^{60}Co	1250	4.56

Table 2.TLD sensitivities at various neutron beams.

Neutron energy (MeV)	Source	Neutron fluence (cm^{-2})	Sensitivity (nC/mSv)
2.5E-08	Reactor	1.45E07	78.56
2.1	Cf-252	1.78E05	2.56
4.5	Am-Be	1.25E05	1.98
14.0	Neutron generator	2.35E05	1.24

CONCLUSIONS

From these results it can be concluded that the laboratory made LiF:Mg,Ti single crystal is suitable for practical application in thermal neutron-gamma mixed field dosimetry around the nuclear facilities. Further, this dosimeter can be used in fasst neutron dosimetry purposes if the dosimeters are embeded with fast neutron moderating materials.

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PAPER TITLE Comparative study of different LiF TLDs for neutron dosimetry

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5. (see page 7)

ABSTRACT (See instructions overleaf)

Intensive investigations of new materials for thermoluminescent dosimetry (especially of ^{6}LiF/^{7}LiF-Mg,Cu,P) are now performing. Therefore the comparison of dosimetric characteristics for different TLDs on the basis of LiF has a great interest. The comparison of russian TLDs: TLD-6011/7011 (Moscow State University-"Practica",Moscow), DTG--4-6/7 (Institute of Geochemistry, Irkutsk) with TLD-600/700("Harshaw") is presented. The comparison has been performed on the basis of neutron dose equivalent measurements of these TLDs in IHEP neutron reference fields. The absolute sensitivity and the ratio of neutron and gamma sensitivities were estimated from the experimental data. The neutron dose equivalent has obtained by a passive monitor which consist of the LiF pairs in 25,4 cm diameter spherical polyethylene moderator. The comparison shows that photon sensitivity of new TLD--6011 is higher than one of TLD-600 by factor 50. The neutron sensitivity of TLD-6011 is higher than one for TLD-600 in 9 times and than for DTG-4-6 by a factor of 4. The neutron-to-photon responce ratio for the TLD-600 is about two times higher than that for TLD-6011 and DTG-4-6. The analysis of TLD perspectivity for area, environmental and individual monitoring given taking into account the new ICRP recommendation is presented.

EVALUATION OF THE FAST NEUTRON DOSE EQUIVALENT USING THE THERMAL NEUTRON RESPONSE OF LITHIUM FLUORIDE TLD

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INTRODUCTION

For estimation of the fast neutron dose equivalent we used the albedo method which consists in the detection of the albedo thermal neutrons. The albedo neutrons are the fast neutrons which leave the body as thermal neutrons after having been scattered in the human body. The albedo neutrons are detected at the surface of the body by means of the detectors for thermal neutrons.

For the measurement of the albedo thermal neutrons we used a Multifunctional Personal Dosimeter which contains 6 detectors of lithium fluoride with natural lithium. This has realized at Institute for Physics and Nuclear Engineering., in Bucharest.

The detectors of lithium fluoride with natural lithium are sensible at the photon radiations, at the beta radiations and at the thermal neutrons. The discrimination is realized with the help of the filters.

This dosimeter was designed so that with a suitable choice of Cd (0,8 mm), Sn (0.8 mm) and Al (1 mm) filters size and thichness he is suitable to measure the thermal neutron dose equivalent, the photon radiation dose equivalent and the beta radiation dose equivalent.

THE CONDITIONS OF IRRADIATION

The irradiation have been effectuated in the centre of a room with the next dimensions: (12 x 20 x 10) m on a support at 3 m height over of the floor.

In this way have been provided the ideal conditions of irradiation.

For to simulate the real conditions of utilization in the personal dosimetry, the dosimeters have been situated on the phantom, with the dimensions: (200 x 300 x 400) mm. The phantom has been filled with the water.

We were started with measurement at the Am-Be source with and without the phantom for to investigate the influence of scattering from the wall and the floor. We have found the thermal incident neutrons representing it a negligible contribution.

RESULTS

The experiments were performed with the neutrons from an Am-Be source and from an ^{252}Cf source.

For each fast neutrons source was determined the calibration factor, with which has been passed from the thermal neutron dose equivalent into the fast neutron dose equivalent.

The distance from the source at the phantom has been 30 cm both for the Am-Be source and for the ^{252}Cf source.

The irradiations of the Am-Be source were performed at the next values for the fast neutron dose equivalent: 0.385 Sv; 0.852 Sv and 1.331 Sv.

The irradiations of the ^{252}Cf source were performed at the next values for the fast neutron dose equivalent: 0.557 Sv and 0.751 Sv.

In the table no. 1 are presented the obtained results from the Am-Be source and in the table no. 2 are presented the obtained results from the ^{252}Cf source.

TABLE No. 1 - Am-Be Source

	$H_{nr, true}$ [Sv]	$H_{nr, ev}$ [Sv]	ε [%]	n
1.	0.385	0.385 ± 0.123	0	7
2.	0.852	0.982 ± 0.396	15.26	8
3.	1.331	1.171 ± 0.391	12,02	11

TABLE No. 2 - ^{252}Cf Source

	$H_{nr, true}$ [Sv]	$H_{nr, ev}$ [Sv}	ε [%]	n
1.	0.557	0.564 ± 0.24	1.25	12
2.	0.751	0.728 ± 0.227	3.06	10

In these tables the significance of the sizes is next:

- $H_{nr,\ true}$ is the true fast neutron dose equivalent at which has been the irradiation;
- $H_{nr,\ ev}$ is the mean value of the evaluated fast neutron dose equivalent;
- ε - the measuring error;
- n - number of the dosimeters which have been irradiated in the identical conditions and for which has been obtained the mean value.

The confidence level has been of 95 %.

The effectuated experiments have been affected in principal of the next sources of the errors:

- the little distance between source and phantom;
- the distribution of the dosimeters on the phantom;
- the long time for irradiation (the largest time has been > 1 month);
- the detectors of lithium fluoride have not been specially selected for this experiment.

From the experimental data presented results that the dose equivalent of fast neutrons can be estimated on the basis of the albedo method using our personal dosimeter with satisfactory results, only with an adequate calibration for each case.

We hope that the results can be improved.

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PAPER TITLE Dose Measurements in Mixed (n,gamma)-Radiation Fields with TLDs Under Consideration of the Peak Height Ratio

AUTHOR(S) NAME(S) M. Noll, N. Vana, W. Schöner, M. Fugger, F. Cecil

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2 (see page 7)

For dose measurements in unknown mixed (n,γ)-fields the neutron energy dependence of the TL-response was measured using TLDs embeded in spheres of 25 cm diameter (ICRU sphere). For this purpose the pair-method using ^{6}Li (TLD-600) and ^{7}Li (TLD-700) was approached. For optimizing the neutron shielding of TLD-700 and TLD-200 a special arrangement of the TLDs inside the spheres was considered.

The irradiation was performed with several neutron sources (thermal, Pu-Be, Am-Be and 14 MeV neutrons) with known equivalent doserates. Using the peak 5 maximum of TLD-600 glowcurves a neutron energy dependent calibration factor was calculated. It is further shown that the analysis of the peak height ratio of the neutron glow-curves of TLD-600 gives information about neutron energy dependent effects in material with different LET. The results were applied for measurements in aircrafts.

For determination of the equivalent dose the neutron spectrum is of great importance. To estimate the neutron spectrum a neutron spectrometer consisting of Bonner spheres with different diameters is used. This spectrometer was tested with known spectra of several neutron sources. The theoretical response of the spheres were calculated using the neutron spectrum and the energy dependent response and were compared with the measured values. The method was applied for determination of the equivalent dose in aircrafts.

MEASUREMENT OF TLD ALBEDO RESPONSE ON VARIOUS CALIBRATION PHANTOMS

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INTRODUCTION

The International Commission on Radiation Units and Measurements (ICRU) has recommended that individual dosemeter should be calibrated on a suitable phantom and has pointed out that the calibration factor of a neutron dosemeter is strongly influenced by the the exact size and shape of the body and the phantom to which the dosemeter is attached. As the principle of an albedo type thermoluminescent personal dosemeter (albedo TLD) is essentially based on a detection of scattered and moderated neutron from a human body, the sensitivity of albedo TLD is strongly influenced by the incident neutron energy and the calibration phantom.(1) Therefore for albedo type thermoluminescent personal dosemeter (albedo TLD), the information of neutron albedo response on the calibration phantom is important for appropriate dose estimation. In order to investigate the effect of phantom type on the reading of the albedo TLD, measurement of the TLD energy response and angular response on some typical calibration phantoms was performed using Dynamitron accelerator and ^{252}Cf neutron source.

DOSEMETER

Table 1 shows the constitution of PNC/Panasonic TLD badge used in this experiment. This TLD badge was designed to detect thermal neutron and fast neutron by using 4 TLD elements and cadmium and tin filters.(2) Element 1 has the sensitivity only to photon and used for photon compensation. Element 2 has major sensitivity to thermal neutron from working field. Element 3 has sensitivity only to epithermal neutron. Element 4 has major sensitivity to albedo neutron which was refracted from human body. The calculation of fast neutron dose equivalent is performed by following equation.(3-4)

$$H = Kn(T4-T1-C(T2-T3))$$

where H is the dose equivalent of fast neutron, Kn is the conversion factor, C is the correction factor for thermal neutron compensation, T1,T2,T3 and T4 are measured values of each TLD element.

Since, the value of Kn is based on T4 sensitivity, it is influenced by phantom characteristics such as material and shape.

EXPERIMENTAL METHODS

The fast neutron experiment was performed by using the Dynamitron accelerator at Tohoku university which produces monoenergetic neutrons of around 210 keV and 560 keV by the Li(p,n) reaction, around 1.1 MeV by the T(p,n) reaction, 5.0 MeV by the D(d,n) reaction, and 14.9 MeV by the T(d,n) reaction. The neutron energy was determined by using time of fright technique. The TLD on the phantom was placed in the forward direction to the axis of the projectile beam. The direct neutron flux incident on the TLD was measured with the ^{235}U fission chamber placed on in front of the TLD. Since the ^{235}U fission chamber is sensitive to low energy neutrons scattered from the surrounding objects, a hydrogen proportional counter which was sensitive only to fast neutrons was also placed at 45 degree to the beam axis as subsidiary monitor of the neutron flux. The experiment on angular response was performed by using the ^{252}Cf source at the PNC calibration facility.

The calibration phantoms used in this experiment were as follows. (a) paraffin phantom, (b) polymethyl methacrylate(PMMA) slab phantom (specified by Japanese Industrial Standard for calibration phantom of gamma ray personal dosemeter; JIS phantom), (c) cylindrical water phantoms. (we used two types, one was full water phantom and the other was cavity water phantom). These were chosen by the point of view of typicality and availability. (d) special anthropomorphic phantom which was made of neutron tissue equivalent material(5-6) and of which the physique was same as normal Japanese man of age from 30 to 35.(7) We considered that the anthropomorphic phantom was equivalent

to living human body for neutron scatter. Table 2 shows phantom specification which were used in this experiment.

RESULTS AND DISCUSSION

Figure 1 (a) and (b) show the TLD energy response on various calibration phantoms. The element 4(T4) had higher sensitivity to fast neutron than a element 2 (T2) and a element 3 (T3) because of the absorption of albedo neutron by Cd filters beside T2 and T3. Sensitivitys of T2 and T3 are same to fast neutron. On the paraffin phantom, the sensitivity of element 4 which has high sensitivity to albedo neutron is same as the anthropomorphic phantom below about 1MeV, but in the energy range over 1MeV, the sensitivity on the paraffin phantom becomes about 1.5 times higher than anthropomorphic phantom. On JIS phantom, TLD response was higher than anthropomorphic phantom at the all irradiated energy region.

Figure 1 (b) shows comparison of anthropomorphic phantom and water phantoms. The response on the water phantom is in good agreement with the response on the anthropomorphic phantom, but cavity water phantom become lower than anthropomorphic phantom by a factor of 0.8. These experimental results suggest that water phantom is suitable for calibration phantom because of a good agreement with the anthropomorphic phantom which is considered as equivalent to human body.

The reason of the difference of energy response between the paraffin phantom and the anthropomorphic phantom seems to be the effect of the lung in the anthropomorphic phantom. The anthropomorphic phantom which have imitated lung with low density material has different scattering and moderating effects from the paraffin phantom which has uniform density.

The reason of the difference between the acrylate phantom and the anthropomorphic phantom seems to be the difference of the mean density. The density of 1.19 g/cm^3 of acrylate phantom seems to be too high as tissue equivalent.

The observed difference of energy response on cavity water phantom and anthropomorphic phantom seems to be because of the difference of lung volume. The cavity water phantom have large cavity of about 8 liters volume for lung which is two times as large as that of anthropomorphic phantom. Therefore incident fast neutron cannot be scattered and moderated well because of large volume of cavity in former phantom.

Figure 2 shows the angular response on various phantoms by using ^{252}Cf source. The distance between source and phantom surface was set at 75 cm and incident angle were changed from 0 degree to 180 degree. As the result of angular response of TLD element 4, the cavity water phantom is in good agreement with the anthropomorphic phantom. The water phantom has the good agreement at the front side incidence (from 0 to 90 degree) with the anthropomorphic phantom, but at the back side incidence(from 120 to 180 degree) it had a different response and the maximum difference ratio was about +400% at the 180 degree.

The acrylate phantom has also different response from the anthropomorphic phantom at almost all conditions and the maximum difference ratio was about -60%.

CONCLUSION

The energy response of the TLD on the 46x30x20cm elliptical cylinder phantom filled with water was in good agreement with the response on the anthropomorphic phantom. The TLD response on the 40x40x15cm PMMA slab phantom became higher than the anthropomorphic phantom at the all irradiated energy region. The TLD response on the 46x30x20cm elliptical cylinder paraffin phantom was in good agreement with the anthropomorphic phantom below about 1MeV, but in the range over 1MeV, the TLD response on the paraffin phantom became about 1.5 times higher than the anthropomorphic phantom. The angular response of TLD on the cavity water phantom considered as lung geometry was in good agreement with the anthropomorphic phantom. The phantom filled with water had different response from the anthropomorphic phantom with the maximum difference ratio of +400% at the back side incidence. The PMMA phantom had also different response from the anthropomorphic phantom at all incident angles and the maximum difference ratio was about -60%.

These experimental results suggest that water phantom (filled/cavity) is suitable for calibration of albedo TLD because of the good agreement with the energy/angular response of anthropomorphic phantom considered as equivalent to human body.

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Table 1 Constitution of the TLD badge

Type	Element No.	TLD and Filter
PNC/ Panasonic UD-809P	T1	(Front) Cd/$^{7}Li_2{}^{11}B_4O_7$/Cd (Rear)
	T2	Sn/$^{6}Li_2{}^{10}B_4O_7$/Cd
	T3	Cd/$^{6}Li_2{}^{10}B_4O_7$/Cd
	T4	Cd/$^{6}Li_2{}^{10}B_4O_7$/Sn

Table 2 Specification of the phantoms

Type of Phantom	General Usage	Shape, Size	Material	Remark
Paraffin Phantom	Calibration of Personal Dosimeters	Capsular Cylinder 40X30X20cm	Paraffin 0.9g/cm³	
Acrylate Slab	Calibration of Personal X,γ Dosimeters	Slab 40X40X15cm	Methyl Methacrylate Resin (JIS K6718) 1.19g/cm3	JIS Z4331
Water Phantom	Test of X ray tube for Diagnosis	Elliptical Cylinder 46X30X20cm	Water	JIS Z4915
CavityWater Phantom	Test of X ray tube for Diagnosis	Elliptical Cylinder 46X30X20cm	Water	JIS Z4915
Anthropomorphic Phantom	Test of Personal Neutron Dosimeters	Average Japanese Man	Tissue:ICRU solution Lung:Kyoto Kagaku LP-430 Bone:Kyoto Kagaku BE-204	

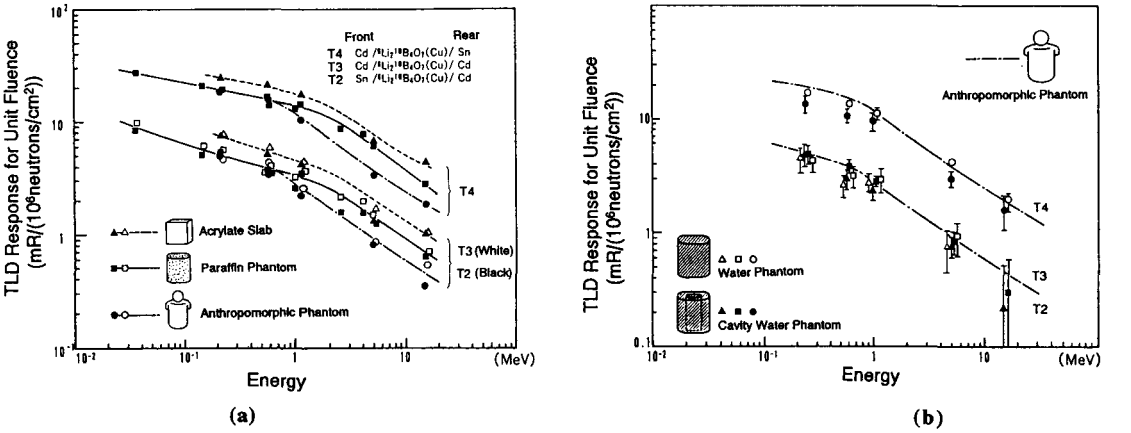

Figure 1 TLD energy response on the phantoms

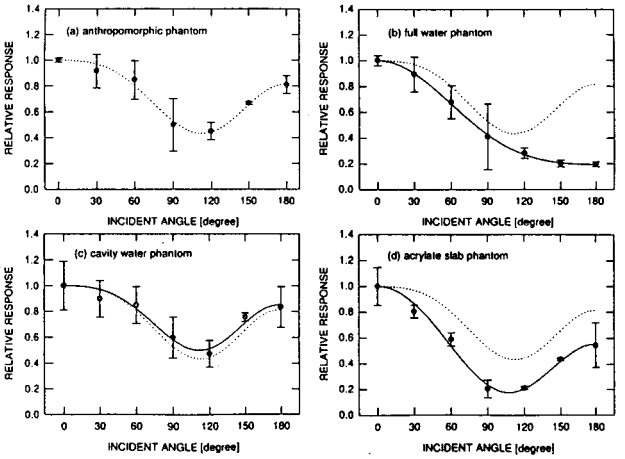

Figure 2 TLD angular response on the phantoms

FACTORS AFFECTING POLYAMIDE PROTOTYPES DESIGN OF ALBEDO DOSIMETERS

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ABSTRACT

This work studies the most important factors which affect the response of albedo neutron dosimeters containing LiF TLDs with the aim to improve their sensitivity. It includes tests of thickness and shape of the polyamide moderator body prototypes, albedo window diameter and TLD position inside the moderator. Analyzing the results, an albedo neutron dosimeter prototype, B_4C covered, was developed. The prototype has a response three times higher than the albedo dosimeter now in use in Brazil.

INTRODUCTION

Albedo dosimeters have been extensively used in neutron personnel monitoring with good results (1). Since 1983, Instituto de Radioproteção e Dosimetria (IRD) operates the only Neutron Personnel Monitoring Service in Brazil. This service monitors about two hundred workers in nuclear and non-nuclear facilities. An albedo dosimeter developed in 1975 is used (2). For ^{241}Am-Be sources, the IRD albedo neutron dosimeter is about 50 times less sensitive than for neutron fields around reactors (3).

The restrictions on individual dose limits recommended in ICRP60 (4), together with the lack of a neutron dosimeter with a low detection limit, mainly for fast neutron fields, have stimulated the research to obtain a more sensitive personal neutron dosimeter. Also, the IRD needs a cheaper and easier to manufacture neutron dosimeter than the actual. The objective of this work is to identify the most important factors which affect the response of albedo dosimeters containing TLDs and improve their sensitivity, taking into account cost-benefit considerations.

MATERIAL AND METHODS

All prototypes were made with a moderator body of polyamide. And, for the detection system ^{6}LiF and ^{7}LiF TLD chips (3.2x3.2x0.89 mm^3 TLD600 and TLD700 from Harshaw Chemical Company, respectively) were used. Three prototype sets with a cavity in the geometric axis for the TLD pairs were manufactured:

a) semispherical shape with same height (24.2 mm) and albedo window diameters from 32 to 52 mm;
b) cylindrical shape with same albedo window (32 mm) and heights from 18 to 36 mm;
c) cylindrical shape with same albedo window (50 mm) and heights from 5 to 20 mm.

To understand the influence of some geometrical factors in the albedo dosimeter response, some tests were carried out, changing the moderator body of the prototypes:

a) the influence of shape: cylindrical or semispherical;
b) the influence of the LiF TLD pair position;
c) the influence of the moderator dosimeter thickness, for a same shape;
d) the influence of albedo window diameter.

The assays took place in a low scattering room with the prototypes on a cubic water phantom of polymetylmethcrylate. In the experimental set-up, the total phantom area for

irradiation of prototypes was 0.4x0.4 m^2. Irradiations were made with a ^{241}Am-Be bare source, at 1 m distance, with doses around 8 mSv. The response of TLD-600 and TLD-700 were evaluated in a Harshaw model 3500 TLD reader, using a nitrogen flow of 0.13 $m^3.h^{-1}$. Annealing procedures for TLD are those recommended by Harshaw: 1 h at 400°C plus 2 h at 100°C. After irradiation, the TLDs are heated during 15 minutes at 100°C before their evaluation. The reading cycle consists of no pre-heat and a linear heating of 10°C.s^{-1} from 70°C up to 350°C. The total heating time was 30 s. For neutron dose evaluations the difference between TLD600 and TLD700 integration value of peak 5 was used.

RESULTS

Concerning the results, it can be summarized that:

- The response increases in proportion with albedo window area, however it tends to saturate.
- Independently of the geometric shape, but for a same albedo window, the response has a linear decrease when the TLD position is changed from surface (phantom) to inside the moderator body. Figure 1 shows this effect for different shapes and thickness of the moderator. These results are in agreement with the ones obtained by Hankins (5) using polyethylene moderator and cadmium absorber.
- For a same geometric shape (cylinder), the response increases for higher height (however it tends to saturate), as it is shown in figure 2.
- The angular dependence difference is statistically non significant for semispherical or cylindrical shape.

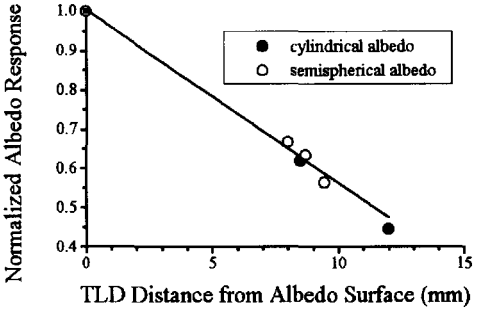

Figure 1. Variation of albedo response as a function of TLD distance from surface.

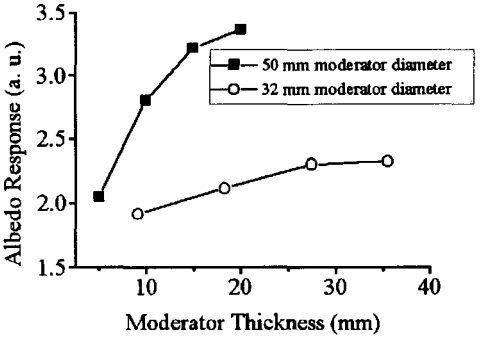

Figure 2. Albedo response as a function of polyamide moderator thickness.

CONCLUSION

Analyzing the results, a cylindrical albedo neutron prototype with a polyamide moderator body (46 mm diameter and 15 mm height), covered with 2 mm of boron-resin, was developed using two LiF TLD pairs: one for albedo and another for incident neutrons (figure 3). This prototype has a response about three times higher than the albedo dosimeter now in use in Brazil. For ^{241}Am-Be bare source, the new low detection limit is around 0.15 mSv, lower than the Register Level (0.2 mSv) adopted in Brazil. This prototype is now in test on other neutron fields.

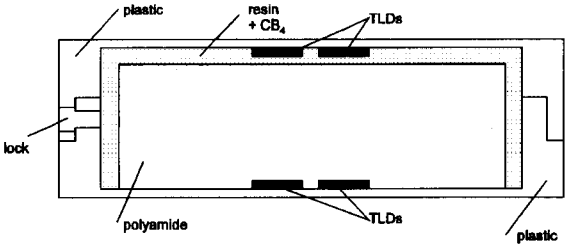

Figure 3. Albedo dosimeter prototype.

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THERMOLUMINESCENT DOSIMETERS (TLD) EXPOSED TO HIGH FLUXES OF GAMMA RADIATION, THERMAL NEUTRONS AND PROTONS

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INTRODUCTION

Thermoluminescent dosimeters (TLD), widely experimented and utilised in personal dosimetry, have some advantageous characteristics which induce one to employ them also in radiotherapy. The new radiotherapy techniques are aimed at selectively depositing a high dose in cancerous tissues. This goal is reached by utilising both conventional and other more recently proposed radiation, such as thermal neutrons and heavy charged particles. In these inhomogeneous radiation fields a reliable mapping of the spatial distribution of absorbed dose is desirable, and the utilised dosimeters have to give such a possibility without notably perturbing the radiation field with the materials of the dosimeters themselves. TLDs, for their small dimension and their tissue equivalence for most radiation, give good support in the mapping of radiation fields.

After exposure to the high fluxes of therapeutic beams, some commercial TL dosimeters have shown a loss of reliability. An investigation has therefore be performed, both on commercial and on laboratory made phosphors, in order to investigate their behaviour in such radiation fields. In particular the thermal neutron and gamma ray mixed field of the thermal column of a nuclear reactor, of interest for Boron Neutron Capture Therapy (B.N.C.T.) and a proton beam, of interest for Proton Therapy, were considered. Here some results obtained with new TL phosphors exposed in such radiation fields are presented, after a short description of some radiation damage effect on commercial LiF TLDs exposed in the (n_{th},γ) field of the thermal column of a reactor.

COMMERCIAL LiF DOSIMETERS IN HIGH FIUXES OF THERMAL NEUTRONS

LiF dosimeters are typically utilised for absorbed dose determination in thermal neutron fields. In fact, as well known, the high cross section (945 barns) for the capture reaction $^6Li(n,\alpha)^3H$ and the small cross section (0.04 barns) for the reaction $^7Li(n,\gamma)^8Li$ gave rise to the standardisation of the employment of a couple of ^6LiF-^7LiF dosimeters for dose determinations from thermal neutrons and γ-rays in personal dosimetry, where the low thermal neutron component makes the response of ^6LiF TLDs reliable and makes the response of ^7LiF TLDs attributable, with very good approximation, to the γ-component alone. In contrast, some difficulties occur when LiF dosimeters, not ^6Li depleted, are exposed to high fluxes of thermal neutrons.

The dosimeters investigated are:

i) LiF:Mg,Ti (chips 3.1× 3.1× 0.9 mm^3) from the Harshaw Chemical Corporation: TLD-600 (with 96.5% ^6Li), TLD-700 (with 99.99%. ^7Li F) and TLD-100 (with 7.5% ^6LiF).

ii) LiF:Mg,Cu,P (circular chips 4.5 mm diameter, 0.8 mm thick) from the Beijing Radiation Detector Work, People's Republic of China: GR-206A (^6LiF:Mg,Cu,P) and GR-207A (^7LiF:Mg,Cu,P).

For all dosimeters the recommended annealing procedures were utilized, i.e.:

LiF:Mg,Ti	400°C for 1 h, 100°C for 2 h
LiF:Mg,Cu,P	240°C for 10 min, followed by rapid cooling down to RT.

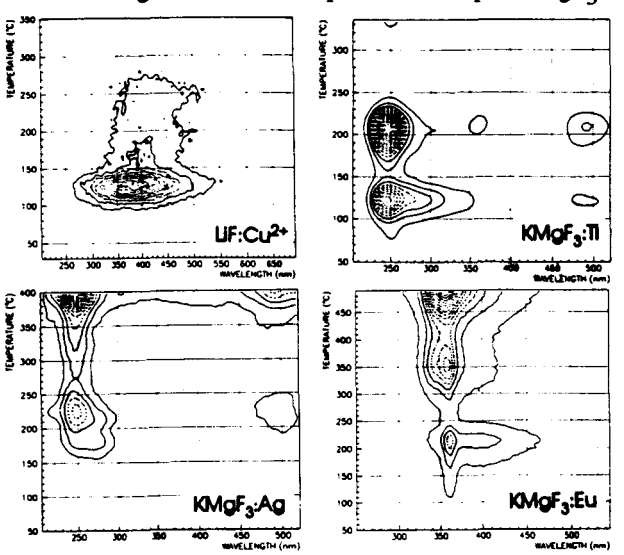
Dosimeters were exposed in the swimming-pool-type facility or in the thermal column of a TRIGA MARK II nuclear reactor at various fluences, up to about 10^{12} n cm^{-2}. The response of TLDs that were never exposed to thermal neutrons before shows a small lack of linearity in the case

of TLD-600 and a dramatic lack of linearity in the case of GR-206A. Moreover, both dosimeters, after high fluences of thermal neutrons, show radiation damage that is not removed by annealing. After exposure to thermal neutrons, read-out and annealing, in fact, their response to γ-rays is lower, the more higher the previous thermal neutron fluence has been. Therefore, ^{6}LiF dosimeters exposed to high thermal neutron fluences do not give a reliable response and, moreover, they can no longer be utilised after such exposure. In contrast, ^{7}LiF dosimeters do not show any sensible radiation damage in the fluences considered, and may therefore be utilised in such radiation fields. In the (n_{th},γ) mixed field of a reactor thermal column, however, the response of ^{7}LiF dosimeters is due to both components of the field, and a method for separating the two contributions has to be properly set up.

TL SINGLE CRYSTALS

With the purpose of identifying phosphors whose emission characteristics allow reliable dose determinations in therapeutic radiation fields, and for which radiation damage do not hinder successive utilization, we have started analyses concerning single crystals, because they have the peculiarity of allowing high annealing temperatures. Single crystals of LiF, doped with Mg,Cu,P or only with Cu, and fluoroperovskite crystals $KMgF_3$, doped with Tl or with Eu or with Ag, were grown using the Czochralski technique. For all crystals the chosen annealing procedure was heating at 500°C for 30 min, followed by rapid cooling down to room temperature. The samples were variously irradiated in n_{th}, proton and γ-ray fields. Thermal neutron exposures were made at the TRIGA MARK II reactor, 13.6 MeV proton irradiations at a CYPRIS cyclotron. ^{137}Cs γ-rays (at a rate of 0.14 Gy/s) for calibrations and ^{60}Co γ-rays (at a rate of 0.23 Gy/s) for high dose studies were utilised. The glow curves were obtained from a Harshaw 3500 TLD Reader, at a constant heating rate of 4°C/s, up to 400°C.

In order to better characterise the materials, some dosimeters from each group were also analysed by means of a laboratory-made spectrometer for wavelength resolved TL measurements. LiF:Mg,Cu,P crystals display an emission at 410 nm. LiF:Cu crystals have undergone, during preparation, an OH^- contamination from aqueous vapour, and consequently they show a relevant emission, centred at 400°C, which masks the characteristic emission of Cu^{++}, at 425°C. A contour plot of the crystal is shown in Fig.1. Fluoroperovskite crystals ($KMgF_3$) show different emission characteristics. Tl and Ag doped crystals have their main emission in the spectral range 240-260 nm, that is outside the wavelength range (350-700 nm) of the commonly used commercial readers. $KMgF_3$:Eu samples show a high peak centred at 360 nm, and so a large part of the emission is detected by commercial readers. In Fig. 1 the isometric plots of all doped $KMgF_3$: crystals after γ-irradiation are shown.



In LiF:Mg,Cu,P the glow curve after γ-irradiation displays two pronounced peaks at 230°C and 350°C and two shoulders on both sides of the 230°C peak and above 350°C. Thermal neutron irradiation mostly increases the peaks at 260°C and 280°C and seems to quench the 350°C peak in favour of two peaks at 330°C and 370°C. This anomalous behaviour is observed to be persistent after a subsequent γ irradiation. The response to γ and neutrons becomes lower with use. The absolute value of the TL signal is much lower (by at least three orders of magnitude) than that obtained from commercial chips.

The glow curve of LiF:Cu^{2+} displays a dominant peak at 150°C and much weaker secondary peaks at 220°C, 270°C and 330°C, whose relative height is doping dependent. The area of the main peak is strongly dependent on the irradiation dose. These samples display a good linearity for γ irradiation in the range 1-5 Gy. The glow curve after thermal neutron exposure displays a strong population of the 270°C and 330°C peaks, proportional to the fluence. Limited damage (about 10%) after strong neutron irradiation was observed. Some samples were irradiated with γ-rays at a dose of 10^3 Gy. No relevant difference in the TL emission was discovered, and only a light sensitisation after this high dose was observed.

Fluoroperovskite crystals $KMgF_3$ have been investigated because these phosphors, recently proposed as dosimetric materials, have shown some interesting features (for example, Tl doped crystals have revealed a linear response even up to doses of 10^3 Gy). The responses after a dose of 1 Gy in tissue from γ-rays or from n_{th} are of the same order. With reference to the possibility of discriminating between contributions in such mixed fields, no different emission characteristics after n_{th} or γ-irradiation were discovered.

The glow curve of $KMgF_3$ samples exposed in the proton beam, shows a peak population different from that after γ-irradiation, not only for a different population of the peaks that are characteristic of γ-irradiation, but also for the evidence of new trap creation. These traps are removed by annealing; this fact suggests that protons induce, by knock-on, a lattice configuration unstable at high temperature.

In the Table the results obtained are shown schematically.

phosphor	λ_{emiss} (nm)	TL peaks (°C)	R_γ / R_γ(TLD-700) * 10^{-2}	R_n / R_n(TLD-700) * 10^{-2}	R_n / R_n(TLD-600) * 10^{-3}
LiF:Mg,Ti (TLD700)	400	210	100.	100.	2.28
LiF:Mg,Cu,P	410	230,350 140,260	0.18	163	3.71
LiF:Cu (OH-contaminated)	425	150 220,270,330	1.83	240	5.49
$KMgF_3$:Ag	245	190,220,280	5.75	3.26	0.074
$KMgF_3$:Eu	360	250,340 150	319.	627.	14.3
$KMgF_3$:Tl	245-370	140,230 190,320	4.38	5.29	0.121

The above data may give only a summary indication of the behaviour of the dosimeters. The reported values are obtained as averages of the measured ones, but the dispersion of experimental values is very high. In fact the trap configuration in a crystal, and consequently the glow curve shape, are strongly dependent upon doping concentration, and this concentration is not uniform in the laboratory-grown crystals out of which our samples were cut. From these results however a qualitative understanding of the behaviour of phosphor is possible, and a direction for subsequent experiments is deduced.

ACKNOWLEDGEMENTS

The authors are grateful to Mr.-G. Di Corato (GAMMATOM - Guanzate, Co, Italy) for his useful cooperation in irradiating samples with the ^{60}Co source.

COMPARATIVE STUDY OF DIFFERENT LiF TLDs FOR NEUTRON DOSIMETRY

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The passive neutron dosimetry based on the thermoluminescent detectors (TLD) is a promising trend in the area, environmental and personnel radiation monitoring of accelerators. However, a successful TLDs application for the neutron dosimetry is limited by their low neutron response. The necessity of increased TLD response is associated with the new national standard in radiation protection, based on the recommendations of the International Commission on Radiological Protection (ICRP, Recommendation ICRP-60). In recent years some new TL materials (such as LiF:Mg,Cu,P) having higher sensitivity as compared to the well-known LiF:Mg,Ti have been developed. The characteristics of the LiF:Mg,Cu,P and LiF:Mg,Ti TL detectors from the point of view of neutron dosimetry was compared[1]. The absolute sensitivity and the ratio of neutron and gamma sensitivities have been estimated from the experimental data for different TL-detectors. The following TL-detectors are compared:

DTG-4-6 (^{6}LiF:Mg,Ti),

DTG-4-7 (^{7}LiF:Mg,Ti), (Institute of Geochemistry, Irkutsk, Russia);

TLD-6011 (^{6}LiF:Mg,Cu,P),

TLD-7011 (^{7}LiF:Mg,Cu,P), (Moscow State University -"Practica",Moscow);

TLD-600 (^{6}LiF:Mg,Ti),

TLD-700(^{7}LiF:Mg,Ti), ("HARSHAW", USA);

The annealing procedure for all detectors was carried out in accordance with the recommendations of the TLD manufactures. The HARSHAW 2080 Reader was used for the readout. The dose measurements are performed according to the

glow-curve integration. The integration has been performed above 100 ^{0} C distiminator level for the glow curve.

The comparison of TLD characteristics has been performed on the basis of neutron dose equivalent measurements with the IHEP passive radiation monitors (PMS). The PMS used for area and environmental radiation monitoring at the IHEP accelerator and consists of the LiF pairs located in the 25.4 cm in diameter spherical polyethylene moderator.

The PMS with TLD was irradiated in the following fields: The standard calibration field with ^{137}Cs proton source in a lead collimator. The reference field based on the ^{239}Pu-Be neutron source.This field is used for PMS calibration in units of the neutron dose equivalent. The reference field based on the ^{252}Cf neutron source. The reference field based on the ^{252}Cf source in the 30 cm-diameter spherical iron moderator (^{252}Cf+Fe). The reference field based on the ^{252}Cf source in the 30 cm diameter spherical polyethylene moderator (^{252}Cf+CH_2).

Table 1. presents the measurement results of the ratio of detector response .The data for DTG-4-7 and TLD-7011 to the TLD-700 values and ones for DTG-4-6 and TLD-6011 are normalized to the TLD-600 values. The of neutron-to-photon response rations are given in Table also.

Table 1.

Detectors	Ratio of photon response detectors , rel. unit .	Ratio of neutron response detectors, rel.units	Neutron-to-photon response ration, rel. units.
DTG-4 -6(7)	4.2±0.3	1.8 ±0.2	0.4±0.05
TLD-7011(6011)	18±1	4.3±0.2	0.6±0.06
TLD-700(600)	1.0	1.0	1.0

The neutron-to-photon response ratio for DTG-4-6 (DTG-4-7) and TLD-6011 (TLD-7011) are the same, but this ratio for TLD-600 (TLD-700) is higher by a factor of two as compared to other detectors. LiF:Mg,Cu,P have lower efficiency to high LET radiation. And low neutron-to-photon response ratio for LiF:Mg,Cu,P is an ionization density effect [2] . The decrease of neutron-to-photon response ratio leads to the increase of the measurement errors of the neutron dose equivalent in fields under large photon contribution into a total dose .

Usually the contribution of photons, muons and charged particles into

the total dose equivalent is less than 10% behind the shield of high energy accelerators. The exception is the so-called "muons torch" at the IHEP experimental hall, when the neutron dose equivalent contribution is less than 5%. Therefore a decrease in ratio of neutron-to-neutron does not restrict applications for the neutron dosimetry at accelerators.

The minimum measurable dose depending on the TLD type is the most

important characteristic for dosimetric methods. The minimum measurable dose for photons was obtained as a "zero dose" of the TLD readings immediately after annealing. The minimum measurable neutron dose equivalent for different detector types and dosimetric method (HARSHAW 2080 reader) without nitrogen atmosphere are presented in Table 2.

Table 2.

Detector type	Minimum measurable neutron dose equivalent of PMS, mSv.	Minimum measurable neutron dose of albedo individual dosimeter, mSv.
DTG-4-7(6)	0.02	0.1
TLD-7011(6011)	0.008	0.04
TLD-700(600)	0.03	0.15

The lower limit of detection according to ICRP-35 Recommendations should be 0.1 of an annual limit. It should be borne in mind that comentionally administrative limits are less than annual limit. The annual level is equal to 2 mSv according to ICRP-60. Therefore in the individual monitoring a monthly measured dose at the lower limit of detection should be 0.17 mSv. It is TLD-6011, 7011 for the albedo dosimeter that provides the neutron equivalent dose measuring at the lower limit of detection. The PMS with TLD-6011, 7011 allows one to carry out the area monitoring with the minimum measurable dose 0.1 of the recommended dose limits for population (1 mSv a year). At the same time the new TL-detectors are intended to be used in area and individual monitoring even under the new ICRP recommendation in radiation protection.

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1996 International Congress on Radiation Protection
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PAPER TITLE Electrochemical etching of CR-39 detectors at low temperature

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5 (see page 7)

ABSTRACT (See instructions overleaf)

Different etching solutions, which are suitable to evaluate the particle tracks in CR-39 detectors at room temperature were systematically tested for electrochemical etching. The CR-39 detectors were irradiated with α particles and neutrons of different energy. The response of electrochemically etched detectors was measured. The background of detectors were determined.

A METHOD FOR CALCULATION OF THE NEUTRON RESPONSE FUNCTION OF AN ELECTROCHEMICALLY ETCHED CR-39 SSNTD UP TO 65 MEV

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Development of a personal dosimeter capable of detecting neutrons of energies above 20 MeV is an important problem of radiation protection at high energy accelerators. The electrochemicaly etched CR-39 SSNTD seems to be applicable for this purpose but the determination of the neutron response function of this detector has many difficulties.

Above 14 MeV, practically it is not possible to produce beam of really monoenergetic neutrons. Neutron spectra of the sources used for calibration have significant low energy tailors. Their contribution to the measured track densities are not negligible because of the low detection treshold of the electrochemicaly etched CR-39.

The evaluation of the neutron response function from measured track densities needs the employment of an unfolding method. Unfolding codes need not only the neutron spectra used for the calibration but a "good" input response function to avoid divergencies in the iteration.

We have developed a method for calculating an "input" response function for the unfolding. We have calculated the number of the latent tracks crossing an internal surface at depth "h" in CR-39 using the neutron spectrum and double differential cross sections of neutron induced nuclear reactions on hydrogen, carbon and oxygen nuclei. Track etch kinetics and the Smythe-model were applied for the calculation of the density of tracks contributing to the electrochemical etching response.

ANALYSIS OF HIGH DOSE NEUTRON IRRADIATED CR-39 USING COHERENT LIGHT SCATTERING

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ABSTRACT

In case of high neutron flux and high doses of irradiation, the Solid State Nuclear Track Detector (SSNTD) CR-39, used in neutron dosimetry, is unsuited for optical microscopy analysis after chemical etching. This is because track overlapping makes counting and exploitation impossible. We therefore developed a new method that allows the reading of irradiated samples based on the measurement of the angular distribution of coherent light (He-Ne laser) transmitted through the SSNTDs after chemical etching. The scattering of coherent light gives us a statistical and comprehensive view of track morphology (size and orientation of track openings).

INTRODUCTION

Although they are themselves not ionizing, neutrons are registered in CR-39 through recoil protons, produced either in the radiator or in the detector. These protons cause latent tracks in the detector. It will be recalled that only a fraction of these tracks is developed after chemical etching (NaOH 7N during 180 min.). Their characteristics are the track density (or even track volume density), track size, shape and orientation. This study aims to examine, with the use of the BTDF (Bi-directional Transmissive Distribute Function), the relationships between the neutron field parameters and those that characterize the microscopic aspect of the CR-39 surface after chemical etching. First we will present a study of SSNTDs irradiated under mono-energetic and mono-directional proton flux. Next, we will move on to neutron irradiations by neutron-proton radiators.

EXPERIMENTAL SET-UP

The light source is formed by a He-Ne laser beam (λ = 632.8 nm). A reference detector is used to enable the computer to ratio out laser power fluctuations. The sample mount is perpendicular to the laser beam. The receiver-rotation stage is motorized and computerized, enabling a full rotation of the input aperture on any point of the sample. Optical microscopy measurements are taken with an Olympus microscope connected to a computerized CCD camera.

PROTON IRRADIATIONS

The list below shows the characteristics of the samples studied. The choice of samples was guided by etching conditions and incident protons energy. Choosing too little energy would have led to over-etching tracks and thus to circular track openings regardless of the incidence angle of the protons. Moreover, the conditions of particle incidence were limited by the etching critical angle, which was itself dependent on the energy of incident particles **[1] [2]**.

Sample	Proton energy	incidence (°)	track density	average area (μm^2)	min. ferret (μm)	aspect ratio
P7	847 KeV	0	9.98×10^5	14 ± 4.44	4.05 ± 0.50	0.8472
P12	850 KeV	20	1.16×10^6	11.9 ± 4.2	3.57 ± 0.65	0.7819

SSNTDs are irradiated under proton flux (Van de Graaf accelerator) with to incidences of 0° and 20°. The minimum diameters of track openings decrease with the angle of irradiation and the aspect ratios (min. diameter/max. diameter) do likewise. Figures 1 and 2 are histograms showing the orientation of the track openings. The orientation of the plane of the track opening is defined as the angle between horizontal reference axis and the largest diameter of the opening track.

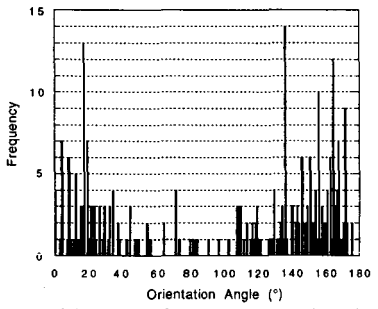

Fig. 1. Histogram of track opening orientations of ssntd P7.

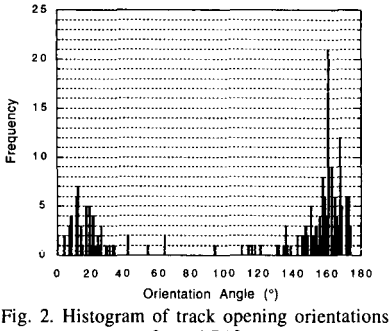

Fig. 2. Histogram of track opening orientations of ssntd P12.

In the case of SSNTD P7, the histogram seems to show two preferential directions in the neighborhood of 20° and 150° (Fig. 1). A close study shows that a quasi-circular track opening does not present a specific direction in the plane of analysis. The direction can thus vary from one track to another. In fact, a 90°-rotation of CCD receiver in its plane gives the same histogram. This stressed the optical defaults of the camera which can subsequently be rectified.
In the case of SSNTD P12 with a proton incidence of 20° (Fig. 2), the orientation histogram is much narrower than the former. This clearly shows a real orientation of the track openings (with a camera rotation of 90°, the shift of histogram is unmistakable).

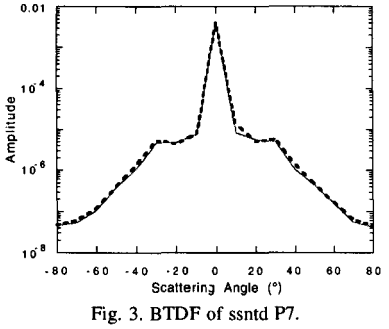

Fig. 3. BTDF of ssntd P7.

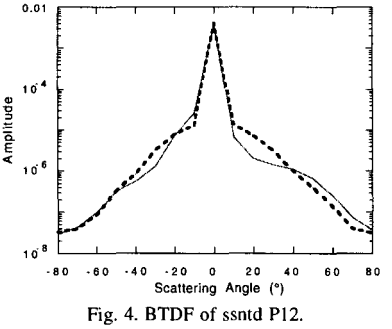

Fig. 4. BTDF of ssntd P12.

The scattering indicators, or BTDF, in the case of P7 sample were measured in the two perpendicular planes through the laser beam axis (Fig. 3) [3]. The two BTDF are symmetrical with the direct transmitted beam (the Optical Transmission, i.e. the scattering angle value 0°). They also have the same profile width. The BTDF characteristics clearly show the symmetry of track revolution with the perpendicular axis of the sample.

In the case of P12, the two BTDFs are not superposed. Their profile widths are different and their symmetry with the 0° scattering angle no longer applies (Fig. 4). These relationships can be corroborated by analyses made by optical microscopy (Fig. 2), which confirm that there is a specific track orientation that can be highlighted by laser scattering.
There is thus a reciprocal relationship between track openings and the shape of the BTDF.

NEUTRON IRRADIATION

1. BTDF as a function of the angle of irradiation.

The irradiated SSNTDs are surrounded by polyethylene (n-p conversion), with varying incidence angles of the incoming neutron flux (0, 30, 45, 60, 90°). Irradiations are carried out on the CEA's (Valduc) Silene reactor with doses ranging from 0.11 Gy to 62 Gy.

The BTDF profile widths of the samples are symmetrical. It is thus quite clear that there is no specific track-orientation, even in the case of multi-angle irradiations. Whatever the angle of irradiation, the tracks are isotropically distributed.

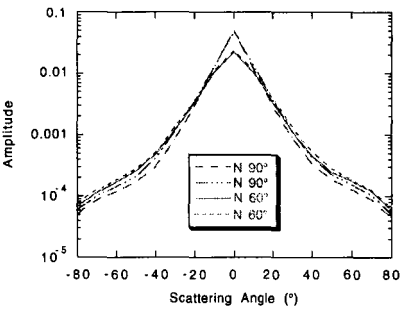

Fig. 5 : BTDF of ssntd's irradiated under neutron flux with incidence of 60° and 90°. We can note the same profile width for these BTDF. The samples have been irradiated with a dose equal to 8.12 Gy.

2- Study of the scattering.

Let us consider a specific intensity $I(\vec{r}, \mathbf{u})$ which is incident (from a distance $\vec{r}$ from the center of the radiation sphere and with a direction $\mathbf{u}$) on a elementary cylindrical volume with a unit cross section and length ds. The volume contains r.ds scattering tracks.
In transport theory **[4]**, the part of the flux which decreases as a result of scattering and absorption is referred to as the reduced incident intensity I_{ri}. Total intensity $I(\vec{r}, \mathbf{u})$ is expressed as follows :
$I(\vec{r}, \mathbf{u}) = I_{ri}(\vec{r}, \mathbf{u}) + I_d(\vec{r}, \mathbf{u})$ where $I_d(\vec{r}, \mathbf{u})$ is called the scattered specific intensity.

$$\frac{dI_{ri}(\vec{r}, \mathbf{u})}{ds} = -\rho . \sigma_t . I_{ri}(\vec{r}, \mathbf{u})$$

Thus, we show that $I_{ri} = I_0 \exp(-\rho.\sigma_t.s)$, where s is the sample thickness pierced by the laser beam.

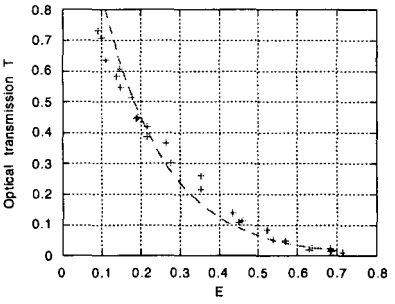

Fig. 6. Representation of T vs E.

The term σ_t refers to the optical cross-section for each particle and includes two contributions, one for light absorption and one for light scattering. We can say that the product $\sigma_t.\rho$ is similar to the exposure E defined by Ilic **[5]** - $E = S_t.\rho$, where S_t is the average track area and ρ the track density. The change of optical transmission vs E is a decreasing exponential function (Fig. 6).
The study of the scattered intensity $I_d(\vec{r}, \mathbf{u})$ must enable us to distinguish (when E is invariable) the variations of the average area of track openings as well as that of track density. The simulation of laser scattering on a set of randomly distributed tracks should enable us to eliminate all ambiguity concerning E.

CONCLUSION

This results obtained show that the study of SSNTDs that are irradiated under strong flux and then chemically developed is possible if the measurement of coherent light scattering is adopted since the data collected is of a statistical nature. The study and simulation of laser scattering by a set of tracks, randomly distributed in size and in orientation will enable us to distinguish the different morphological parameters. This will lead to a better evaluation of the dose.

ACKNOWLEDGEMENTS

We wish to thank J. Dhermain and F. Vettese (ETCA-Arcueil) for hepful discussions and for the financial and irradiation supports.

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TRACK-ETCH DOSEMETER RESPONSE TO NEUTRONS UP TO 300 MEV

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Abstract

Electro-chemical and chemical track-etch dosemeters were obtained from commercial suppliers (American Acrylics and NE Technology) and exposed to neutrons produced at the LAMPF WNR white neutron source at 15° with no shielding and filtered by polyethylene blocks of 2.5, 5.1, 10.2, 20.3 and 40.6 cm thickness. The neutron spectrum was determined using calculations. Mean energies from 28 to 300 MeV were produced. Dose was calculated from the NCRP-38 flux-to-dose conversion. The results are compared with NTA film which was exposed in the same configuration. The response of track etch dosimeters was found to reach a miniumum and then rise as the the average neutron energy increased. The response of the NTA film increased as the neutron energy increased.

Introduction

NTA film produced by Eastman Kodak is the common method for measurement of dose from high energy neutrons. An alternative is the use of proton sensitive etched track detectors. Two systems were chosen for testing: an electro-chemical etch system described by Hankins (1) and a chemical etch system used by Harvey (2). The exposures were made at the 15 degree, 90 meter flight path of the LAMPF Weapons Nuclear Research (WNR) white neutron source. Development was according to standard methods.

Experimental Method

Materials

NTA film was obtained from Eastman Kodak. The films were placed in plastic holder with a 0.066 cm cadmium pocket. The plastic holder is sealed and holds desiccant. The plastic holder is sealed in a dry nitrogen atmosphere and holds a desiccant. Dosimetry grade American Acrylics and Plastics CR-39 which was 0.063 cm thick covered on both sides with polyethylene of thickness 0.01 cm was laser cut into 3.5 cm. by 1.6 cm foils. The foils were mounted on a cardboard backing for exposure. PN3 foils (NE Technology) were precut into 2.0 cm by 2.5 cm by 0.15 cm foils. The PN3 foils were placed in a five element pyramidal holder. The holder has two dosemeters at the base of a triangular pyramid with the sides inclined at 40° to the base. Three dosemeters are placed on the faces of the pyramid. The configuration is held by a holder consisting of a base and a domed cap. The overall diameter is 5.8 cm and the height is 2 cm.

Neutron Irradiation

The neutrons were produced by an 800 MeV proton beam striking a tungsten target. Exposures were made to the unfiltered beam and with polyethylene filters of 2.5, 5.1, 10.2 , 20.3 and 40.6 cm thickness. Electronic dosimeters (ALOKA CO., LTD., PDM-303) were used to scan the field uniformity. The beam was flat to within 5% over a square field of 10 cm width.. The dosemeters were exposed on an ISO water phantom. The dose for each irradiation was determined using a neutron beam monitor, a small plastic scintillator, normalized by the proton current on the target. The dose per proton was calculated using the LAHET code, the calculation provided the neutron spectra per proton/cm^2 /MeV at the tungsten spallation target. The neutron spectra were then corrected for attenuation by the polyethylene filter and for air in the 90 meter flight path. The NCRP-38 fluence-to-dose conversion was then used to convert the neutron spectrum to a dose equivalent. A 5 kG magnet at the front of the detector shed was used to remove charged particles from the beam. The spectra are shown in Figure 1.

Development

The NTA film are developed in 68 °F baths. The first bath is a Kodak Rapid X-Ray Developer for five minutes with constant agitation. Development is stopped by a two minute running water bath. The film is transferred to a Kodak fixer bath for thirty minutes and then to a final wash bath with running water for thirty minutes. The electrochemical method used CR-39 (American Acrylics) with a three cycle etch. The dosemeters are placed in a 60°C oven overnight before the read cycle begins. The first cycle is a pre-etch for 1.75 hours in 6.5 N KOH solution which is added at the beginning of the cycle. Next a three hour cycle with 3000 volts at 60 Hz is used followed by 3000 volts at 2 KHZ for fifteen minutes. The temperature for all three cycles is 60°C. The dosemeters were taken from one sheet. Each batch of twenty-four foils contained four background foils and four control foils which were used for normalization between runs. The chemical etch procedure used PN3 plastic from NE Technology taken from one sheet. The dosemeters are given a pre-etch in a mixture of 60% by volume methanol and 6.25 N NaOH for one hour at 70°C. This is followed by six hours in a 6.25N NaOH solution at 70°C and ending with a through washing.

Track Counting

The NTA films were read using a microscope with a field size of one mm^2. The CR-39 foils were read using a standard microscope with a video camera. The image of the camera is input to a Biotran III Digimatic Indicator. Three fields totaling 0.6 cm^2 are read to determine the track count. The PN3 foils were read using the NE Technology AUTOSCAN 60. The reader used light incident on the edge of the PN3 plastic which undergoes total internal reflection from the faces where a defect or pit exists. The foil is viewed through a vidcon tube and camera lens and processed using an image grabber by a personal computer. The area read is 1.6 cm^2.

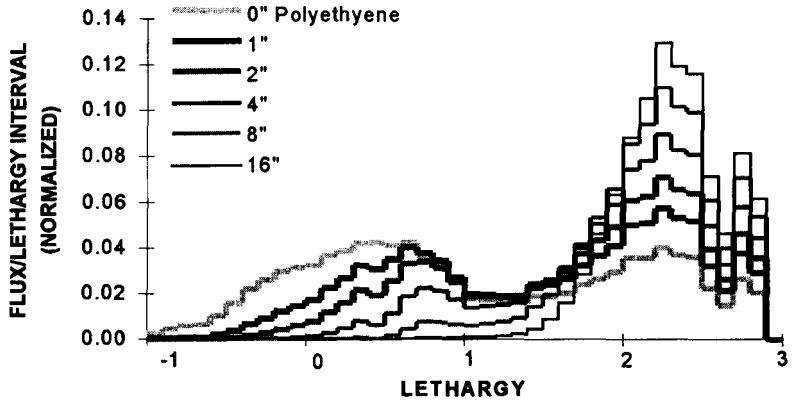

Figure 1. Fluence per Unit Log Energy (Lethargy) Normalized to Total Flux

Results

The results are presented below in Table 1. An interesting observation is the common effect in both plastic dosemeters of a minimum of response ration followed by a rise in response. This would be consistent with the rise in kerma as the energy increases above 70 MeV to 1000 MeV.. A corresponding increase is also observed in the NTA film. The results for the spectra corresponding to lower average energies are consistent with previous work (3). Further studies which could be useful would be around 35 MeV and 66 MeV to fill in the middle energy portion. These studies when combined with the known energy response function from 0.2 MeV to 20 Mev and both a Monte Carlo estimate and comparison with kerma should allow a reasonable estimate of the energy response.

Table 1 Dose Equivalent Response of NTA Film, CR-39 and PN3 to High Energy Neutron Spectra

Material	Bare Cf Response ± SD ($cm^{-2}mSv^{-1}$)	Background ± SD	Filter Thickness (cm)	Response Relative to Bare Cf N	Ratio ± SEM
NTA	1090 ± 120	one mm^{-2} (Poisson Distributed)	0.0	2	1.88 ± 0.29
			2.5	2	2.40 ± 0.35
			5.1	2	2.48 ± 0.34
			10.2	2	4.33 ± 0.56
			20.3	2	5.40 ± 0.69
			40.6	2	6.11 ± 0.81
CR-39	399 ± 30	50 ± 14 cm^{-2}	0.0	3	0.27 ± 0.02
			2.5	3	0.33 ± 0.04
			5.1	3	0.16 ± 0.03
			10.2	3	0.11 ± 0.01
			20.3	3	0.11 ± 0.01
			40.6	3	0.15 ± 0.02
PN3 Planar	212± 13	95 ± 10 cm^{-2}	0.0	4	0.36 ± 0.03
			2.5	4	0.51 ± 0.05
			5.1	4	0.45 ± 0.04
			10.2	4	0.35 ± 0.05
			20.3	4	0.39 ± 0.03
			40.6	4	0.50 ± 0.05
PN3 Pyramid	189 ± 10	95 ± 9 cm^{-2}	0.0	6	0.49 ± 0.03
			2.5	6	0.55 ± 0.05
			5.1	6	0.50 ± 0.07
			10.2	6	0.38 ± 0.06
			20.3	6	0.37 ± 0.07
			40.6	6	0.52 ± 0.04

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PAPER TITLE: HIGH SENSITIVITY, REUSABLE SUPERHEATED LIQUID NEUTRON SENSOR

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MAJOR SCIENTIFIC TOPIC NUMBER: 3... (see page 7)

ABSTRACT (See instructions overleaf)

Superheated liquid neutron sensor containing superheated liquid drops of CCl_2F_2 (M-12 India) in Gel medium has already been reported (Radiation Measurements, Vol. 23 No. 4 pp. 753-755, 1994). Recently authors have developed a superheated liquid neutron sensor (SLNS-2) based on polymer medium. This sensor has better long term stability compared to sensor having gel medium. The temperature range of stability is 10 to 35°C. The experimental results of this sensor pertaining to stability, neutron dose response has already been reported. (Radiation Measurements, 1995 in press).

A reusable device containing SLNS-2 as sensor has been developed recently. The device is portable and can be used in active as well as passive mode also. The device is made of polycarbonate tube of 20 mm Diameter and is 124 mm in length. It is attached with a repressurizing system. The diameter of droplets is in the range 15-20 microns. The density of droplets can be controlled as per sensitivity requirement. The highest sensitivity achieved so far is 60 bubbles/ 10fSv. The device is insensitive to neutrons when the sensor is pressurized by the built in pressurizing system. When the pressure is released, the sensor is ready for use. The bubbles formed due to neutron exposure can be recompressed by pressurizing the device. The performance of the device pertaining to it's response to various neutron sources, shelf life, reusability, temperature dependence, storage and transport conditions will be discussed.

BUBBLE DETECTORS IN INDIVIDUAL NEUTRON DOSIMETRY

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INTRODUCTION

Bubble detectors would be able to fulfill requirements following from the ICRP 60 recommendations as far as individual neutron dosimeter is concerned. Particularly, the lowest limit of detection seems to be decreased down to about 10 μSv or even lower. At the moment, there are two types of such detectors commercially available: bubble damage neutron detectors (1) (BDNDs - Bubble Technology Industries, Chalk River), and superheated drop detectors (2) (SDDs - Apfel Enterpr., New Haven). Both these types have been tested in our studies from the point of view of personal dosimetry. Particular attention is devoted to their energetical dependences and their responses in the fields in which they should be used to determine occupational exposures.

EXPERIMENTAL CONDITIONS

BUBBLE DETECTORS TESTED

BDNDs' tested had issued from two deliveries. The first one was constituted from the type BD 100R (threshold neutron energy 100 keV) of two different sensitivities to AmBe neutrons (0.8 and 2.2 bubbles per 1 μSv of H*(10)). The second delivery constituted again of BD 100R (1.0 bubble per 1 μSv), not compensated for the temperature, and temperature compensated PND of the same nominal sensitivity and similar energy dependence.

SDDs' tested had issued from the single delivery. It consisted of AMS reader and sets of detectors with three different energy thresholds: 100 keV (SDD 100), 1 MeV (SDD 1000) and 6 MeV (SDD 6000). The sensitivity of all of them ought to be about 3.3 bubbles (counts) per 1 μSv of H*(10) from neutrons with energies above the threshold.

NEUTRON FIELDS USED FOR TESTING

The response of bubble detectors has been studied in neutron beams and fields characterized in Table 1.

Table 1. Neutron fields used for testing.

Neutron field	$\bar{E}_N$ [MeV]	H*(10) [1] [Sv.cm^2]	Neutron field	$\bar{E}_N$ [MeV]	H*(10) [Sv.cm^2]
SIGMA (3)	0.070	2.29 -11	PuBe	4.2	3.80 -10
CANEL+ (H_2O) (4)	0.096	3.47 -11	AmBe	4.4	3.80 -10
CANEL+ (4)	0.185	7.26 -11	JINR - soft (5)	0.25	2.90 -11
^{252}Cf/D_2O/Cd	0.54	9.10 -11	CERN - iron (6)	1.9	1.60 -10
AmF	1.5	3.40 -10	JINR - hard (5)	12.5	1.05 -10
^{252}Cf	2.1	3.40 -10	CERN - concrete (6)	49.8	2.80 -10

[1] ICRP 21 conversion factors

RESULTS OBTAINED

Multiple irradiation of several pieces of BDNDs' (type PND) has been realized with neutrons of an AmBe source. The standard deviation of a single reading was found to be close to $\sqrt{N}/N$, where N is the number of bubbles. The precision about ± 30% is hence achieved when total number of bubbles is above 10.

The relative responses of BDNDs' studied in different neutron fields are presented in Table 2.

Table 2. Relative responses of BNDDs' (BD 100R and PND - 2nd delivery) to neutrons.

Neutron source	Relative response	
	BD 100R	PND
SIGMA	0.52 ± 0.04	0.54 ± 0.04
CANEL+ (H_2O)	1.13 ± 0.09	1.06 ± 0.08
CANEL+	1.22 ± 0.09	1.06 ± 0.08
AmF	1.10 ± 0.08	1.16 ± 0.08
^{252}Cf	1.34 ± 0.14	1.38 ± 0.15
AmBe	1.00[1)]	0.94 ± 0.07
PuBe	1.05 ± 0.07	1.03 ± 0.07
JINR - soft	-	1.13 ± 0.13
CERN - iron	1.28 ± 0.13	1.17 ± 0.10
JINR - hard	-	0.70 ± 0.07
CERN - concrete	0.63 ± 0.04	0.62 ± 0.04

[1)] Taken as a reference

One can see there that, with the exception of thermal neutron source SIGMA (50% of H*(10) from thermal neutrons) and high energy reference fields the relative response of both BDNDs' is not too far from 1.0. However, it should be mentioned that the results obtained with BD 100R of the 1st delivery have been slightly different, generally about (19±5)% lower for sources CANEL+ (H_2O), CANEL+, AmF, ^{252}Cf and CERN-iron, i.e. closer to 1.00 as compared with results presented in Table 2. It could be connected with slightly different energy dependence, the BD 100R from the 2nd delivery being relatively more sensitive to neutrons with energies from 0.1 to 1-2 MeV.

The relative responses of SDDs' studied in different neutron fields are presented in Table 3.

Table 3. Relative responses of SDDs to neutrons.

Neutron source	Relative response of SDD with the threshold, [MeV]		
	0.1	1.0	6.0
SIGMA	0.60 ± 0.05	0.12 ± 0.01	0.010 ± 0.003
CANEL+ (H_2O)	0.23 ± 0.03	0.025 ± 0.005	0.0054 ± 0.0010
CANEL+	0.48 ± 0.05	0.006 ± 0.002	0.0028 ± 0.0005
AmF	0.66 ± 0.05	0.081 ± 0.010	0.028 ± 0.005
^{252}Cf	1.03 ± 0.06	0.17 ± 0.02	0.090 ± 0.010
PuBe	0.90 ± 0.08	0.35 ± 0.03	0.16 ± 0.02
AmBe	1.00[1)]	0.38 ± 0.02	0.16 ± 0.02
CERN - iron	0.70 ± 0.10	0.17 ± 0.02	0.09 ± 0.02
CERN - concrete	0.50 ± 0.05	-	-

[1)] Taken as a reference value

One can see there that:

- the drop down in the relative responses for neutrons with energies going down below 1 MeV to 100 keV is faster for SDD 100 as compared with BD 100R or PND;
- the comparison of responses of SDD 100 with SDD 1000 (or 6000) permits clearly to distinguish fission from more energetic neutrons;

the relative response of SDD to high energy neutrons (CERN - concrete) is also, as for BDNDs, lower.

There are two types of occupational exposures for which the use of bubble detectors could be particularly fruitfull:

- degraded spectra of fission neutrons behind the shielding of power and/or research reactors;
- neutron fields behind the thick shielding of primary high energy particles (high energy accelerators, cosmic radiation in the Earths atmosphere.

In both cases, bubble detectors can be used as an individual dosimeter. At reactors, BDNDs with threshold 100 keV can be used even without important corrections on the actual neutron spectrum. As far as SDDs and both types using in high energy neutron fields are concerned, it seems to be appropriate to multiply their readings by a factor of about two to compensate for their energetical dependence.

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USE OF BUBBLE DETECTORS FOR THE DOSIMETRY OF COLD NEUTRONS

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INTRODUCTION

Several different types of detector are presently used for health physics oriented neutron detection and measurements. These detectors are based on a wide variety of different principles, ranging from activation reactions induced by neutron radiation in materials, nuclear reactions in gas counters, as well as on the use of photographic emulsions or solid state track detectors.

Several years ago Apfel (1-2) and Ing (3) developed a new type of detector, bubble detectors, which have ever since raised considerable interest among radiation protection specialists. These neutron detectors are insensitive to X and γ radiations and are extremely useful for mixed photon + neutron field measurements.

The present article describes studies performed in Orphée, a nuclear reactor, located in the Saclay Nuclear Research Center, providing different wavelength ($\lambda < 20$ Å) cold neutron beams. We have thus determined the response of « thermal neutron » bubble detectors from 3 to 15 Å (i.e. corresponding to energies in the 9 meV to 0.36 meV range) and compared the results obtained with those determined using Li-6F.

Images obtained with a scanner reveal that the heterogeneity in the distribution of bubbles in the detection medium is a function of λ.

1) REMINDER OF PRINCIPLES

Bubble detectors are based on a principle similar to the one used for the bubble chambers employed in high energy physics. Bubble detectors consist of a gel or transparent polymer in which micro droplets of freon have been dispersed. In the absence of radiation these droplets are thermodynamically stable and capable of remaining in a metastable state over a period of several months. When exposed to a neutron field, charged secondary particles induce these droplets to undergo a liquid-to-gaseous phase change. Bubbles can be seen with the naked eye. The dosemeter composition is such as to assure proportionality between the number of bubbles and the dose equivalent (ref. NCRP 38). Dosemeters commercialized by Bubble Technology Industries cover a range of sensitivities (0.03 to 3 bubbles/μSv) together with different energy ranges (thermal energies, energies in the 200 keV to 15 MeV range etc.).

2) MATERIALS AND METHODS

Measurements were performed with BDT (thermal) type bubble detectors from Bubble Technology Industries with sensitivities in the 1.5 to 3 bubbles.μSv^{-1} range for thermal neutrons (sensitivity given by the manufacturer for a temperature of 20 °C).

The experimental equipment used is shown in Figure 1. In order to ensure measurement reproducibility, the detector is housed in a support located at a distance of one centimeter from the beam window (3 cm in diameter). The position of the detector with respect to the beam was checked with a Polaroid photograph. In this way, a corrective factor was determined to take into account the fact that the detector is not completely exposed to the beam. Exposure times, varying between 30 and 90 seconds were measured with a chronometer.

Prior to the positioning of the bubble detector in the beam, flux measurements were performed with an RTC make 4.74 mm diameter CFUF.32.S type fission chamber. Chamber measurements were corrected to take the variations in response as a function of neutron wavelength into consideration.

Measurements were also performed using simultaneously exposed Li-6F (sensitive to both neutrons and photons) and Li-7F (only sensitive to photons) detectors. The dose equivalent due solely to neutrons is obtained by subtraction.

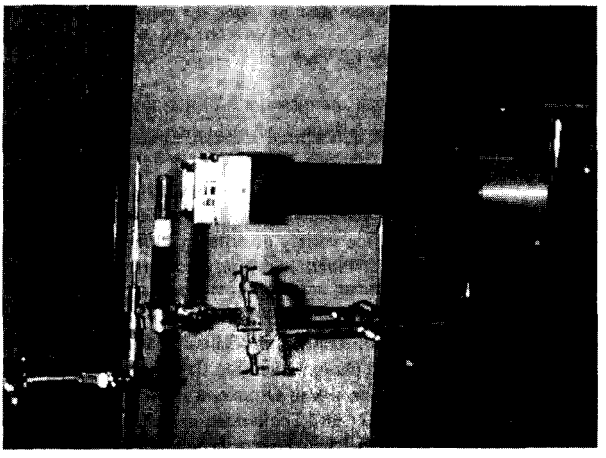

Fig. 1 : Bubble detector and fission chamber in the cold neutron beam

It should be noted that « equivalent thermal neutrons » units are employed for both LiF and bubble detector measurements. It is assumed that bubble-dose equivalent conversion factors or digital - dose equivalent factors remain constant for $\lambda > 1.8$ Å values.

3) RESULTS AND DISCUSSIONS

Figure 2 indicates that bubble detector and thermoluminescent detector responses slightly increase with λ. Extrapolation of the experimental curve to the theoretical « thermal wavelength » (1.8 Å) is in good agreement with the « thermal » point calculated for reference NCRP 38 (4) and shown in Figure 2.

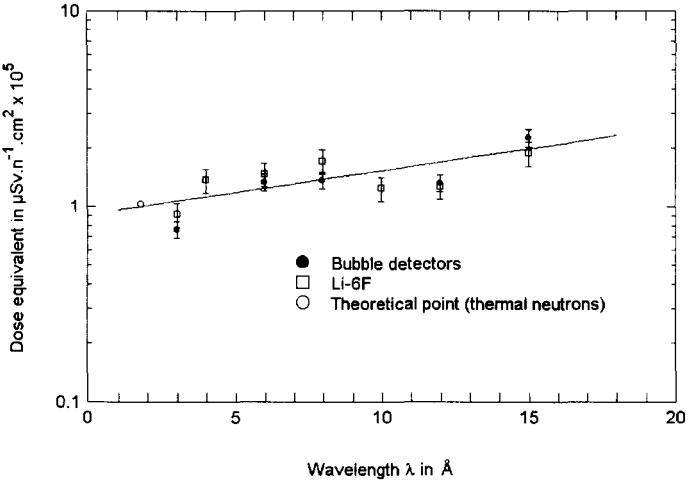

Fig. 2 : Response of Bubble detectors and Li-6F for cold neutrons

Attention is drawn to the fact that dose equivalent (in thermal equivalent units) is practically independent of λ (to within ± 30%), thus enabling bubble detectors to be used as fluence meters. Besides being easy to use in comparison with fission chambers, these detectors also present the advantage of providing a direct readout in terms of bubbles-n.cm^{-2} and do not necessitate the use of any λ dependent correction factor.

Finally, images obtained with a scanner for 3, 5, 10 and 15 Å radiations reveal an increasing bubble distribution heterogeneity with λ (see Figure 3).

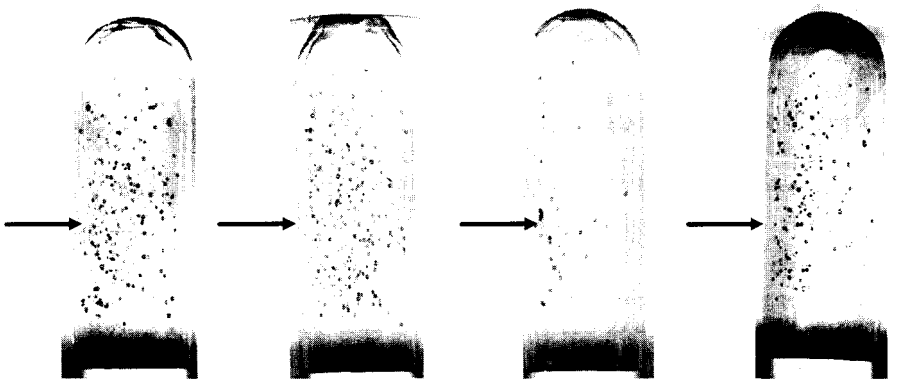

Fig. 3 : Bubble detectors irradied by different wavelength neutrons (3, 5, 10 and 15 Å) (⟶ beam direction)

4) CONCLUSIONS

Radiation protection requirements can be satisfied with the sensitivities and operational energy ranges covered by bubble dosemeters. The responses of bubble detectors have been validated by different laboratories (5,6,7). This preliminary study shows that the « thermal neutron dose equivalent or fluence » responses of these dosemeters to cold neutrons with energies in the 9 meV to 0.4 meV range are practically energy independent (to within ± 30%); furthermore the responses of Li-6F and bubble dosemeters have been shown to be very similar. Radiation protection specialists thus have at their disposal a new easy-to-use tool with a satisfactory response for fluence or dose equivalent estimations in or in the vicinity of cold neutron beams.

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THREE-DIMENSIONAL ABSORBED DOSE DETERMINATIONS BY N.M.R. ANALYSIS OF PHANTOM-DOSIMETERS

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INTRODUCTION

Magnetic resonance imaging of a tissue-equivalent phantom is a promising technique for three-dimensional determination of absorbed dose from ionising radiation. A reliable method of determining the spatial distribution of absorbed dose is indispensable for the planning of treatment in the presently developed radiotherapy techniques aimed at obtaining high energy selectively delivered to cancerous tissues, with low dose delivered to the surrounding healthy tissue.

Aqueous gels infused with the Fricke dosimeter (i.e. with a ferrous sulphate solution), as proposed in 1984 by Gore et al. (1-2), have shown interesting characteristics and, in spite of some drawbacks that cause a few limitations to their ulilisation, they have shown the feasibility of three-dimensional dose determinations by nuclear magnetic resonance (NMR) imaging.

Fricke-infused Agarose gels with various compositions have been analysed, considering the requirements of the new radiotherapy techniques, in particular Boron Neutron Capture Therapy (B.N.C.T.) and proton therapy. Special attention was paid to obtain good tissue equivalence for every radiation type of interest. In particular, the tissue equivalence for thermal neutrons, which is a not simple problem, has also been satisfactorily attained.

The responses of gel-dosimeters having the various chosen compositions have been analysed, by mean of NMR instrumentation. Spectrophotometric measurements have also been performed, to verify the consistence of the results.

METHOD

The proposed system is composed of a ferrous sulphate solution (which is the main constituent of the standard Fricke solution) incorporated into a polysaccharide (Agarose SeaPlaque, from the Fluka Chemical Corporation). Agarose is a gelling agent whose role is that of maintaining the spatial localisation of the absorbed dose.

In ferrous sulphate solutions ionizing radiation produces a conversion of ferrous ions (Fe^{2+}) to ferric ions (Fe^{3+}). A modification in some parameters of the system is therefore produced and may possibly be detected. The standard Fricke dosimetry takes advantage of the modification of the optical absorption in the visible spectrum, measurable by means of spectrophotometric instrumentation. In addition, both the spin-spin relaxation rate (R_2) and spin-lattice relaxation rate (R_1) have been shown to be linearly correlated to the absorbed dose, and the linearity is good up to a dose that is dependent on the dosimeter composition, because it requires that the Fe^{2+} ions are not significantly depleted.

We have optimized (3) the protocol for the gel-dosimeter preparation, and therefore we have analysed the response of dosimeters having various compositions. The best sensitivity was achieved with the following composition of the dosimeter:

ferrous sulphate solution	*1 mM $Fe(NH_4)$ $6H_2O$, 50 mM H_2SO_4* in the amount of 50% of the final weight
Agarose SeaPlaque [$C_{12}H_{14}O_5(OH)_4$]	in the amount of 1% of the final weight
highly purified water H_2O	in the amount of 49% of the final weight.

Samples of this gel, contained in cylindrical polyethylene vials 2.5 cm in diameter and 7 cm in height, were exposed to the γ-ray field of a ^{137}Cs biological irradiator, acting at a rate of 0.14 Gy s^{-1}, and than analysed in a Somatom Siemens NMR imaging system operating at 1.5 T and 63 MHz. The transverse relaxation rate $1/T_2$ was measured, because it is more sensitive to ferric ion variations than the longitudinal relaxation rate $1/T_1$, in such low frequency instruments. The dose-response curve is found to be linear up to ≈ 40 Gy, with a slope equal to 0.2 s^{-1} Gy^{-1}. The G value (Fe^{2+} oxidized per 100 eV absorbed energy) of this dosimeter is equal to 183 (Fe^{3+} ions per 100 eV). Good result reproducibility is achieved if the dosimeter response R is defined as the difference between the relaxation rate measured in the irradiated sample and that measured, at the same time, in a non-irradiated sample from the same gel preparation:

$$R = (1/T)_{irr} - (1/T)_{blank}.$$

Samples with different gel compositions prepared, as described below, to satisfy certain specific requirements, such as tissue equivalence for thermal neutrons, have been shown to be less sensitive. In any case, the sensitivity is considerably higher than that of the standard Fricke dosimeter.

The tissue equivalence of all gel dosimeters to γ-rays is very good.

SPATIAL DETERMINATION OF ABSORBED DOSE IN B.N.C.T.

In tissue exposed to thermal neutrons the absorbed dose is due to the energy released by the reaction products. Consequently, a good tissue equivalence is obtained only if the isotopic composition of the substitute is identical to that of the tissue to be simulated, at least as regards the isotopes that give the main contributions to the absorbed dose, namely (in all tissues but bone) ^{14}N and ^{1}H, which give the reactions:

$$^{14}N(n,p)^{14}C \qquad ^{1}H(n,\gamma)^{2}H$$

The nitrogen contribution to the absorbed dose is dominant in small volumes (< 1 cm^3). Therefore we have augmented the gel previously described with the organic compound carbonyldiamide (urea [CH_4N_2O] from the Fluka Chem.Corp.) in the amount of 4% of the final weight. The dosimeter so obtained is a good substitute for brain tissue. In fact, the elemental composition of the gel is found to be very near that of brain (from ICRU-44, 1989), as is shown in the Table. Moreover, Z_{eff}(brain) = 7.4 and Z_{eff}(gel) = 7.8.

Some dosemeters have also been prepared by augmenting the gel with 40 μg/g of ^{10}B, which is the quantity typically accumulated in tumours. Both the dosemeters (i.e. with and without ^{10}B) have been shown to have the same sensitivity to gamma-radiation.

We have analysed (4) dosemeters made with both boronated and non-boronated gel, contained in small cylindrical Teflon vials. When the dosemeters with boron are exposed in the thermal column of a TRIGA MARK II reactor, 86% of the total response is found to be related to the absorbed dose due to ^{10}B. The remaining part, that is 14% of the total, includes contributions from thermal neutrons in tissue, fast neutrons, reactor background and activation of the materials of the containers and holders. The sensitivity of the boronated dosemeter to the secondary particles produced by thermal neutrons reacting with ^{10}B is found to be equal to 0.028 D_B, where D_B is the absorbed dose (in Gy) due to ^{10}B. In Fig.1 the image of a boronated gel phantom after exposure in the thermal column of the reactor is shown. The results confirm the promising features of this NMR dosimeter, and have encouraged us to pursue investigations by analysing broader samples.

TABLE

	PERCENTAGE BY MASS				
	H	N	C	O	al.
BRAIN	10.7	2.2	14.3	71.2	1.4
GEL	10.9	2.2	1.4	85.4	0.1

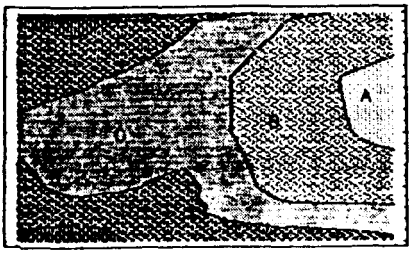

FIG.1

DEPTH-DOSE PROFILING IN PROTON THERAPY

The feasibility of utilising a Fricke-infused Agarose gel for depth-dose profile determination in tissue exposed to a proton beam has been investigated. Considering the fact that ion diffusion requires a prompt NMR analysis after irradiation, if a good spatial resolution is needed, and since at present we do not have the possibility of performing such rapid analyses, we have carried out the experiment by employing the following alternative technique: thin glass capillaries (1.1 mm internal diameter), filled with the ferrous-sulphate-infused gel, were pierced in a plexiglass phantom; the phantom was exposed to a 13.5 MeV proton beam from a CYPRIS cyclotron and, the next day or later on, they were measured in a research NMR analyser (BRUKER AC-300) operating at 7.05 T and 300 MHz. Owing to the high operation field, the longitudinal relaxation rates $1/T_1$ were determined.

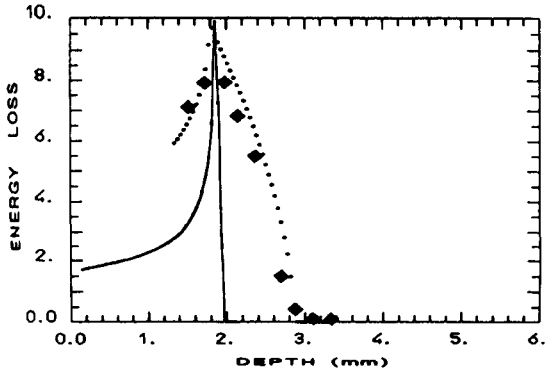

Fig.2

The results are shown in Fig. 2, where each experimental point is obtained as a mean of two values. In the figure the computed curves are also shown: *a) (solid line)* specific energy released in Plexiglas by 13.5 MeV protons as a function of depth, and *b) (dotted line)* energy released in a gel dosimeter enclosed in glass containers with square section, 1mm internal side, 0.2 mm thickness, located at increasing depths in plexiglass. The Bragg peak position is obtained within a tenth of a millimeter uncertainty, and the widening in the ramps is within 1 mm.

Moreover, we have also tested our dosimeter gel by means of spectrophotometric analysis in order to verify the consistence and correctness of the results obtained from the NMR analysis. To this end, we have utilised a UV/VIS Diode-Array Spectrophotometer. An enhancement of the spectrophotometric measurement sensitivity is obtained by adding to the gel components a proper metal ion indicator, which yields absorption in the visible spectrum. We have chosen Xylenol Orange ($C_{31}H_{28}N_2Na_4O_{13}S$), which shows a peak at 580 nm, in the amount of $1.25 \cdot 10^{-4}$ M. We have analysed samples exposed both to gamma and to proton radiation: the good agreement of these results with those obtained by NMR analysis are a further verification of the validity of the method.

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Application of Low Pressure Tissue Equivalent Proportional counter for IHEP Radiation Protection

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Radiation behind shields of high energy charge particles accelerators (for example, the IHEP 70 GeV proton synchrotron (U-70)) is of a mixed composition. The major component of radiation is high energy neutrons with energy above 20 MeV. The dose equivalent measurement of high energy neutrons is a difficult problem for the radiation protection dosimetry, because there are no standard devices for this energy region. Now only a measurement method on the base of low-pressure tissue equivalent proportional counter (TEPC) is used as universal in mixed radiation fields. This dosimetric method has more than 20 year history of application at the IHEP accelerator. The method has been improved in measurements, the counter design and application. The linear energy transfer (LET) spectrometer (SLET-03) on the base of TEPC[1] is used for metrological measurements in IHEP radiation protection.

The spherical tissue equivalent proportional counter has been developed in collaboration with the Institute of Biophysics (Moscow) [2] .The TEPC energy response for neutrons and photons is dependent on the counter design (the thickness and the composition of the counter body, the thickness of the aluminum cover). The dependence of TEPC neutron energy response on the counter design has been shown in [2], [4]. The photon energy response of TEPC in the energy range from 29 to 114 keV has been investigated by the National Secondary Standard Field of absorbed dose (VNIIFTRI) [3].

The energy responses has been calculated for neutrons from 0.01 to 800 MeV [4] , [2]. The TEPC energy response agrees closely with H*(10) in the neutron energy range for 0.3 MeV to 800 MeV. The TEPC response is less than H*(10) function for neutron below 0.3 MeV. Table reflects the TEPC measurements results over the ones obtained with other dosimeters. There is the normalization with the TEPC data and the normalization with the expert data for the IHEP reference fields.

Irradiation fields	Dosimeters	Neutron Dose equivalent, rel.units	Photon and charged particles dose equivalent, rel. units
Radiation field behind the top of the IHEP proton accelerator. The average neutron energy is equal to 71 MeV.	TEPC	1.0	1.0
	ACR	0.60	1.12
The radiation fields behind the iron calorimeter . The average neutron energy is 51 MeV.	TEPC	1.0	1.0
	ACR	0.36	0.98
Pu-Be source (IHEP reference field)	TEPC	1.0	1.01
	ACB	1.0	0.83
	BS	0.99	-
The IHEP reference fields based on the ^{252}Cf neutron source.	TEPC	0.95	0.90
	ACB	0.99	0.80
	BS	1.06	----
The reference field based on the ^{252} Cf neutron source in the 30 cm diameter spherical iron moderator.	TEPC	1.00	0.80
	ACR	0.75	0.0
	BS	0.99	---
The reference field based on the ^{252}Cf neutron source in the 30 cm diameter spherical polyethene moderator.	TEPC	0.95	1.07
	ACR	1.63	0.99
	BS	1.04	---

The comparison was carried out with the following dosimetric systems :

The analog component remmeter (ACR) [1], that includes an argon-filled ionization chamber, tissue-equivalent chamber, ^{3} He-filled ionization chamber into the 25.4 cm diameter spherical polyethylene moderator. To minimize the neutron dose equivalent measurement systematic error, the correction method based on additional information about a behavior of the ionization chamber neutron response in different neutron spectra is applied. The Bonner multisphere neutron spectrometer (the dose equivalent measurement method by 6-spheres has been certified by VNIIM)[5].

The photon dose equivalent data carried out by TEPC are not significant as compared with those measured with other dosimeters. The comparative results show that the TEPC data are systematically lower than those of the Bonner spectrometer within 20 % .The systematic error of neutron dose equivalent measurement may be reduced with a correction factor ,when the average spectra neutron energy is less than 0.5 MeV. The correction factor may be obtained by a procedure based on neutron energy dependence of the ratio of neutron response of TEPC to neutron response of another detector [6].

The application of TEPC for IHEP radiation protection dosimetry showed that one may be successful in a variety of radiation fields of a high energy accelerator.

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PERFORMANCE OF RADIOACTIVE MEASUREMENTS BY GAMMA SPECTROMETRY WITH SEMICONDUCTOR DETECTORS USING BIPHASIC SAMPLES

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ABSTRACT

An adaptation of the solid-liquid extraction technique has been developed to determine Hg-203 by gamma spectrometry with a semiconductor detector using biphasic samples, as a part of a study on the retention of mercury on the external membrane of bacterial cells. Three procedures have been tested to extract the radioactive isotope. One of them, consisting of a one-stage extraction using an optimized liquid volume, was seen to be preferable in this case. The three procedures might be useful to monitor the uptake of a contaminant (radioactive or not) by a solid phase, especially in small samples.

INTRODUCTION

A variety of techniques have been applied to the separation of toxic elements. Sorption of metals by microorganisms has proved useful in different types of samples (1). Living and dead cells are able to retain certain elements, like mercury, in a selective way (2). The monitoring of mercury in environmental and biological samples is of great concern because of the high toxicity of this metal. The use of a very sensitive technique like gamma spectrometry may be appropriate to determine mercury in very low concentrations.

Some series of experiments on the retention of mercury on the external membrane of bacterial cells were to be carried out using Hg-203 as a radiotracer (3). The size and shape of the samples are important factors to consider in gamma spectrometry with semiconductor detectors. The use of monophasic samples is usually considered an indispensable condition in this technique. However, it was difficult to obtain homogeneous samples with a specified shape from the small amount of bacteria used in the experiments. On the other hand, repeated separation of phases by solid-liquid extraction may lead to a decrease in accuracy. In these cases the performance of radioactive measurements in biphasic samples could be an option.

MATERIALS AND METHODS

A Canberra GR 2520 high purity Ge shielded detector was used for counting the 279.2 keV gamma rays of Hg-203 , in combination with a Canberra System 30 Plus multichannel analyzer. Detector efficiencies were calculated and radioactive measurements were optimized before gamma counting.

The Hg-203 radiotracer solution was in the form of mercury (II) chloride. The specific activity of the labeled solutions was 522.5 Bq/μg Hg. Nitric acid and other chemicals used in

this study were of the highest quality.

The bacterial cells used were Escherichia coli and Pseudomona putida. They were first cultivated in a solid medium TSA at suitable conditions. The bacteria were pelleted and lyophilized.

A 25 mL aqueous solution containing the radioactive mercury was equilibrated with 25 mg of the lyophilized bacteria for 30 minutes at room temperature at the working pH. The bacteria were separated from the supernatant by centrifugation at 12400 g, and then treated with a fixed volume of a 3.6 M nitric acid leaching solution. The amount of mercury was determined by gamma spectrometry in the biphasic sample obtained and in also in the supernatant. At least three measurements were made for each point.

Living cells, cultivated in a TSB medium, were also used. The density of the bacterial suspension was controlled by measuring the absorbance at 600 nm. Then the suspension was treated as in the lyophilized cell experiments.

RESULTS AND DISCUSSION

A problem was initially present because counting was made in a Hg-biomass pellet treated with a nitric acid solution and a part of the Hg could still be retained by the pellet. The optimization of radioactive measurements by the extraction process was performed in three ways.

One-stage extraction and use of an empirical factor

The following equation was used,

$$\Sigma C_t = \Sigma C_1 (E_p/E_1) + \Sigma C_p f \qquad [1]$$

where C_t are the total counts, C_1 and C_p are the counts from the supernatant and pellet, and E_p and E_1 are the detector efficiencies for the radiation from the pellet and supernatant, respectively. The efficiency varies because the volume of the biphasic sample was 4 mL, while that of the supernatant was 25 mL. The summatory means that counting was made in a number of different samples. An empirical statistical factor is introduced taking into account that the mercury content in the biomass pellet is not homogeneously distributed into the 4 mL volume. The values found for this factor, f, were 0.591 for E. coli and 0.517 for P. putida.

Two-stage extraction

The following set of equations was derived,

$$C_p = C_{12}(E_p/E_1) + xE_0 \qquad [2] \qquad x = (1-s)y \qquad [3]$$

$$y = C_{12}/E_1 s \qquad [4] \qquad C_{p2} = C_p(1-s) \qquad [5]$$

$$z = C_1/E_1 \qquad [6] \qquad P(\%) = [y/(z+y)]\,100 \qquad [7]$$

where C_{p2} are the counts in the second pellet sample (the second biphasic sample obtained after the separation of the 4 mL nitric acid solution and treating again the pellet with a 4 mL nitric acid solution, C_{12} is the counting in the supernatant from the first 4 mL volume after dilution to 25 mL, x is the

number of gamma rays emited in the time unit by the Hg present in the solid phase of the first biphasic sample, y and z have the same meaning as x in relation to the Hg content in the whole first biphasic sample and in the first supernatant, respectively, E_0 is the detector efficiency for samples situated at the bottom of the sample container, s is a separation factor meaning the fraction of Hg present in the biomass and extracted by the leaching acid solution going to the solution named "12", and P(%) is the percentage of the Hg initially retained by the pellet. The separation factor values were 0.92 for E. coli and 0.60 for P. putida.

One-stage extraction using an optimized volume

The usual one-stage extraction was employed combining it with volumes of solution varying from 4 to 100 mL in order to obtain the optimum volume. The equation used was

$$C_t = C_1(E_4/E_{25}) + C_p(E_4/E_v) \qquad [8]$$

where E_4, E_{25} and E_v are the detector efficiencies for samples with a volume of 4, 25 and v mL, respectively. We have found that the counts from the biphasic samples (corrected for the different efficiencies) decrease with the incresing volume of leaching acid. For volumes greater than 25 mL there is no variation, so the optimum volume is assumed to be 25 mL.

The standard errors of the mean values obtained using identical samples by the three extraction procedures, in the same order as described, were respectively 3.1%, 1.7% and 1.4% for E. coli, and 9.2%, 4.0% and 3.3% for P. putida. Using 4 mL of extractant liquid the results are only approximate. However, using 25 mL of leaching acid the Hg is completely released. In the case of P. putida the acid leaching is less effective, and hence the errors are greater.

CONCLUSIONS

Three procedures to measure gamma radiation from biphasic samples have been tested. One-stage extraction and use of an empirical statistical factor is a simple procedure needing less measurements, but it is less accurate. The two-stage extraction procedure improves the accuracy, but the operational complexity is greater. One-stage extraction with an optimized volume of liquid had the best accuracy and was preferable in this case, though more measurements are needed. The three procedures might be useful, according to the circumstances, especially in small biological or other types of samples, and particularly in cases in which the distribution of a contaminant (radioactive or not) between two different media should be monitored.

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PAPER TITLE RADIATION DAMAGE AND PROTECTION OF PLASTIC SCINTILLATORS WITH A BASE OF POLYSTYRENE AND EPOXYPOLYMERS

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RADIATION DAMAGE AND PROTECTION OF PLASTIC SCINTILLATORS WITH A BASE OF POLYSTYRENE AND EPOXYPOLYMERS. N.Barashkov[1], F.Markley[2], A.Pla-Dalmau[2], G.Foster[2], M.Rivard[3] 1) University of Texas at Dallas, Richardson, TX 75083-0688,USA; 2) Fermi National Accelerator Laboratory, Batavia, IL 60510, USA; 3) University of Michigan, Ann Arbor, MI 48109-1200, USA

Plastic scintillators are often used in dosimetry of ionising radiation at high and low levels. Some experiments in high energy physics such as modern colliding beam experiments subject scintillators to higher doses of radiation than ever before. Plasticizers (diffusion enhancers) have been used to provide the radiation protection of scintillators. We have shown that the introduction of plasticizers in common polymer matrix has at least two disadvantages. These consist of making the scintillators too soft and increasing the probability of diffusion of the scintillation dyes out of the matrix. Two new approaches for the preparation of scintillators with improved radiation stability have been investigated. The first approach consists of preparing crosslinked copolymers of styrene or epoxypolymers which can be used as the matrix with a large concentration of plasticizers. The second approach eliminates any luminophore migration by covalent bonding of the scintillation dyes into the polystyrene and epoxypolymers. We investigated the dependence of the radioluminescence intensity of these new scintillators on the concentration of the macroradicals formed in the process of gamma-irradiation.

FIBER-OPTIC-COUPLED DOSIMETER FOR REMOTE OPTICAL SENSING OF RADIATION

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Remote sensing technologies for the detection and measurement of ionizing radiation exposure are of current interest for applications such as patient dose verification during radiotherapy and the monitoring of environmental contaminants. Fiberoptic-based sensing is attractive due to the advantages of small size, low cost, long life and freedom from electromagnetic interference. Several fiberoptic-based radiation sensing systems have been described that utilize radiation induced changes in the optical characteristics of the fiber such as reduced transmission as a result of darkening of the glass [1], optical phase shifts due to heating [2], or changes in the birefringence of a polarization-maintaining fiber [3]. The measurement of radiation induced darkening is limited in both sensitivity and dynamic range and requires long fiber lengths. Phase shift measurements require the use of single-mode lasers, phase sensitive interferometric detection, long fiber lengths and complex signal processing techniques. Alternatively, thermoluminescent (TL) phosphor powders have been coated onto fiberoptic cables and remote dosimetry measurements performed using traditional laser heating techniques [4]. The sensitivity is limited by the requirement for a very thin layer of phosphor material, due to problems associated with light scattering and efficient heating by thermal diffusion.

In this paper we report the development of an all-optical, fiber-optic-coupled, thermoluminescence dosimeter for remote radiation sensing that offers significant advantages compared to previous technologies. We recently reported the development of an optically transparent, TL glass material having exceptionally good characteristics for traditional dosimetry applications [5]. We also reported a modified TL glass incorporating a rare earth ion dopant in order to absorb light from a semiconductor laser and utilize the absorbed light energy to internally heat the glass and release the trapped electrons [6]. This TL glass contains nanocrystalline ZnS particles and Cu^{1+} ions in a Vycor glass matrix, to which approximately 0.6%, by weight, of Nd^{3+} ion is added. The Nd^{3+} ion is well suited for this application for several reasons: 1) it absorbs strongly at 807 nm (GaAlAs laser wavelength); 2) it transforms a significant amount of the absorbed light energy to heat that is deposited in the glass matrix to stimulate TL; and 3) the absorption of the Nd^{3+} ions does not significantly overlap the thermoluminescence emission wavelengths and therefore does not substantially attenuate the signal. For this fiber-optic-coupled dosimeter work, Nd^{3+}-doped TL glass rods were hand-drawn to ~200 micron diameter. These TL fiber dosimeters (1 cm long) were fusion spliced to commercial, multimode optical fibers (2 m long, 200 μm core diameter).

A schematic of the fiberoptic-coupled, laser-heated thermoluminescence dosimeter is shown in Fig. 1. The 807 nm output of a 1 watt diode laser array was collimated using a microscope objective. The polarized 807 nm light was transmitted through a Glan prism and imaged with a second microscope objective onto the input face of the multimode fiber. The laser light stimulated thermoluminescence from the TL glass fiber dosimeter previously exposed to ionizing radiation from a ^{60}Co source. A fraction of the 500 nm, laser stimulated

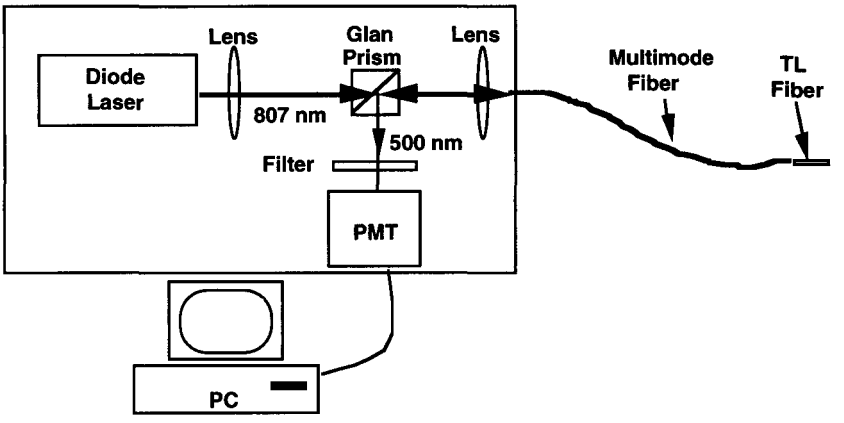

Figure 1: Schematic of the laser-heated, fiber-optic-coupled, remote sensing, thermoluminescence dosimeter.

TL light emission was trapped by total internal reflection and was directed back through the multimode fiber, collimated by the microscope objective and reflected by the Glan prism into a photomultiplier tube. A color glass filter augmented the Glan polarizer in selectively blocking diode laser light while passing the TL signal. The PMT was cooled and the signal was detected using photon counting. The signal collection efficiency could be increased by a factor of two if a dichroic beamsplitter were used in place of the Glan prism.

A laser heated TL signal obtained from a fiber dosimeter exposed to a ^{60}Co γ-ray dose of 2 krad is presented in Fig. 2. The peak of the signal occurs soon after turning the laser on (at t~3 s in this trace), followed by a decay of several seconds. This behavior is in marked contrast to previous studies of laser heated dosimetry [4] in which a TL glow curve, similar to that observed using traditional contact heating methods, is observed. All prior laser heating methods rapidly increase the bulk temperature of the dosimeter to high levels. The laser heating observed in the present work is also quite rapid, however, the mechanism of the laser heating is radically different from that of previous methods and the bulk temperature is not significantly increased. The explanation of this unique behavior lies in the microscopic nature of the localized laser heating and has been described in detail [6]. The key feature of this dosimeter is the use of the optically transparent TL glass that permits efficient local heating but minimal bulk heating of the fiber sensor.

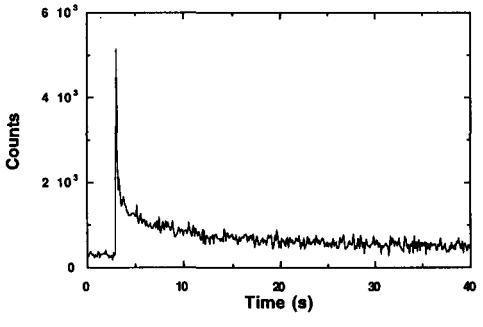

Figure 2: Laser-heated thermoluminescence signal obtained from fiber-optic-coupled TL dosimeter dosed to 2 krad and read using 1 W cw power at 807 nm.

The ^{60}Co γ-ray dose dependence of the fiber dosimeter is shown in Fig. 3. The plot displays the integrated signal area as a function of the dose. Both the integrated areas and the peak values of the TL signals yield identical dose

response curves. The dose dependence is linear up to a value of approximately 500 rad. The dose range of interest for medical diagnostic and therapy applications, 10 to 1000 rad, can be accurately measured with the prototype dosimeter. The optical transparency of the glass permits increasing the sensitivity of the dosimeter simply by increasing the diameter of the fiber sensor. Laser induced TL originating from any point within the fiber can be efficiently collected and utilized, a feature not possible using traditional, highly scattering TL materials. The sensitivity can also be increased by optimizing the concentration of the Nd^{3+} dopant ions and improving the optics and detection apparatus. We estimate that the ultimate sensitivity of our remote dosimeter will be less than 1 mrad.

In summary, we have developed an all-optical, fiberoptic-coupled radiation dosimeter that has excellent performance characteristics for remote radiation sensing applications. The enabling feature of the dosimeter is an optically transparent, Nd^{3+}-doped TL glass that exhibits efficient localized laser heating. The dosimeter utilizes relatively low cost components such as a solid state diode laser, commercial multimode optical fiber, simple optical elements and direct photodetection. Dose information is correlated with either the signal amplitude or the total integrated count, and minimal signal processing is required. Remote sensing applications for this fiberoptic-coupled dosimeter include medical applications, such as near-real-time, in vivo, dose monitoring in patients undergoing radiation therapy, and environmental applications, such as monitoring for leakage of nuclear wastes around contaminated sites.

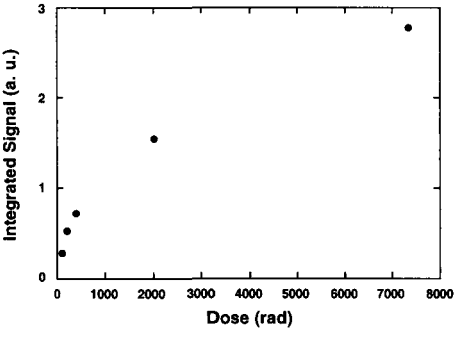

Figure 3: Dose response of the laser-heated, fiber-optic-coupled dosimeter.

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PAPER TITLE The Investigation of LiF:Mg,Cu,P Thermoluminescent Detectors; The Dose Response to Gamma Radiation

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ABSTRACT (See instructions overleaf)

The dose response of LiF:Mg,Cu,P TLDs has been investigated for gamma radiation in the dose area between 0.001 to 100.0 Gy. It has been determined that the dose response for this dose range is non-linear. This effect can be explained by the following reasons:

A) The dependence between the resolving time of the photomultiplier tube of the TLD reader and the photon fluence rate from the TLD during the heating, acquisition cycle.

B) The relative dose response of the LiF:Mg,Cu,P TLD is a function of the number of free electron centers with in the detector. Our data demonstrates a very close approximation with our theoretical model.

COMPARATIVE STUDY OF COMPUTERISED GLOW CURVE ANALYSIS IN ROUTINE DOSIMETRY

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A computer program for the glow curve deconvolution analysis in radiation dosimetry, mainly personal and environmental dosimetry carried out with LiF: Mg,Ti and CaF_2:Dy has been studied. The program is based on knowledge of the physical parameters of the glow peaks determined using two different peak shapes. The first peak shape is calculated by the first order kinetic and the second one by a gaussian technique.

Three ways of determining these parameters have been investigated. They consist of marking the initial parameters on the curve, initiating the deconvolution process from choosen parameters and calculating the parameters automatically in the global program. The intercomparaison of the capacity to reproduce the real curve involving the three options, and their performances in field applications for two processing options (research or production mode), will be presented. The advantages of these methods of glow curve analysis in routine dosimetry on precision and low detection limits compared to the conventional methods will be discussed.

MOLECULAR OXYGEN AS THE RADIATION PROTECTER OF THE PLASTIC SCINTILLATORS

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INTRODUCTION

The yield of the luminescence plastic scintillators decreases on the action of an ionizing radiation. The absorbed dose at which the luminescence yield decreases by 20 % is equal to 15 - 20 kGy for such the most suitable polymeric matrices as polystyrene (PS), polyvinyl xylene (PVX), polyvinyl toluene (PVT) and polymethylmetha crylate (PMMA). The luminescence yield decreasing is caused by the different reasons, for example, the deactivation of the excited states of polymeric matrices by the intermediates that are formed under the scintillator radiolysis, the lowering of the scintillator transparency as a result of new absorbing or scattering events (1). The paper summarizes the results of the study of the formation and reactions the macroradicals in vacuum and presence of oxygen and deterioration of the plastic scintillators under gamma-irradiation.

RESULTS AND DISCUSSION

The dependencies of the stabilized macroradical concentration and light yields vs absorbed dose in vacuum are antiabatic (at 300 K). The same changes of macroradical concentration and light yields vs heating time at 360 K are observed. Thus, the light yeild recover as a result of macroradical decay. The decreasing of the macroradical concentration take place on the postradiation oxydation process. On the contrary the light yeild increases in the presence of an atmospheric oxygen. The macroradical concentration and the light yeild dependencies on the absorbed dose in PS and PVX in presence of dissolved oxygen are an induction period on the curves. The induction period is absent when the dissolved oxygen is removed from the sample (2). These results show that an atmospheric oxygen and dissolved oxygen induces the decay of the macroradicals and promotes the light yield preservation. The molecular oxygen is the "victim" type. The reversible character of the radiation damage makes it evident that the macroradicals are the exited state quenchers. The overlapping of the energetic donor and acceptor levels are necessary for quenching. In the case of PMMA, PS and its substituted, this condition is fulfilled for both dopant containing matrix and macroradicals. The energy transfer from the matrix and primary dopant to the macroradical is allowed. Probably the macroradical quenching of the electron-exited states has an unlike mechanism caused by the spescific structure of macroradicals near the dopants. The dopant locations are the defects in which all radiation processes (such as the localization of the absorbed energy, the formation and reaction macroradicals and so on) proceed. The mechanism of the radiation-chemical processes in the plastic scintillators shows the following ways of the scintillator radiation resistance increasing: it is necessary to promote the oxygen diffusion into bulk of the scintillators by incorporation of low molecular additives or by decreasing the scintillator sizes; the decreasing of the macroradical radiation-chemical yield in polymeric matrices; the increasing of the macroradical dycay rate by the intensifying of the matrix molecular mobility (plasticizing, rise of the temperature and so on).

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PAPER TITLE ESR Dosimetry on Lyoluminescence Materials

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

In radiotherapy solid state dosimetry is the common method for in-vivo dosimetry. The "one shot" character of the lyoluminescence method suggests the search for alternatives, which provide most of the advantages of this technique like tissue equivalence and reliability.

The ESR (electron spin resonance) spectroscopy allows non destructive premeasurement of lyoluminescence probes and hence confirmation of the results of that method.

In this study some organic materials (glucose, dextrose, mannose) were used for both techniques to test their comparability. The dosimetric materials were irradiated with photons (1.2, 6, 10, 15, 25 MeV), electrons (6, 10, 15, 18 MeV) and protons in the dose range of 1 to 20 Gy. ESR measurements were made with a commercially available low cost ESR device developed for identification of irrdiated foodstuffs.

STUDY OF X-RAY PHOSPHORS USED FOR THE VISUALIZATION OF IONIZING RADIATION

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There is considerable interest in Ukraine after Chernobyl accident in dosimetric X-ray luminescent materials which can convert the energy of ionising radiation into the visible emission.

A special request is for X-ray luminophors for high-dose and high-temperature conditions.

The main requirements which X-ray phosphors have to satisfy for the visualization purposes are:

i) high converting efficiency (high intensity of visible emission);

ii) radiation stability (preserving the luminescent properties under the high-dose and high-energy radiation)

The aim of the study is - to test the feasibility of application of X ray phosphors currently used under irradiation of soft X-ray (in X-ray screens in medicine) for high-energy, high-dose and high-temperature conditions.

Materials - solid powders of europium-doped X-ray phosphors.

Techniques - 1) steady-state X-ray luminescence. For luminescence excitation X-ray, γ- and β- radiation in the range of energy from 40 keV to 5 MeV were used.

2) time-resolved laser-induced photoluminescence (λ-excitation 337 nm). On the basis of the luminescence decay measurments the life-times were calculated

The main results obtained in the study (summarized in the table) are the following:

№	Composition of samples	Visible emission, λ nm	life-time, τ	luminescent center
1	BaFCl·Eu	390	6.3 μs	Eu^{2+}
2	$Sr_3(PO_4)_2$·Eu	410	32 ns	Eu^{2+}
3	Y_2O_2S·Eu	585	67.2 ns	Eu^{3+} cluster
4	Y_2O_3·Eu	400	10.2 μs	Eu^{3+}
		630	~4000 μs	

i) all tested phosphors under high-energy excitation revealed the bright visible emission in blue-green and yellow-red region

ii) luminescent properties are preserved after high-dose (300 Gy) irradiation and high-temperature annealing (300 C^0) of samples.

The next important feature of tested phosphors is that in most cases the spectral shape of radioluminescence is practically the same as for laser-induced luminescence.

For the BaFCl·Eu and for $Sr_3(PO_4)_2$·Eu the luminescence consists of mono-band emission and can be attributed for Eu^{2+}.

Spectrum of Y_3O_3·Eu contains two luminescent species - mono-band emitting near 400 nm and the second one with fine-structure centered near 630 nm. The last spectrum consists of f-f transitions characteristic for Eu^{3+}. It must be noticed that these two luminescent species have also considerably different life-times.

The influence of the host solid-matrices on the life-time of the luminescent species will be discussed.

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PAPER TITLE: RADIATION PROTECTION SURVEILLANCE OF OCCUPATIONAL EXPOSURE TO RADON PROGENY

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MAJOR SCIENTIFIC TOPIC NUMBER: 5.1 (see page 7)

ABSTRACT (See instructions overleaf)

A review is given on the surveillance of workplaces exposed to short-lived radon progeny in the Eastern part of Germany. According to the present different legal regulations in Germany only in the Eastern part of Germany workplaces with exposures to short-lived radon progeny are regularly monitored not only in uranium mining and uranium milling and in the closeout of the uranium industry (since 1991), but also in non-uranium mining and other workplaces, such as caves, radon spas and water treatment facilities.

20 years experience in this field shows that a special system for quality assurance is the prerequisite to reliable results for the individual exposure of workers. This system includes the selection of suitable measuring methods and devices, the first and the repetition calibration of the devices and the careful interpretation of the results.

The equipment available in the Federal Office for quality assurance, the measuring methods and devices used at workplaces and the results of surveillance are described. Recommendations for further development are given.

OCCUPATIONAL RADIATION EXPOSURE IN SOME EGYPTIAN PHOSPHATE MINES

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ABSTRACT:

Radiation levels in some working phosphate mines in the Egyptian Eastern Desert have been investigated to estimate the occupational exposure to the workers in those sites. Such results may help in the preparation of the corrective actions as well as the improvement of the safety measures if needed in those working mines. Beta and gamma levels as well as radon gas concentration and its decay products have been measured. Active techniques are employed to fulfill the objectives of measuring radon gas and its daughters.

Some working conditions and environmental parameters such as the working time, type of available ventilation, temperature and humidity have been studied during the period of measurements. The maximum reported values for radon daughter concentration in units of working level are 1.28, in Safaga area south mine, 1.22 in Hamraween area B mine and 0.67 in El-Quser area Youns C mine. The maximum annual dose for the worker in all locations under investigation is about 100 mSv/y which is clearly much higher than the recommended international value. According to the above estimated values the question of ventilation economics in such mines is created. The classification of the miners in conventional mines as radiation workers should also be put into consideration.

INTRODUCTION

The main phosphate mines in Egypt are sited in the Eastern Desert. There are three regions Safaga, Hamraween and El-Quser which are considered as the producing mines. The three regions are separted by about 60 and 20 km from each other. The depth of the mines ranges from 10-50 meters.

It is well known that phosphate rocks contain the trace elements of uranium, thorium and their decay products in equilibrium. All technical processing leads to a high release of long and short half-life radionuclides from uranium mining and milling. (5).

In this study a radiation monitoring programme was carried out in the mining areas, γ and β levels are measured as well as radon gas concentration and its decay products in units of working level.

EXPERIMENTAL

1. Monitoring External Beta and Gamma

An Eberline survey meter model L.B. 1200, was used.

2. Radon Gas Measurement

Alpha scintillation cells was used. Scintillations from Lucas cell were counted using a photomultiplier tube in a light-tight enclosure and a counting system made by EDA of Canada.

3. Air Sampling

Air samples were taken using air sampler having a flow rate of 1-10 1/min.

4. Rodon Daughters Measurement

Air was passed through high-efficiency filter paper (milipore) to trap radon daughters. Filter papers were counted using the same counting system by placing the filter paper on a scintillation tray coated with the same material as the Lucas cell.

5. Calculation of Radon Daughter Concentration

Rolle method was used to calculate radon daughter concentration in units of working level, (2,3).

RESULTS AND DISCUSSION

The results of radon and radon daughters concentrations, effective dose equivalent and annual dose for the workers are shown in table (1) for Safaga, Hamraween and El-Quser areas.

It is clear from table (1) that γ and β levels in the three mines are within the range (0.03-0.25 mR/h) which is higher than the natural background level.

The maximum radon daughters levels in units of WL were found 1.2 in south mine at Safaga, 1.2 in mine B at El-Hamraween and 0.6 in Youns C mine at El-Quser. These levels are higher than the ICRP recommended value for the workers (0.3), also the maximum values of radon gas concentration are found 90 pCi/1 in south mine, pCi/1 in mine B and 108 pCi/1 in Youns C mine. (3)

The maximum values for the annual dose for the workers are found 121 mSv/y in south mine, 116 mSv / y in mine B and 65 mSv/y in Youns C mine. We notice that these levels are higher than the recommended value (20 mSv/y) (4).

It is clear from the results obtained for radon, radon daughters concentrations and gamma exposure rates that most values are higher than the recommended limits and require corrective actions. These higher levels are due to bad ventilation.

The following corrective actions are recommended :

1. An efficient mechanical ventilation system should be establised to decrease the annual dose equivalent to the accepted limit of 20 mSv/y.

2. Job rotation could be used as a helping action to decrease the exposure.

Table (1) : A Summary of Exposure Rates, Effective Dose and Annual Dose in Phosphate Mines.

LOCATION	POINT OF STUDY	GAMMA AND BETA LEVEL mR/h	RADON DECAY PRODUCTS WL	EFFECTIVE DOSE EQUIVALENT uSv/h	ANNUAL DOSE FOR THE WORKERS mSv/y	TEMPERATURE AND HUMIDITY C°, %
SAFAGA	1	0.03	1.283	83.19	121.46	26.85
AREA	2	0.03	1.174	76.38	111.51	26.85
SOUTH	3	0.03	0.959	62.94	91.89	24.80
MINE	4	0.03	0.984	64.5	94.169	24.80
	5	0.03	0.939	61.69	90.067	24.80
	6	0.03	0.887	58.44	58.32	24.80
HAMRAW-	1	0.03	0.451	31.19	45.54	25.72
EEN	2	0.03	0.708	47.25	68.98	26.75
AREA	3	0.03	1.225	79.56	116.16	27.87
MINE B	4	0.03	1.009	66.06	96.45	27.87
	5	0.03	0.817	54.06	78.93	27.85
	6	0.03	0.624	42	61.32	26.80
EL-QUSER	1	0.25	0.034	4.625	6.75	26.82
AREA	2	0.25	0.095	8.44	12.32	26.82
YOUNS	3	0.25	0.157	12.31	17.97	26.82
MINE C	4	0.25	0.261	18.81	27.46	27.85
	5	0.03	0.476	32.75	47.46	27.85
	6	0.03	0.671	44.94	65.61	28.87

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PAPER TITLE Occupational And Environmental Radiation Monitoring Programmes at an Australian Underground Copper and Uranium Mine

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MAJOR SCIENTIFIC TOPIC NUMBER 5.1 ..... (see page 7)

ABSTRACT (See instructions overleaf)

WMC (Olympic Dam Corporation) is a major Australian producer of copper and uranium and consists of an underground mine and multifunction metallurgical plant. The facilitites have been in production for 8 years and the company is currently examining a major expansion. Occupational and environmental radiation protection is considered to be a critical management function and is formalised through a radiation safety management system. The performance of the system is audited by a group of radiation protection professionals via an established radiation monitoring programme. The programme successfully quantifies the occupational and environmental radiation exposure levels and demonstrates the effectiveness of the management system. The results of the programmes indicate that occupational and environmental exposures are low and well controlled with occupational levels averaging at 20 percent of the 20 mSv limit and the most exposed public group being less than 10 percent of the member of public limit of 1 mSv.

The monitoring programme is only one part of the overall radiation safety management system and this paper will outline the monitoring programme and exposure and dose assessment methodologies from a practical perspective.

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PAPER TITLE Comparison Of Occupational Radon Daughter Dose Assessment Strategies

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MAJOR SCIENTIFIC TOPIC NUMBER 3.1 ..... (see page 7)

ABSTRACT (See instructions overleaf)

The assessment of radon daughter exposure and dose to underground miners at WMC (Olympic Dam Corporation) copper and uranium mine is required to ensure that doses remain low and well controlled. The current methodology for exposure and dose assessment involves collecting information from employee time cards and combining this with the results of the routine workplace monitoring programme. This method, while very comprehensive, is labour intensive. Consequently, the approach to radon daughter dose assessment underwent a formal review to determine the most appropriate method. The review primarily examined various types of personal radon daughter dosimetry as options to the manual method and found that of the dosimetry examined, none was appropriate to conditions at WMC (Olympic Dam Corporation).

This paper will outline the review process and the results of the review.

RADIATION PROTECTION IMPLICATIONS FOR THE NORTH RANGER MINE

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INTRODUCTION

The North Ranger ore deposit is located approximately 20 kilometres north of the existing Ranger mine and mill and 250 kilometres east of Darwin in the Northern Territory of Australia.

The proposed mining operation will use "open stoping" as its preferred method for recovering the ore; subsequently the ore will be transported to the existing Ranger mill for processing.

The Radiological assessment of an underground uranium mine is complex; it needs to assess the radiation exposure from three different pathways, namely gamma ray, radon progeny and radioactive dust exposure. Of the three pathways the assessment of radon progeny exposure is by far the most complex.

At the North Ranger mine radon progeny exposure has been minimised by good ventilation practice and is the least significant of the three exposure pathways.

Gamma radiation is the most significant exposure pathway and this is due to the grade of ore being mined and the mining method.

There are two major phases to the mine development (Case 1 and Case 2 - 600 000 and 900 000 tonnes per annum respectively) and therefore two distinct radiological assessments have been made in line with the mine ventilation and production schedule.

RADON PROGENY MODEL

In ventilating a uranium mine the knowledge of the distribution of radon progeny throughout the ventilation system is essential. The major source of radon gas into the air is from the walls, floor and roof as the air travels down the drives. In the ventilation model the width of the ore intersection for each airway is entered as well as the estimated uranium ore grade. This was achieved by overlaying the ventilation design plans and sections with the geological block model and where any development intersected ore greater than 0.05 % U_3O_8, ore grades and widths were calculated. An emanation rate of 49 Bq $m^{-2}s^{-1}$ per % U_3O_8 was used (2).

The model calculates the average radon emanation rate, the air transit time, and the average radon progeny concentration for each branch, finally the average radon progeny concentration was calculated for each worker category. Full details of the radon progeny model can be found in (3).

GAMMA RAY DOSE RATE MODEL

The theoretical calculation of gamma exposure rates for a rectangular tunnel was used. The various elements such as attenuation of gamma ray by shields and build-up of gamma rays due to a broad beam source (ore) and thick shields (waste rock on the floor and shotcrete on walls and roof) were considered.

The theory shows that geometry is critical to the exposure rates calculated. However, the theory can not take into account the non-uniform nature of the ore in a tunnel and the shielding provided by machinery (4). It appears that theoretical gamma dose rate models may overestimate the actual dose rate by a factor of about two.

This study assesses the theoretical model "Gamma Model A" and a practical model "Gamma Model B" that uses a dose rate of 50 μSv h^{-1} / % U_3O_8 (4).

Gamma Model A uses theoretical calculations (5), it derives the exposure rate per unit solid angle in an orebody and gives equations for calculating the solid angle subtended by various surface slopes. The resulting theoretical gamma dose rate was 109 μSv h^{-1} / % U_3O_8.

DUST MODEL

The literature survey by Leach (6) showed that for 0.1 per cent uranium ore grade, the typical long-lived alpha activity is, at worst, 0.05 Bq m^{-3}. Similar results have emerged at Olympic Dam, a modern copper/gold/uranium underground mine.

To calculate the effective dose due to the inhalation of uranium ore dust, an average long-lived alpha dust concentration of 0.1 Bq m^{-3} was assumed. This figure was extrapolated from data from Olympic Dam. It is recognised that the actual dust concentrations will vary greatly depending on locality and activity.

WORKPLACE OCCUPANCY

For the purpose of calculating radiation doses to workers it has been found useful to group various workers into categories. This process assists in the administration of radiation protection practices and dose assessments. Five major categories have been determined with these categories being further broken down into various work functions. The locations which the various worker categories occupied during a shift are divided into three specific work place areas, namely high, medium and low grade areas (0.45 %, 0.13 % and 0.02 % U_3O_8 respectively).

DOSE ASSESSMENT

The dose assessment model was developed to estimate likely doses that may be received by the various worker categories.

For each worker category the percentage of the effective hours worked that was subject to shielding were estimated together with a shielding factor. Three forms of shielding were included in the model: equipment shielding, work place shielding due to waste rock on the floor and shotcrete on the walls and roof of ore drives, and, thirdly, the attenuation of dust provided by the air-conditioning on the various types of mobile equipment.

The dose assessment model uses the following shielding factors:

Equipment shielding for gamma	-	50 %
Equipment shielding for dust	-	50 %
Workplace shielding for gamma	-	57 %

The model gives individual work function dose estimates as well as average dose estimates for the five other category groupings.

Table 1 gives the estimated doses for the various work categories using the Gamma Models A and B for Case 1 and Case 2 and the various shielding combinations.

RESULTS AND CONCLUSIONS

Table 1 shows that workers involved in development and supervision receive the highest doses. It is likely that the practice of shielding afforded to the development crews will be less than that for production workers because of the work they perform. In the same way, supervisors will receive higher doses as they experience less equipment shielding. However, when shielding is applied to the high grade cross-cuts to attenuate gamma rays it is possible to reduce the predicted radiation doses to all work categories.

For development and production drillers in particular, advances in equipment technology permitting remote or automatic drill operation may be feasible in the future and would further reduce radiation levels.

If the ventilation design principles are adhered to then the exposure to radon progeny will be low.

A breakdown of the three components of the dose received by development and production categories can be seen in Table 2.

By designing the ventilation system to provide fresh air to each work place as quickly as possible and then exhaust the air, radon progeny exposure is minor.

The gamma radiation is by far the most significant exposure pathway and this is primarily due to the grade of the ore and the mining method. It is therefore necessary to attenuate the gamma rays by use of shotcrete on the walls and roof and waste rock on the floor in the high grade cross-cuts.

Proper management and minimisation of time spent in high grade cross-cuts is critical for the overall control of radiation doses.

With a properly designed ventilation system, shielding, and management control of operator exposure, it is feasible to mine North Ranger in a regime of 20 mSv per annum radiation limit.

Table 1. Radiation Dose Summary.

Main Work Grouping	Gamma Model A					
	No Shielding		Equipment Shielding Only		Equipment and Work Place Shielding	
	Case 1 (mSv)	Case 2 (mSv)	Case 1 (mSv)	Case 2 (mSv)	Case 1 (mSv)	Case 2 (mSv)
Development	21.8	23.3	17.2	18.8	14.3	15.8
Production	20.3	21.5	15.2	16.3	11.2	12.3
Fill & Services	14.5	16.1	13.1	14.8	10.8	12.5
Supervisors	21.7	23.2	19.7	21.1	14.6	16.1
Foreman/Technical	10.9	11.6	10.3	11.1	8.6	9.3
	Gamma Model B					
	No Shielding		Equipment Shielding Only		Equipment and Work Place Shielding	
Main Work Grouping	Case 1 (mSv)	Case 2 (mSv)	Case 1 (mSv)	Case 2 (mSv)	Case 1 (mSv)	Case 2 (mSv)
Development	13.1	14.7	10.4	11.9	9	10.6
Production	12.7	13.8	9.5	10.7	7.7	8.9
Fill & Services	9.7	11.4	8.8	10.5	7.7	9.4
Supervisors	13.2	14.6	12	13.4	9.7	11.1
Foreman/Technical	6.6	7.3	6.3	7	5.5	6.2

Table 2. Gamma Model A Doses Pathways

	Development (mSv)			Production (mSv)		
	Dust	Radon	Gamma	Dust	Radon	Gamma
No Shielding						
Case 1	4.5	1	16.2	4.5	1.6	14.2
Case 2	4.5	2.6	16.2	4.5	2.7	14.2
Equipment Shielding Only						
Case 1	3.5	1	12.7	3.1	1.6	10.5
Case 2	3.5	2.6	12.7	3.1	2.7	10.5
Equipment and Work Place Shielding						
Case 1	3.5	1	9.8	3.1	1.6	6.5
Case 2	3.5	2.6	9.8	3.1	2.7	6.5

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April 14-19,1996
Vienna, Austria

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PAPER TITLE
SUBJECTIVES FACTORS IN THE OCCUPATIONAL EXPOSURE MONITORING (URANIUM MINES IN ROMANIA)

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MAJOR SCIENTIFIC TOPIC NUMBER: 5.1 (8.8)

In Romania, untill 1989, state economical and political reasons requiered to give up using individually dosemeters, in the uranium mines.

It atteined that object by an intensive disinformation campaign based on the extreme poverty of workers and their desir to obtein "unpropers" benefits - special rights for the exposed people (almost vital, for anyone). The slogan was that radiations danger for the human health is only "fairy tales". Ideas were:

1. If EDE admitted is reached, it significates that miners must nomore work in exposed places so, **they will nomore have the special benefits.**

2. If EDE admitted is not reached, it significates **there are no reasons to give them these special benefits.**
So, the individually dosemeters were completely discredited and nobody regretted them!

The Special Service (S.A.D.) created to monitorise the occupational exposure in any working place had insufficient earmarked money and its members were forced to don't register unpleasant data.

In present, in other forms but partly from the same reasons, the situation goes on (individually dosemeters are still not used).

A sustainable solution is to aware the workers. Public NGOs could assume an important role but they are not agreed (it attached to them the image of "public ennemy no. 1" and their members working in nuclear lostes their working places). In Romania, almost everything about radiations is still "top secret" so, NGOs interrested must activate in other domains, for the moment, and approaching to radioactivity with maximum of prudence.

RADIATION DOSE CONTROL IN THE MINING OF HIGH GRADE URANIUM ORES

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Introduction

The control of radiation doses received by uranium miners is an unusually complex procedure, as three separate components of their total effective dose may be significant and may have to be evaluated separately. Apart from external and internal doses evaluated in the usual way, it is also necessary to evaluate the inhalation dose from radon progeny separately. Although this essentially forms part of the internal dose received, it is not evaluated in the conventional way since the associated dose equivalent must be derived from conversion factors based on epidemiological studies, instead of by the usual approach of calculating the dose to tissue from the inhaled activity and multiplying this by a recognised conversion factor to derive a whole body effective dose. Historically the traditional unit used for monitoring the concentration of radon progeny in a workplace is the Working Level (WL), this is now defined as a concentration such that the potential alpha emission from all the short lived progeny present in the sample will total 1.3×10^8 MeV per m^3. The corresponding unit of exposure is the Working Level Month (WLM) and is the exposure that would be received by a reference man working in such an atmosphere for a standard working month lasting 170 hours. Unfortunately the relationship between exposures, measured in WLM, and the conventional radiation dose to the target tissues is complex and calculated values depend greatly upon the assumptions made in the lung model that must be used. Risks are therefore still controlled by limiting exposures in WLM on the basis of epidemiological studies of lung cancer incidence among miners employed at a time when the magnitude of the risk was not fully appreciated, and cancer incidence was high enough to permit reasonably accurate risk estimates to be derived directly from exposures in WLM.

The Mining of High Grade Ores

Where the grade of the ores being mined is relatively low, the risk to underground workers is dominated by the inhalation of radon progeny; and frequently in the past the contributions to the worker's total radiation dose from gamma radiation, or the inhalation of ore dust containing uranium and its long lived decay products, was neglected in comparison. This is no longer acceptable, both because the majority of ores currently being mined are of far higher grade than was usual in the past, and because radiation dose limits for occupationally exposed workers have been reduced significantly. In other industries the introduction of lower dose limits has generally not led to serious difficulties, revised working procedures, coupled with better shielding, have ensured that the new standards can still be achieved. This can be much more difficult in the case of the mining of high grade uranium ores. Conventional mining techniques can result in the worker being surrounded by ore, this leads to an omnidirectional gamma flux from which it may be very difficult or impossible to shield the worker effectively. Radon concentrations are traditionally reduced to levels that are acceptable by ventilating the workings with fresh air brought in from outside. If the radon emission rate is too high this may require such massive ventilation systems that all operations have to be carried out in near gale force winds. Such high ventilation rates lead to very exhausting working conditions and will, in turn, also increase the resuspension of dust in the workings. When the grade of the ore is very high, this can also lead to a large potential dose to the miners from long lived radioactive dust inhalation. Ultimately, in the case of very rich ore deposits these problems become so severe that conventional mining methods cannot be attempted . Consequently in Saskatchewan, where several such very high grade deposits are currently under development, the approach has been for the development of new technologies in which the miners are excluded from the workface and operate from higher or lower stopes lying outside the main ore body, with all ore being extracted by means of remote boring operations.

Such automated mining methods need to be planned in detail right from the moment when the development of a new mine is first considered, they cannot usually be economically introduced into an existing mine. In such mines, meeting the more stringent standards required to comply with ICRP 60 recommendations may have to be accomplished without the introduction of fully automated mining methods. The provincial Occupational Health and Safety Branch has therefore been investigating the magnitude of the problems likely to be encountered, and the changes in working procedures which may be possible and may help in alleviating these problems. Since each mine is fundamentally different there is not a large body of existing data that can be accessed, and it was found to be necessary to conduct a series of studies in some of the existing mines which were designed to determine the relative importance of the three different components of the miner's dose for groups of workers in different categories.

Contributions to the Miner's Effective Dose

The whole body dose limits for occupationally exposed workers recommended in ICRP 60 are 50 mSv effective dose in any one year with an average of not more than 20 mSv per year over any five year period. In the case of uranium miners, compliance with these recommendations requires that for both the one and the five year period the external gamma ray dose received by the worker, expressed as a fraction of the one or five year limit, added to the inhalation dose from long lived radioactive dust, expressed as a fraction of the ALI (or five times the ALI as appropriate), and to the separate inhalation doses from radon and thoron progeny expressed as fractions of the corresponding one or five year limits, does not exceed unity. Clearly maintaining records designed to provide both one and five year summation doses on an ongoing basis for every miner places complex requirements on the mine operator. Monitoring, recording and reporting all these components of the miners total dose also involves a number of very complex considerations, including the relative importance to be attached to personal and workplace monitoring. The recommended procedures for meeting these requirements are still being extensively studied and no final decisions have yet been made as to the procedures the mine operator will be expected to adopt, although it is recognised that the relative importance of the various contributions to the total radiation dose experienced will differ considerably between one mine and another. This will prevent agreed universal dosimetric protocols being developed and means that the procedures adopted by each mine will have to be agreed between the appropriate regulatory agency and the mine operator at an early stage. To enable realistic requirements to be formulated it will be necessary for all parties be very clear as to the relative importance of each contribution to the workers dose in each individual mine.

The only type of personal dosemeter which is currently available and which has the capability of being used to determine all these contributions to the workers total dose is the CEA track etch monitor which has been further developed in Canada and is supplied here through the Canadian Institute for Radiation Safety. Essentially this monitor employs an active filtration technique, it is worn on the workers belt and samples the ambient air in which he works. Alpha particles from the radon progeny collected on the filter paper pass through a mechanical spectrometer and impact on a conventional plastic foil where their tracks are displayed following etching. The alpha tracks in each energy group enable the radon and thoron progeny present to be individually identified. Subsequently the residual activity on the filter paper can be used to evaluate the long lived alpha emitting dust that was collected. If the external gamma dose is not being monitored by other methods, this can also be determined by incorporating a TLD chip into the dosemeter system. Personal dosemeters of this type have been in routine use in some Saskatchewan uranium mines for the evaluation of radon progeny dose for more than a decade, but until recently no attempt has been made to use them to determine the internal dose received from the inhalation of long lived radioactive dust. In 1991 the Saskatchewan Occupational Health and Safety Branch, in conjunction with the Canadian Institute for Radiation Safety, initiated a research project designed to determine what proportion of the dose received by workers in different categories was associated with each of the contributing factors that have been discussed above, and also what problems might be expected in attempting to meet the more stringent dose limits of ICRP 60 in some of the operating mines in Saskatchewan. Some of the results of this study can be seen on the two figures below which show the distribution of effective doses among different categories in the workforce both as they are under the present dose limits and as they would be under the new dose limits recommended in ICRP 60.

Conclusions from this study

It is important to recognise that evaluating the risk from the inhalation of uranium dust involves consideration of chemical toxicity as well as radiological dose, and that both of these quantities will be strongly influenced by the solubility of the uranium in the dust. In this respect the dust inhalation risk for millworkers, which arises primarily from processed yellowcake, will be quite different to that for underground miners. Neither can be directly calculated in terms of the listed ALI without consideration of the solubility of the dust compared with the solubility of the compounds for which the ALI was determined. These considerations may warrant a re-examination of the absolute value of the dust component of the workers dose used in preparing the following figures, but nevertheless the data presented here is not generally available from other sources and provides a great deal of useful information about what are likely to be the principle problem areas when ICRP 60 limits are first implemented in existing Saskatchewan uranium mines. The most important conclusions to be drawn from this study are that mill workers already receive doses well below the intended new dose limits, and that this also applies to several other categories of the total workforce. It is only in the case of underground miners extracting high grade ore by largely traditional methods that there appears to be any significant likelihood of individual annual effective doses exceeding 20 mSv. With high grade ores, sufficient mill stock for an extended period of milling can be extracted in a relatively short time and these underground mines are therefore generally operated on a seasonal basis. This has led to a situation where the existing practice in Saskatchewan is for such mining to be carried out under contract and not by permanent members of the mine workforce. With careful monitoring, it would then be easy for the contractor responsible for the mining operation to divert members of the workforce who were approaching their acceptable dose limit into a non-uranium mine where no significant radiation dose was encountered. This makes it unlikely that there would be any intrinsic difficulty in implementing ICRP 60 in such mines. Whether it would be regarded by the regulatory agencies as acceptable to continue mining in working conditions where the permissible annual dose limit might well be approached after significantly less than a full years work, is a different issue. Ultimately ALARA and optimisation are best satisfied by operating mines under conditions where there is a maximum of product per unit of collective dose received by the workforce. With high grade ores, large volumes of yellowcake can be produced by a relatively small workforce; and, even though individual doses may be higher, this requirement is therefore much more likely to be satisfied than when mining lower grade ores.

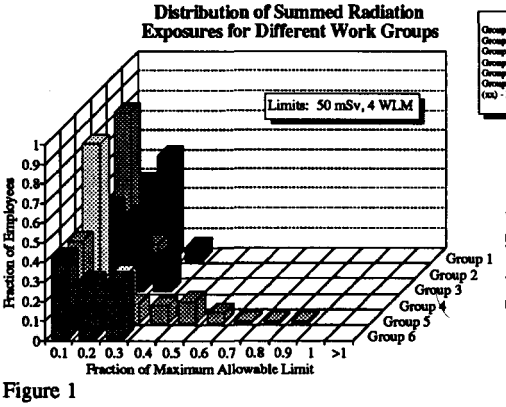

Figure 1

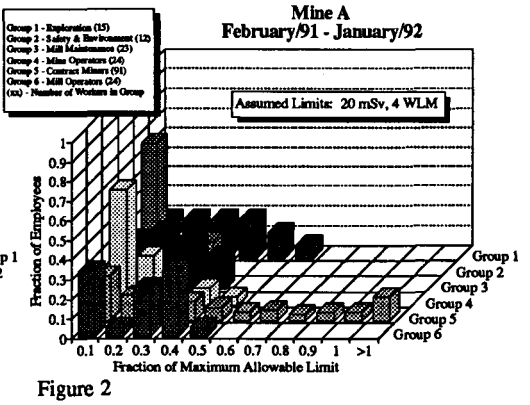

Figure 2

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SEARCHING FOR A LOST RADIOACTIVE SOURCE IN A MINING-MILLING FACILITY

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THE INCIDENT

The mining-milling facility in question has a radioactive level gauge installed at the beginning of its conveyor-belt system, used to turn it on when the run-of-mine ore reaches the chemical plant. The gauge was not in use when the incident occurred (radioactive source installed, detector and associated electronics not installed), the conveyor-belt being controlled manually.

During a routine inspection, a maintenance worker found the inner part of the gauge's lead shielding on the floor. Since he knew that that equipment was "dangerous", he called a technician supposed to know how to deal with it. The technician monitored the fallen piece, found no activity and concluded, erroneously, that the radioactive source was inside the shielding (actually, his detector was not working). The fallen part was put in its place. All this happened in a Saturday, October 1st, 1994, beginning of a long weekend, so the occurrence was only reported to the responsible for the radiation protection in the following Tuesday.

THE SOURCE

The lost source was a 185 MBq (5 mCi) Cs-137 source, of unknown making. A bibliographic search led us to believe that the radioactive material was prepared as a vitrified pellet.

AN ANALYSIS OF THE CAUSES OF THE OCCURRENCE

The gauge is installed on an equipment subjected to a high level of vibration. Since no preventive maintenance was routinely made, the vibration destroyed a safety pin which impeded the inner part of the shield to be removed. This was the cause of the mechanical failure.

Notwithstanding, it should be concluded that the incident, the loss of control over a radioactive source, was caused primarily by the mismanagement of the radiation protection matters in the facility without trying to be exhaustive, we can point: lack of preventive maintenance, source installed without motive since the detector was not installed; incompletely trained technician; lack of a routine to check the functioning of the sole radiation monitor of the facility; responsible for radiation protection with other more pressing activities.

THE SEARCH FOR THE SOURCE

On October, 18th, 1994 the engineer in charge of radiation protection of the facility, feeling unsecured with the situation, contacted our Center, asking for help, 17 days after the probable loss of the source.

Our emergency group, using scintillation detectors, searched for the source all over the facility, as well as the houses of all employees with even a remote possibility of inadvertently having take the source home. We found nothing.

During this unfruitful search for the source we noticed that the ore that have fallen under the conveyor-belt

was frequently swept and introduced again in the process. Reckoning that there was a good probability of the source being also processed we started looking for activity in ore samples collected for process control.

The samples were counted in a HPGE detector, in a low background shielding, for periods of 24 hours each. We found clear peaks of Cs-137 in the samples collected at points 1, 2 and 3 (Figure 1).

The first activity was detected at point 1, on October, 14th, that is, 14 days after the date on which we believed the source was lost. This delay should be attributed to the big ore storage bin that exists between the conveyor-belt and the input to the flotation cell bank. At the same day the activity was detected at the output of the flotation cells (overflow, point 2), which is understandable if we consider the low inventory of material in the flotation cell bank.

On the other hand, it took at least 2 more days for the activity to appear at point 3, the output of the concentrate stock tanks, were it appeared extremely diluted. This time leg and dilution can be easily explained by the volumes of the mill, thichener and concentrate stock tanks.

THE FATE OF THE CONTAMINED ORE

With the sparse data we get, the best conclusion to which we could arrive is that a good part of the activity went to backfill the underground mine and to the tailings pond. Part of it, however, entered the process and, probably, reached the end product. In this product, we were not able to find any activity, at least in the limits of sensitivity of portable scintillation detectors.

It should be noted, too, that the dilution of the radioactive material was very high, leading to contamination levels below regulatory limits (1,2).

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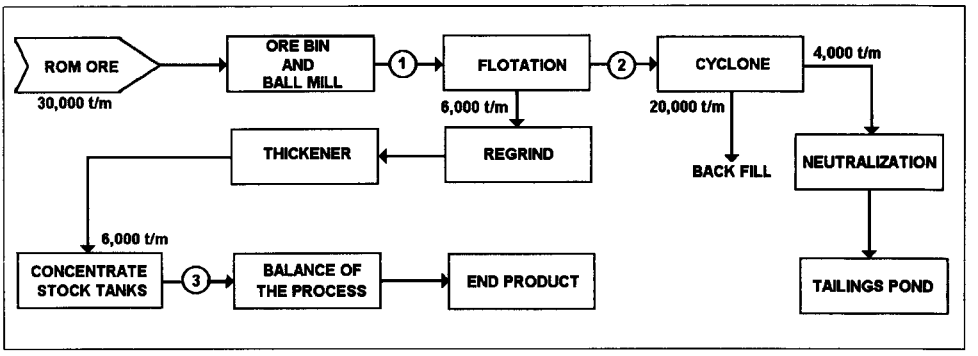

Figure 1 - Process flowsheet

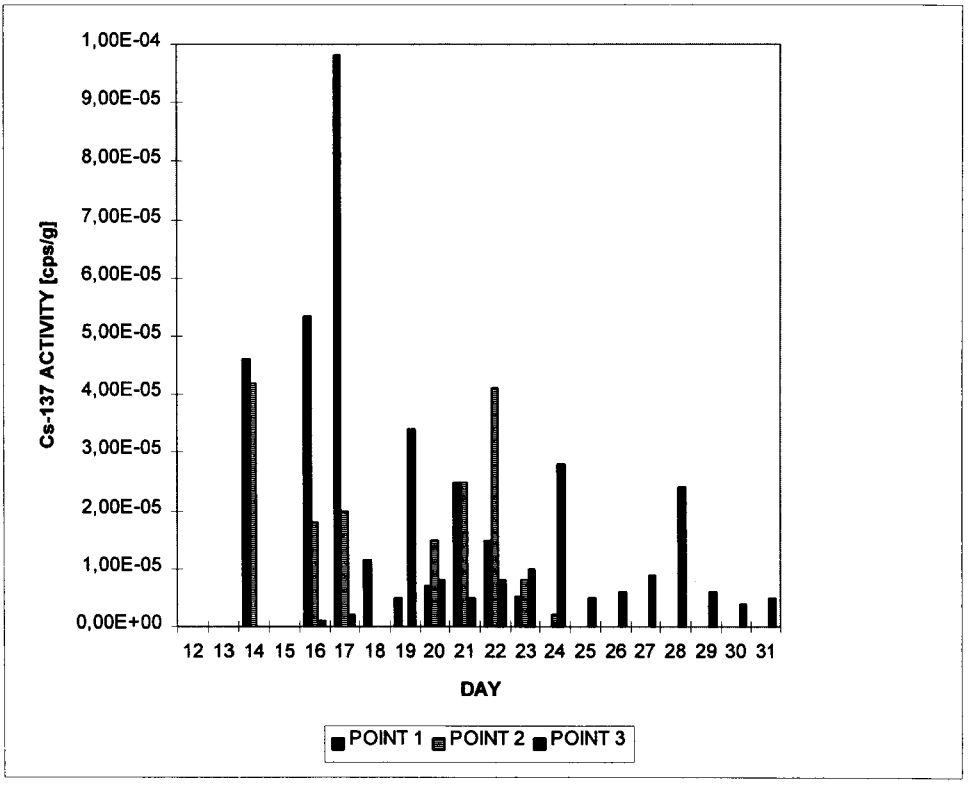

Figure 2 - Actitivity detected in the ore samples

ANALYSIS OF DIFFERENT MEASURING METHODS APPLIED IN EVALUATION OF INDIVIDUAL EXPOSURE OF MINERS TO RADON DAUGHTERS.

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INTRODUCTION

Various measuring techniques are used for the evaluation of individual exposure of miners to radon daughters in Polish mines. Among the most widespread techniques are: passive dosimeters, mining radiometers and ALFA-31 detection unit to passive Barbara 3A dustmeters. The individual, active dosimeter is also prepared to be used.

MEASURING METHODS INCLUDED IN THE COMPARISON

The Mining Radiometer

The mining radiometer (RGR) is a portable device destined to the measurement of momentary concentrations of alpha potential energy of radon daughter products in the air. The radiometer can work in the automatic cycle, acc. to Markov's method [1], or the manual cycle, acc. to an optional method. The measurement is based on the filtration of a particular portion of air, containing radioactive aerosols, through the filter. A number of alpha particles emitted from the filter is counted by means of the semiconducting detector. The usual time of filtration is 5 min. The flow of air is 2 l/min.

The ALFA - 31 detection unit to Barbara 3A dustmeters

The ALFA - 31 detection unit to Barbara 3A dustmeters is used to the periodic measurement of concentration of radioactive aerosols in the mine air. The measurement is based on pumping over of air portions through the filter. The concentration of radioactive aerosols is calculated by means of thermoluminescent detectors $CaSO_4$: Dy. The time of air filtration is about 8 h [2].

The Passive Dosimeter

The passive dosimeter is used to the measurement of exposure to radon daughters or the periodic concentration of these daughters in the air. The track detector of alpha radiation, type Kodak LR 115 foil, is used in the dosimeter [3]. Chemical etching of tracks detectors enables the analysis of tracks' density on their surface, on which basis it is possible to establish the exposure or the periodic concentration of radon daughters.

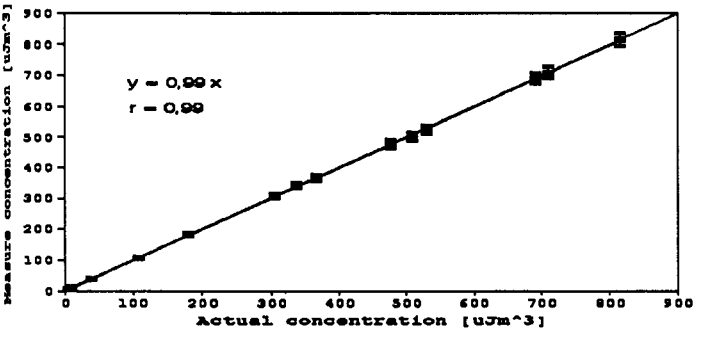
The Active Dosimeter

The active dosimeter is applied to measure the exposure to radon daughters or the periodic concentration of these daughters in the air. It comprises a measuring head joined to a micropump by means of a flexible tubing. The air is sucked in by the micropump and radioactive aerosols deposit on the filter. Alpha particles emitted from the filter are registered by the track detector (type Kodak LR-115) placed in the head.

THE COMMENSURABILITY ANALYSIS OF MEASURING METHODS

Research of commensurability of measuring methods was carried out in radon chamber [4]. The measurement of radiational conditions in the chamber was made by means of a special measuring system calibrated with radon generators (Pylon Company from Canada). The measurements were carried out by means of the method of two filters.

Concentrations of radon daughters were measured simultaneously by means of the above mentioned methods. The results of measurements for the ALFA-31 detection unit and the RGR mining radiometer are shown on the figure 1.



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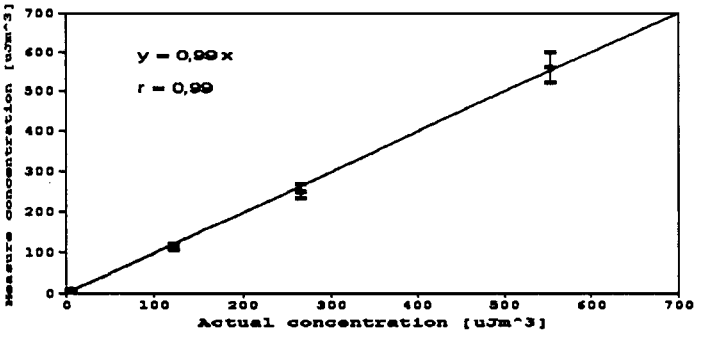

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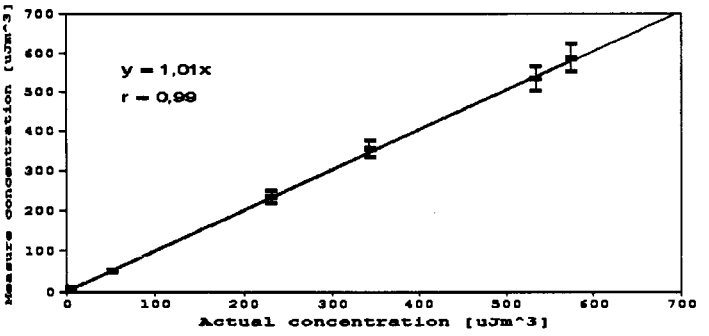
Figure 1

The results of measurements the momentary concentrations of radon daughter made by means of the mining radiometer (a) and periodic concentrations of radon daughters made by means of the ALFA-31 attachment (b).

Figure 2 presents measurements of periodic concentrations of radon daughters made by means of passive and active dosimeters.

The calculated correlations confirmed high measuring compatibility between referential measurements and those made by means of the mining radiometer, ALFA-31 detection unit, passive dosimeter and active dosimeter.



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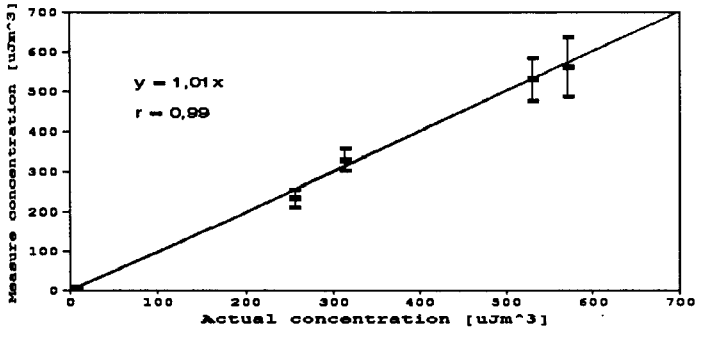

b

Figure 2.

The results of periodic measurements of radon daughters' concentrations made by means of the active dosimeter (a), the passive dosimeter (b).

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POLISH ACTIVE DOSIMETER FOR MEASUREMENT OF INDIVIDUAL EXPOSURE OF MINERS TO RADIOACTIVE RADON DAUGHTER PRODUCTS.

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INTRODUCTION

The widespread occurence of radon ^{222}Rn and its daughters, which together with dust make radioactive aerosols, in Polish mines requires several alternative methods of measurement, considering the wide range of existing concentrations. One of these methods is the individual, active dosimeter worked out in The Institute of Occupational Medicine in Łódź used to measure exposures to radon daughters which a miner gets while working under ground.

CONSTRUCTION OF THE ACTIVE DOSIMETER

The active dosimeter for the measurement of individual exposures to radon daughters consists of a measuring head connected with a micropump by means of a flexible tubing. This solution enables to place the measuring head in miner's breathing zone.

Construction of the active dosimeter was based on the micropump AP-02 made in Poland. The chosen technical data of the micropump: air flow - 30 ml/min; working time from accumulators' batteries NiCd - 24 h; micropump's mass - 0,4 kg; casing dimension - 140x75x56 mm.

The micropump is supplied with the liquid-crystalic projector of cycles' number of the pump's work [1].

The measuring head used in the active dosimeter was designed in the Institute of Occupational Medicine. The scheme of the head is presented on the figure 1. The head is supplied with a microcyclon which has to separate particles of dust of the diameter bigger than 5 μm. The air sucked in by the micropump passes through the microcyclon, and then through the head's chamber, filter and pump's chamber. Radioactive aerosols deposit on the filter. Alpha particles emitted from the filter are registered by the track detector; type Kodak LR-115 [2,3] placed at the bottom of the measuring head, opposite the filter. That flow of air causes that the succeeding layers of dust are not a barrier for the alpha particles coming from the products of radon disintegration. After the exposure, the track detector undergoes chemical etching and microscopic analysis, so as to read the density of tracks on its surface [4].

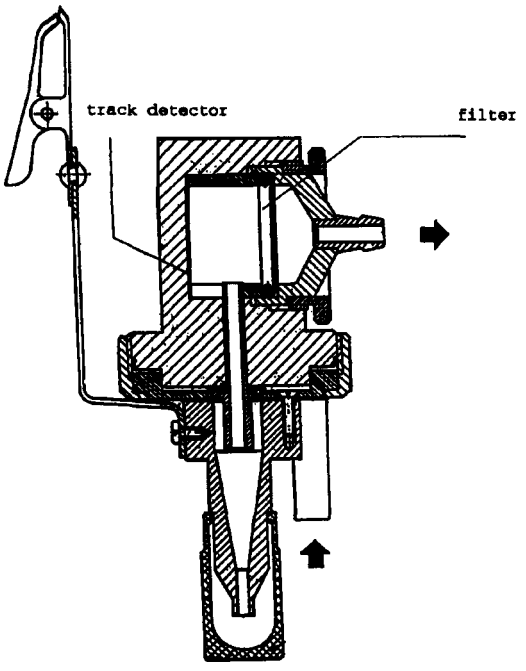

Figure 1. Scheme of the measuring head of the active dosimeter.

PREPARATION OF THE ACTIVE DOSIMETER FOR WORK

Preparation of the active dosimeter for work is mainly based on carrying out the calibration of the dosimetric head. The aim of the calibration is to set the factor K_{WL} defined as:

$$K_{WL} = \frac{g}{E_{RnDp}}$$

where:

g - [trs/mm^2] - surface density of tracks read out from the detector;

E_{RnDp} - [mJh/m^3] - exposure to radon daughters.

Calibration of active dosimeters was carried out in the radon chamber [5]. Each active dosimeter has its own, set calibration factor, considering the fact that there are individual differences of speed of air flow between micropumps. Calibration research showed the independence of the calibration factor from the coeficient F for dynamic radioactive equilibrium between radon and its decay products. The average value of the K_{WL} factor was 10,5 ± 0,5 [(trs/mm^2)/(mJhm^3)]. Research of the stability of the flow of air through the micropump, made by means of the measurer of Digital Flow Calibrators (made by SKC company) showed high stability of the air flow. Measurements of flow were made before and after the exposure in the radon chamber, putting the clean filter to the head. Standard deviation of the measurements was 1%.

MEASUREMENT OF INDIVIDUAL EXPOSURES OF MINERS TO RADON DAUGHTERS

The active dosimeter will be individually given to a miner. The micropump of the active dosimeter will be turned on in the moment of miner's going under ground, and it will be working all the time till he will finish his work. After the end of the work there will be the change of pump's accumulators lasting until the moment of the next miner's going under ground with the dosimeter. The filter of the measuring head is changed twice a working week so as to enable the continuous air flow. After the measuring cycle (1 month), the track detector is changed. On the basis of the read out density of tracks from the detector's surface, there will be established the exposure to which a miner was exposed. There is the special manual for miners who make measurements of individual exposures to radon daughters by means of the individual active dosimeter.

SUMMARY

The active dosimeter worked out in the Institute of Occupational Medicine was tested in the mine of zinc and lead. Fifty monthly measuring cycles were made. The test confirmed the usefulness of the active dosimeter in measuring individual exposures of miners to radon daughters.

However, taking into account high costs of the measurer and great labouriousness of measurements, the use of this dosimeter will be limited to cases where there is considerable radiational danger in a mine.

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RADIATION EXPOSURE IN THE WISMUT MINES

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ABSTRACT

Uranium mining grew fast in saxonia after second world war. No radiation protection was performed in the first "wild" years (1946-1954). Winning with air hammer and drilling had been done without dust reduction and led to an enormous airborne dust concentration. These bad working conditions were unique world-wide. Measurements of dustborne activity concentrations had not been taken. To reconstruct the exposure conditions of miners in these years, four series of experiments under original working conditions were carried out. Stress must be laid on the fact that these measurements should result in the received doses according to uranium and it's long living daughters. Personal and stationary air samplers were used to collect the dust. Activity concentration measurements were done by Gamma spectrometry.

INTRODUCTION

To verify the order of magnitude of the exposure to long living radon-daughters the miners had been exposed to in the first years of the WISMUT, the exposure conditions should be reconstructed as original as possible. To do so, the old pickhammers, the original drilling equipment and other original tools had to be used. Also the poor ventilation conditions had to be reconstructed. In the first years of mining, no technical ventilation was used.(Ventilation speed under 0.1 m/sec). These ventilation conditions could be reached through additional ventilation doors, installed near the experimental area.
Substantial for the large dust concentrations during the investigationes was the fact, that drilling and ore winning had to be done dry as it was done from 1946 until 1954. Four experimental series were carried out in the ore fields of Schlema-Alberoda, Ronneburg, Johanngeorgenstadt and Schneeberg.([1],[2],[3])

MATERIALS AND METHODS

Substantiell contributions to the dust emission did only result from dry drilling and dry winning with pick hammer. The other working activities could be omitted as the measurements had shown.
The dust-sampling was done with stationary and personal sampling-eqipments. Fine- and total dust was sampled separately.. To get an idea of the particles diameter cascade impactors were used.
Each experiment series resulted inmore than about 100 samples which had to be analysed later on in laboratory. The dust concentration had been evaluated by determine dust mass and division through sampling volume.
After verifying the radiological equilibrium with alpha spectrometry, the determination of activity was done by Gamma spectrometry. After decay of the short living Rn-daughters, the activities of the two nuclides Pb-214 and Bi-214 could be measured as representatives for the uranium decay series. Uranium concentration was although measured by chemical analyses

when signal for this purpose was high enough. The results of both methods show sufficient agreement.
Taking into account the exhausting work of drilling and picking a breath rate of 1.5 m^3/h was used for further calculations. The different work done during the shift was analysed and a matrix was evaluated containing values for breath rate, working activity and yearly working time. Multiplication to the measured activity concentration resulted in a yearly intake. With regards to the radiological equilibrium the amount of the yearly inhaled activity is the same for each nuclide in the uranium decay series. Based on the new lung model selected tissuedoses were calculated.

RESULTS
The following tables show the result of measured nuclide-concentration, the estimated yearly intake and some tissuedoses.

Schlema-Alberoda

occupation	time-h/year	breath-rate m^3/h	activity-concentration from	Bq/m^3	up to	input Z^{238} from	Bq/year	up to
picking	525.0	1.5	14.9		71.3	11,734		56,149
drilling	472.5	1.5	1.2		5.3	851		3,756
rest	1,522.5	1.2	0.08		0.15	146		274
sum						**12,731**		**60,179**

(mSv) per yearly intake of all nuclides of the uranium decay series in the determined range of about 13 kBq up to 60 kBq:

lung	bone surface	kidneys	bone marrow	liver	spleen
800 - 3600	900 - 4000	80 - 400	70 - 300	130 - 600	40 - 190

Schmirchau
From the 480 minutes of one shift only 120 minutes drilling time were taking into account, because in Schmirchau we have sediment ore and therefore no worth mentioning picking was done.

time-h/year	breath-rate m^3/h	activity-concentration from	Bq/m^3	up to	input Z^{238} from	Bq/Year	up to
628	1.5	1.333		4.540	**1,256**		**4,277**

Estimated dose equivalent commitments (mSv) per yearly intake of all nuclides of the uranium decay series in the determined range of about 1.3 kBq up to 4.3 kBq:

lung	bone surface	kidneys	bone marrow	liver	spleen
80 - 258	90 - 287	8 - 29	7 - 22	13 - 43	4 - 14

Johanngeorgenstadt

occupation	time-h/year	breath-rate m^3/h	activity-concentration Bq/m^3 from	up to	input ^{238}U Bq/Year from	up to
picking	472.5	1.5	2.33	2.82	1,654	1,997
drilling	630.0	1.5	0.17	0.45	162	421
rest	1,417.5	1.2	0.08	0.15	146	274
sum					**1,962**	**2,692**

Estimated dose equivalent commitments (mSv) per yearly intake of all nuclides of the uranium decay series in the determined range of about 2 kBq up to 2.7 kBq:

lung	bone surface	kidneys	bone marrow	liver	spleen
119 - 160	132 - 178	12 - 16	10 - 14	20 - 27	6 - 8

DISCUSSION

Until now radiation exposure of miners was presumed resulting substantially only from short living Radon daughters. The presented investigations show, that the exposure to the long living decay products beeing taken up by the dust, could contribute fairly high to radiation exposure. Of course this is valid especially for the first years of the WISMUT, while dry drilling and dry ore winning had been performed.

The results are of course first restricted to the experimental situationes. Next step will be to evaluate models to expand the results to other situationes. It is no doubt, that only few of the large number of influencing parameters can been taken into account. Nevertheless will this lead to best known approach to the dosimetric situation of the miners.

The experiments already resulted in a reavaluation of the probility of causation for extrapulmonary cancers of the miners. As known, in former times only lung cancers had been acknowledged as occupational diseases of uranium miners induced by radiation. Taking into account the new results, these cases are now under revision.

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RECONSTRUCTION OF THE COMPLEX JOB-EXPOSURE-MATRIX FOR IONISING RADIATION IN DIFFERENT SITES OF THE URANIUM MINERS IN THE FORMER DDR

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As now well known, in the uranium mines of the former DDR higher collective dose had been accumulated as workforce had ever before been exposed to.
More than 400,000 workers had been employed between 1946 and 1990. More than 5000 cancer diseases have been acknowledged as occupational induced by ionising radiation.

The German workmen compensation institutes, who are responsible for the rehabilitation and compensation of occupational accidents and diseases, started a great investigation and research program to secure and evaluate millions of data files and medical reports, stored in the magazines of Wismut, which can contribute to our knowledge of the biological consequences of ionising radiation. Also included in this program is a project for evaluating a job-exposure-matrix [JEM] of those workers.
Of course the project is divided in several sub-projects and a lot of institutions are contributing.

The Bergbau BG (Industrial Injury Insurance Institut [IIII] for the mining Industry) together with the Institute of Radiation Protection of the IIII is working on the JEM.

The idea is, to evaluate -either by data files or by experiments- mean annual exposition from all sources of ionising radiation (that means: Rn; Rn-daughters, Uranium in dust, external gamma exposition) for certain reference-jobs and reference sites and estimate the job-related exposition by scaling for other jobs and sites.

To do so, it must be taken into account, that the expression „WISMUT" does not describe a single site ore factory but a trust which had run more than 30 different mines with more than 400 shafts and tunnels. Surface mining as well as processing plant were operated.

The situation becomes even more complicated when knowing, that a larger number of those facilities had only short operating time and had been shut down 20 or 30 years ago.

The largest problem is given by the missing radiation measurements of the first 10 to 15 years. The first, single measurements of Radon Gas and Photons have been performed 1955. Systematically performed measurements of Radon and short living Radon daughters have not been started before 1967.
That also indicates, that the exposure situation in the first ten years of operation, can not be evaluated by existing measurements. (Experiments to fill this gap of information are described elsewhere in this proceedings.)

Some information is expected to be delivered by a study which is performed by colleges in the former CSSR which has to be implemented in the project.
Other information had already been taken from files who had been kept in Russian records in Moscow. More than 20,000 pages of (written) text and 8,000 mine-drawings have been bought back from there and have already been evaluated.

The project will be finished during 1996 with the reports of the 6 main sub-projects:

1. Radiation protection in the Wismut Mines

2. Radiation exposure in the surface mines of the Wismut

3. Radiation exposure in the processing plants of the Wismut (finished 1995)

4. Radiation exposure in the Thuringia underground mines of the Wismut

5. Radiation exposure in the Saxonia underground mines of the Wismut

6. Radiation exposure by Radon daughters in public mines of the former DDR.

It is of course expected, that the derived JEM, which is first evaluated for compensating purposes is of great value for the just started epidemiological research on the subject.

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MONITORING URANIUM DUST CONCENTRATION IN A MINING CONCENTRATE CONVERSION PLANT INTO UF4 AND UF6

COMURHEX is a plant for converting mining concentrates into UF4 and UF6. The operations are carried out in various facilities. The atmosphere of these shops is monitored daily using APA's (Aerosol Sampling Apparatus) placed in selected locations depending upon the work stations used by the operators.

The samples are measured daily; a first count is made immediately after sampling and a second one three days later. The first and second counts are plugged into a computer program developed within Comurhex-Malvesi. This program enables the data in $Bq\alpha/m^3$ and $Bq\beta/m^3$ from the APAs to be displayed for each shop on individual diagrams and for the whole of the APAs of the plant on a single diagram.

Guide values are reference values corresponding to the usual values for the shop. They are indicated on the diagram for immediate comparison with the present values.

It is possible to display each APA over a set period. The period that is displayed by default is the last 30 days. Incidents and gradual drifts may thus be identified immediately.

Each incident may be commented and the commentary introduced into the program. A very quick access to them makes querying given events easy.

This program allows, from a medical standpoint, to target urinary uranium analyses on workers: any increase of the APA values over a given threshold and/or a certain period of time results in another urinary uranium analysis. These analyses are done on top of the systematic analyses performed periodically.

This program comes also in support of the operators for the maintenance of their shop. It allows the facilities where repetitive incidents occur to be pinpointed much more easily and, better still, to locate those where a gradual drift occurs.

For Comurhex, this program is a tool which makes the medical surveillance of workers easier and helps the operators in the maintenance and management of their shop.

MANUSCRIPT CUT IN LENGTH !

FOR FURTHER INFORMATION PLEASE CONTACT AUTHOR

Standpoint of the EDF operator of nuclear power plants

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1 - EDF General Policy in terms of and remote systems for the nuclear generation capacity

Safety and competitiveness are the sine qua non conditions to the durability of the nuclear industry in France, as in the other countries involved in electronuclear programs.
Improving safety, increasing availability and mastering the costs are inseparable and interactive aims, and the cooperating actions are to be managed simultaneously.

Another aim is the undivided attention given to radiation protection. The average annual collective dose per unit has slightly decreased since 1993 despite an increase in the circuit activity and the volume of the works. This result is not thoroughly satisfactory if compared to the performances carried out abroad on similar reactors.

EDF has therefore decided to revise the establishments of contracts with the contractor companies on the following bases :
- a lasting partnership bases on common aims, mutual obligations, and shared benefits
- complete and better integrated services, giving the companies a greater responsibility on the means to be implemented to achieve those aims
- pluriannual contracts, enabling the companies an improved insight of the future load, and thus giving them the opportunity to invest (in personnel training, in tools or in innovation processes, etc.),
- permanent research in the field of the best technologies available in France and abroad, through supplier diversification and international experience feedback.

2 - Maintenance: main intervention field of robotized or remote systems

Robotics is an improvement factor for **maintenance** together with the optimization of programs and organizations and the improvement of performances of the repair **processes** implemented.

Robots are used to implement, in a generally hostile environment, maintenance processes for which the maintenance contractors and EDF have at their disposal specialized skill, mainly :

- welding and related techniques (machining, cutting, ...),
- non-destructive examinations (visual, ultrasonic, eddy currents, dye penetrant testing .),
- handling measurement instrumentation (metrology, vibration, temperature, exposure/contamination ...),
- setting up specialized tools (component assembly or disassembly ...),
- constraint works (setting up shields, air-locks, decontamination ...)

In the field of maintenance, robots and remote systems concern :
- the reduction of personnel exposure to irradiation and the anticipation of more severe regulations in this field,
- productivity increase aiming at an improved availability and a decrease in contractor costs,
- a higher tool changeability to reduce the direct development costs, the costs induced by the storage and the maintenance of contaminated components, as well as an improved integration of work sites requiring several successive processes during outage,
- a higher process reproducibility,
- an improved interventions data memorization capacity,
- adaptable solutions for exceptional or accidental situations in which man cannot act.

This list illustrates the many sides to developing robotized systems. Considering the nuclear plant specificities (geometry, admittance, environment), the development costs and the numerous targets aimed at, the purely technical approach is no longer adequate: the goal is to establish a system in which the maintenance processes, the economy in the broad sense of the term, and the participants are carefully assessed and measured out according to the aims. Apart from this "system" approach, developing robotized systems entails very severe technical and economical risks.

Robots and remote systems are globally considered as possible intervention means competitively to other means, in the following situations :

- Planned maintenance

The nuclear generation capacity maintenance is essentially preventive. The preventive maintenance operations concern components in a non-failure state and cover various techniques: inspections, examinations (eddy currents, ultrasonic, X rays,...), repairs (excavation cavity, welding,...). They are defined and planned according to preventive maintenance programs. Most are carried out during refueling outage occurring every twelve or eighteen months.

- Exceptional maintenance

Exceptional maintenance corresponds to repairing, refurbishing or replacing parts subject to unexpected damages. As an example, the crack in the vessel head adapters required the implementation of a three-fold problem determination strategy: adapter control to detect possible cracks, repair and refurbishment of the vessel heads. to this example. robots played an important part in the control.

- Post accidental interventions

The post accidental situations taken into consideration go as far as partial core meltdown with or without radioactive release outside the plant.

A distinction must be made between the needs in identification and preparation of the environment on the one hand, and the actual maintenance or operation interventions on the other hand.

Due to the high number of unknowns which are the main concrete characteristics in this type of scenarios, it has not been estimated realistic to enter into robots or remote systems to achieve maintenance (components repair or replacement ...) or operations (device operations...) strictly speaking.

The development focuses as a matter of priority on robotized or remote control systems for environmental survey (radioactivity, temperature, obstacles, identification of the state of equipment ...) prior to the operation itself. Robots or remote systems may nevertheless have a useful application as a means of assistance to the operator (tool carrier, protection shields ...) or even to carry out first level light interventions (sealing off leaks, picking up objects ...).

These applications are entrusted to GIE INTRA, in charge to developing and maintaining a set of mobile equipment into operation on behalf of three French nuclear operators : EDF, CEA (the French Atomic Energy Commission) and COGEMA.

- Robotized or remote systems maintenance of the future standardized plants

Experience feedback has shown the difficulties in relying on robots or remote systems when the interventions have not been considered in the original design of the plants. Being the industrial architect of its own plants, EDF takes part in designing the future plants so as, subsequently, to facilitate the implementation of robotized or remote systems. The arrangements contemplated mainly concern the high capacity communication networks, easy ways to achieve absolute locating in space, a less constrained spatial environment.

3 - Short historical review

During the design of nuclear power plants, a certain number of arrangements were made to facilitate maintenance operations, in so far as they could be planned. A certain number of accesses, of utilities, of pressure equipment access ports have thus been planned according to the corresponding timely criteria: maintenance type and volume, technology of the maintenance means, degree of equipment contamination, health physics regulations. In addition to the design rule book, there were compulsory in-service inspections criteria for the important to the safety equipment. To this effect, a certain number of tools were designed and supplied from startup: In-service inspection machine, automatic controller of the steam generator tube bundles

("Spiders"). As for a certain number of maintenance operations directly linked to plant operation, tools have been designed almost as early as the beginning to reduce personnel integrated doses or to improve the quality of operations (consideration for the parameters, traceability, productivity): simultaneous stud tensioning machines, fuel handling station.

These two concomitant factors modified the initial data and justified deep changes:

- the progressive discovery of youth defects requiring, on the one hand, interventions on equipment not initially planned in the maintenance programs, and, on the other hand, the intensification of control or maintenance programs on a certain number of equipment,

- the evolution of technologies in the field of maintenance tooling (power actuators, instrumentation and control equipment, sensors ...) which progressively enabled the development of remote machines such as spiders towards robots allowing a greater site integration and an increased supervision of operations.

Competition was therefore stimulating and the technical evolution enabled the realization OT the t projects. The robots thus replaced automatic machines. The [movements of the automatic machines (Spiders, SMOR...) are tdesign indexed according to the steam generator tube plate. They are controlled by on-off sequencers. The multi jointed robots (ARAMIS for FRAMATOME, ROSA for WESTINHOUSE, RITMIC for INTERCONTROLE, FLEXIVERA fort ABB-CE) can reach the entire work area, change tools or process without carrier replacement (See figures 1,2,3).

Technologically, this evolution was only possible because of the contributions from the industrial sectors independent from the nuclear industry: this is probably one of the main lessons for the future. To be more precise, this evolution was due to the contribution of hardware and software progress to instrumentation and control. All the instrumentation control "layers" have undergone profound evolution, from the lower level of servo-control up to the supervisor responsible for quality and data memorization capacity.

4 - Research and Development in Robots and Remote control at EDF

3.1 - Main objectives

EDF intend keeping a Research & Development activity in Robotics and remote control operations for several reasons :

- on the one hand, to have some knowledge and an ability of assessing the technologies available in France and abroad, a standing action of technological survey being carried out on the subject,
- on the other hand, to carry out research and development actions in a certain number of fields considered significant so as to have a technical expert assessment available, and to participate in the removal of certain technical locks; these: actions are either achieved on technological components (or technological bricks), or in selected subject themes.
- and lastly, to stress the interest of the process with laboratory experiments, and the possible improvements.

Part of this process is achieved within a pluriannual Research and Development Project: the START (Projet de Système de Téléintervention Avancé Robotisé Transposable, or "Advanced Remote Robotics system for Intervention project"). Part of this R&D also concerns the use of remote systems and robots for non nuclear EDF applications interventions on energized electric lines, robotized repair of hydraulic turbines bladings, etc.

4.1- Handling systems

The current approach consists in identifying the handling systems available on the market and to assess their capacities. Some adaptations and improvements may then be suggested and carried out. One of the handling systems withheld is the hydraulic industrial robotics arm (SCHILLING TITAN) with a high load capacity of around a hundred kilos as comparable to its own weight. It has been characterized and has been implemented for various demonstration operations in remote control or robot mode.

It proved necessary to have a second and lighter handling system, with a smaller load capacity and yet higher repeatability characteristics to carry out more precise operations (in particular welding). Following a study of the world market supply, the choice fell on a ten-kilo-capacity industrial electric handling system weighing thirty five kilos, with 7 degrees of freedom. This is an interesting characteristic feature from the point of view

A GENETIC ALGORITHM APPROACH TO OPTIMIZATION FOR THE RADIOLOGICAL WORKER ALLOCATION PROBLEM

– DISCUSSIONS ON DIFFERENT HARD CONSTRAINTS –

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INTRODUCTION

The worker allocation optimization problem in radiological facilities inevitably involves various types of requirements and constraints relevant to radiological protection and labor management. Some of these goals and constraints are not amenable to a rigorous mathematical formulation. Conventional methods for this problem rely heavily on sophisticated algebraic or numerical algorithms, which cause difficulties in the search for optimal solutions in the search space of worker allocation optimization problems.

Genetic algorithms (GAs) are stochastic search algorithms introduced by J. Holland in the 1970s based on ideas and techniques from genetic and evolutionary theories. The most striking characteristic of GAs is the large flexibility allowed in the formulation of the optimal problem and the process of the search for the optimal solution. In the formulation, it is not necessary to define the optimal problem in rigorous mathematical terms, as required in the conventional methods. Furthermore, by designing a model of evolution for the optimal search problem, the optimal solution can be sought efficiently with computational simple manipulations without highly complex mathematical algorithms.

We reported a GA approach to the worker allocation problem in radiological facilities in the previous study (2). In this study, two types of hard constraints were employed to reduce the huge search space, where the optimal solution is sought in such a way as to satisfy as many of soft constraints as possible. It was demonstrated that the proposed evolutionary method could provide the optimal solution efficiently compared with conventional methods. However, although the employed hard constraints could localize the search space into a very small region, it brought some complexities in the designed genetic operators and demanded additional computational burdens. In this paper, we propose a simplified evolutionary model with less restrictive hard constraints and make comparisons between the two models.

MODEL CONSTRUCTION

A simple test problem is specified as follows : a job is assumed to be accomplished when the required working time on the job is fulfilled. The predicted dose that a worker is exposed to is estimated by multiplying the Area Dose Rate (ADR) with the Working Time (WT):

$$\text{Predicted Dose} = \text{ADR (mSv min}^{-1}) \times \text{WT (min)}.$$

- Input Conditions:
  - (*A*) 12 workers to be assigned to 5 jobs (each with one workplace) to accomplish the project.
  - (*B*) The Cumulative Dose for a worker before the assignment is CD[1-12] = 0.4, 2.2, 1.5, 0.7, 1.2, 0.0, 0.9, 0.0, 0.0, 0.2, 1.8, 0.0 mSv.
  - (*C*) The Area Dose Rate for a job is ADR[1-5] = 0.1, 0.02, 0.05, 0.04, 0.04 mSv min^{-1}.
- Dose Limitations:
  - (*D*) Worker exposure should not exceed the Dose Limit, DL mSv. Here, DL is set to 5 mSv.
  - (*E*) Working assignments to a workplace should satisfy the Dose Constraint of that workplace, DC[1-5] = 5.0, 3.0, 4.0, 3.0, 4.0 mSv.
- Labor and Safety Constraints:
  - (*F*) The Working Time required to carry out a Job is JWT[1-5] = 120, 200, 360, 300, 100 minutes.

(*G*) The working time for a worker should be longer than 30 minutes, but should not exceed 120 minutes.

(*H*) For special skill requirements, worker[1] is required to work at workplace[1] for more than 50 minutes, and worker[2] to work at workplace[3] for more than 80 minutes.

(*I*) In this test model, workers [1-6] are set as skillful workers, and their total working time in workplace[3] should be more than twice as long as that of workers [7-12].

(*J*) All the working times assigned to workers are integers.

GA APPROACH

Genetic algorithms(GAs) are search and/or optimization algorithms based on the mechanisms of natural evolution. GAs maintain a population of a finite number of individuals during the evolutionary process. The evolutionary process is an iteration process which seeks better and better solutions from generation to generation. In each iteration, a new population is formed by applying genetic operators to the old population. Based on the fitness ranking, individuals can hand down their genes to the next generation by one of the following three genetic operators : reproduction, crossover, or mutation. After the evolutionary process approaches an equilibrium state, the best individual is considered as the optimal solution.

For a highly constrained problem like the worker allocation problem in radiological facilities, various types of constraints including even mutually conflicting ones should be taken into account in the process of optimization. In the previous study, we presented an evolutionary model where the concepts of hard constraints and multiple fitness were introduced to search efficiently for the optimal solution. The constraints (D) for dose limits and the constraints (F) for labor power requirements were employed as hard constraints. The optimal solution was sought among feasible candidates defined as individuals that satisfy both of the hard constraints. However, the proposed genetic operations required complicated algorithms and even repair mechanisms to keep the feasibility of each individual in the evolutionary process. In this paper, we propose an evolutionary model that can seek the optimal assignment more efficiently with simpler genetic operations. The proposed model is outlined as follows :

Chromosome : The chromosomes of the individuals are specified by two dimension matrices whose entries, for example WT_{ij}, express the working time of the worker[j] at workplace[i].

Hard Constraints : For simplifying the genetic operations, only the constraints (F) are employed as hard constraints. Other constraints are treated as soft constraints. Weighting coefficients are assigned to the soft constraints in accordance with their priorities in the radiological protection and labor allocation. The first priority is given to the constraints (D) for dose limits to compensate for being removed from the hard constraints.

Evaluation Functions : Three types of fitness functions with different priorities are introduced in this model. 1) The total number of satisfied soft constraints is used as the principal fitness function; 2) The total sum of the weighted degree of violation of the unsatisfied soft constraints is defined as the secondary fitness function. 3) The total number of times required for workers to change workplaces during the project is employed as the third fitness function. A certain portion of individuals having low fitness are discarded from the population at every generation in the evolutionary process.

Mutation : The mutation defined here is much simpler than that in (2). The individual to be mutated is selected randomly. The mutation is performed by reassigning some workers' working times at some workplaces. For example, ith row is selected to be mutated, 2 entries in ith row are selected randomly, at least one of them is not zero. The working time is reassigned randomly in one of the following two ways: 1) assign a worker the maximum working time within the dose limit, and allocate the rest of the working time to the other; 2) redistribute the total working time of the two workers randomly between them.

Crossover : The parents are selected from the individuals and the row positions of the matrix chromosome for the crossover operation are determined randomly. A simple arithmetical crossover is applied to the selected rows of the parent chromosomes. In the crossover process, two offspring are born to maintain the number of the population during the evolutionary process. Since the genetic operators employed here always ensure the feasibility of the offspring, therefore the repair algorithms as defined in

(2) are not necessary.

RESULTS AND DISCUSSIONS

With the evolutionary model described above, the first feasible solution was found within 600 generations, and an equilibrium state was gradually reached after 15,300 generations. The results show that the labor allocation satisfies all other soft constraints except the special skill requirements. In addition, by virtue of the special skill requirements, the skilled workers, [1] and [2], are assigned to the specified workplaces with the maximum times allowed under dose limits. The number of times required for workers to shift workplaces can be reduced to 7 times. An optimal labor allocation is given in Table 1.

	Individual												
Workplace	1	2	3	4	5	6	7	8	9	10	11	12	Total [min]
1	46	0	0	0	0	0	16	0	0	48	0	10	120
2	0	0	0	58	40	0	102	0	0	0	0	0	200
3	0	56	66	62	0	80	0	47	49	0	0	0	360
4	0	0	0	0	75	23	0	64	63	0	75	0	300
5	0	0	0	0	0	0	0	0	0	0	0	100	100

Table 1: Results of the present model.

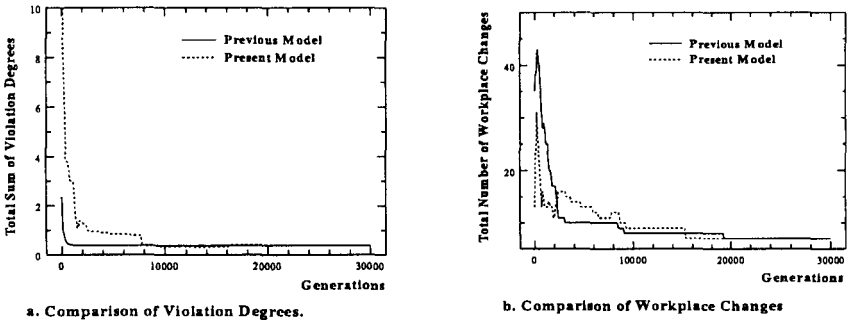

Figure 1: Comparisons between the two models.

For the purpose of comparison, the model proposed in (2) is also applied to the same problem. The results are almost the same as those of the model provided in this paper. The first feasible solution was found within 100 generations, and the best solution was found after 19,300 generations. The comparisons between the two models are shown in Fig.1.

From the comparisons, we found that the previously proposed model gave the first feasible solution very quickly, while the present model established the best solution faster than the previous one. As previous model employed more hard constraints than the present one, the search space became smaller. As a result, it was easier to find the first feasible solution. On the other hand, because of the much more strict restrictions of the previous model, it brought some complexities such as repair algorithms in designed genetic operators, which influenced the convergence efficiency.

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RADIATION PROTECTION IN THE DECOMMISSIONING OF A POST ACCIDENT REACTOR

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ABSTRACT

This paper describes the control and limitation of dose uptake to operators during the early stages of decommissioning of the Windscale Piles. This was achieved by careful planning, the use of inactive trials, thoughtful use of remote handling techniques and review and feedback of information. Built between 1947 and 1950, the Windscale Piles were shut down following the Windscale Incident in 1957. UKAEA Government Division are now undertaking the early stages of decommissioning of these facilities, removing material from the air and water ducts and preparing for subsequent core removal. As part of the overall strategy of UKAEA GD, this work is being carried out using contract staff including the use of a Managing Agency, W S Atkins (Northern).

Decommissioning utilises the same means of dose reduction and control as any other nuclear operation although sometimes in novel ways. In the Windscale Piles, fully remote operations have been used to remove fuel and debris from the environs of the core which was damaged during the 1957 incident. Much use has also been made of training in mock-up facilities allowing manual techniques to be used for some jobs. The implications of using various different contractors rather than an in-house team is also discussed. It is concluded that decommissioning of major facilities can be carried out within acceptable dose uptake criteria by utilising both novel and adaptations of traditional, active handling techniques.

INTRODUCTION

The Windscale Piles were constructed between 1946 and 1950 primarily to produce material for the nuclear weapons programme but also to produce radioisotopes for industrial and medical uses. The first pile went critical in October 1950. The Piles consisted of a graphite moderator block containing 3444 fuel channels running horizontally through the core and cooled by forced draught filtered air which after passing through the core was exhausted via the 125m chimney at the top of which were filters. Fuel cartridges were introduced via a hoist at the charge face and discharged into water ducts at the back of the pile from where they were taken for reprocessing. The reactors operated at a power of 180 MW until 1957.

On 10 October 1957 an accident occurred in Pile 1 during a routine Wigner energy release, involving severe overheating of parts of the core. The emergency, which was to be the most serious for several decades was finally brought under control on 12 October 1957 by shutting down the ventilation blowers and by the application of copious amounts of water to the Pile. Following the fire, both piles were shut down and fuel and isotope cartridges discharged, except for some 10% of the Pile 1 core which could not be moved. Sealing of the pile concrete shells and inlet air ducts was carried out and the associated buildings cleared and put to other uses. The piles were kept under care and until planning for decommissioning was begun.

Recently surveys were carried out and work has begun on the removal of loose fuel and debris from the air ducts and water ducts. This work has involved handling highly radioactive materials in difficult and poorly documented areas. In line with UKAEA policy the work has been carried out by various companies under contract and has been assisted in the project management by a Managing Agency (W S Atkins). This work has shown how standard radiological protection methods can be adapted even in these severe situations and the value of harnessing the experience of the wider contracting community has been shown. Some examples of this are given in this short paper.

AIR INLET DUCTS

The inlet air ducts were the route from the blower buildings either side of each pile for cooling air to enter the reactor cores. Visual and radiological surveys showed that the ducts contained dust and debris and in Pile 1, a quantity of fuel in various conditions. Using the results of these surveys, methods of clearing the ducts were developed. Radiological containment tents with local ventilation were used at points where access was gained to the ducts by cutting through the duct walls. These allowed the deployment of remotely operated vehicles (ROVs). Dose budgets were prepared for each task and dose uptake was monitored using daily issue electronic personal alarming dosimeters.

The Pile 2 air inlet ducts were cleaned and seismically qualified barriers were installed near to the Pile. The barriers consisted of steel formwork and precast, interlocking concrete blocks. They were installed manually using portable lead screens to shield the workers from radiation from the core. Bulkhead doors are now in place and work is complete. The total dose uptake for the installation of the Pile 2 barriers was 9.4 man.mSv with a maximum individual dose of 3.1 mSv.

Because of the fuel in the Pile 1 inlet air ducts, the dose rates were higher than for Pile 2 (700μSv/h cf 120μSv/h). Cost benefit analysis for the clearance work would have resulted in a manual method but a policy decision was taken to use remote techniques. A mock-up facility was used for the testing of the ROV and the training of the operators. The ROV was then used to collect all the fuel and debris in the ducts and put it into specially designed canisters which were then removed by flask. The ducts were then vacuumed remotely. A more remote method had been considered, but on completion of the clearance work the dose rates were sufficiently low (320μSv/h) to allow the same manual construction of the barrier as had been used in Pile 2. Temporary shielding was used and minor modifications to the installation method allowed reduced occupancy times (eg delivery of grout though a hole in the duct roof). The total dose for the installation of the Pile 1 barriers was 15.6 man.mSv with a maximum individual dose of 3.1mSv.

AIR EXHAUST DUCTS

As with the Air inlet ducts, considerable amounts of fuel and debris lay in the air exhaust ducts. It was necessary to collect this material and transfer it to Disposal/storage locations. The obvious access route was via the base of the Pile chimney on the adjacent BNFL Nuclear Licensed site. Installation of equipment through this route would have been dose-intensive and there would have been potential radiation exposure from the packages as they were removed. There were also administrative complications since nuclear materials would be transferred to another site and back again.

An alternative scheme for the exhaust air duct clearance was adopted. This scheme took advantage of the existing shielding of the Pile bioshield. Air dams installed in the duct during an earlier part of the decommissioning could be modified to provide an ROV deployment route which did not require personal significant radiation exposure.

A jib crane was designed for deployment through the existing pile inspection holes. This crane was used to suspend fuel containers in a position where they could be loaded by ROVs. The crane then allowed the containers to be lowered onto the floor of the water duct so that they could be removed along with fuel from the water duct itself (see below).

WATER DUCT

The clearance of the pile water ducts was broken down into a number of tasks: Barrier Installation; Sludge pumping; Fuel/Debris removal; Decontamination; Drainage.

The installation of a seismically qualified barrier was done early in the process to secure containment of the pile should an earthquake or large dropped load breach the duct. Work on two tasks could then take place at the same time, one in the section of duct behind the pile and the other in the external section of the duct. Most work in the ducts was done by ROV.

The flocculent sludge found in the duct had to be cleared from the duct to prevent debris being hidden and allow it to be retrieved by ROV. The method used for this was almost entirely remote using a sludge pipeline from the duct

The flocculent sludge found in the duct had to be cleared from the duct to prevent debris being hidden and allow it to be retrieved by ROV. The method used for this was almost entirely remote using a sludge pipeline from the duct to the Pond. The ROV was then connected to the pipeline and deployed. Dose uptake on the sludge pumping was almost entirely due to the operations to install the pipeline and the deployment/retrieval of the ROV. The ROV was purpose designed for the job and could not practicably be proved to very high levels of reliability before work started. Several retrievals and redeployments of the vehicle were needed before the task could be finished and this had the effect of increasing the dose uptake.

Another purpose designed ROV was used for fuel and debris removal. This vehicle was equipped with a remote manipulator arm, a γ dose rate probe and stowage for fuel and debris containers. The γ probe was used to identify any items such as ^{60}Co pellets which would have a high dose rate when retrieved at the access shaft. Fuel and isotope cartridges were retrieved into shielded containers for transfer to a flask loading facility. Careful assessment of the dose rates from fuel, based on the calculated inventory of the material and dose rate measurements in the air ducts, made it possible to relax the shielding requirements for the fuel transport flask loading facility. It was possible for workers on a platform at the flask top to transfer fuel from the recovery containers underwater within the flask itself using long handled tools.

All work on the water ducts needed careful planning to ensure that an ALARP solution was found. Outline designs were prepared and subjected to a HAZOP study. From this any need to compare options to achieve the optimum dose uptake was identified. The firm design was prepared by the contractor selected for the work and ALARP meetings held to discuss the detailed design where the HAZOP had identified the potential for high dose uptake. Regular progress meeting with the implementation contractor allowed the opportunity to review the dose uptake implications of the work and discuss controls to minimise dose. Contractors were selected with proven experience of nuclear work and, where appropriate, experience of remotely operated technology. Contractors were selected from companies from UK and France.

The total dose uptake predicted for water duct clearance operations to date was 61.1mSv. The total actually accrued is 44.8mSv. No individual received more than 4μSv, well within the dose restraint objective of 8mSv/year. A more detailed breakdown of the dose uptake figures is given in the table below:

Task	Predicted (mSv)	Actual (mSv)
Water Duct Barrier Installation	20.7	11.8
Water Duct Sludge Pumping	15.6	15.3
Water Duct Fuel/Debris Removal	36	15.3 (work continuing)
Exhaust Air Duct Clearance	6.8	2.4

CONCLUSION

At UKAEA Windscale experience is being gained in the decommissioning of facilities which present a much greater challenge than power reactors will. This experience has indicated that in terms of radiological protection the operations involved, even in the extreme example of the Windscale Piles, are little different from those encountered in everyday operations of a nuclear establishment. By careful planning, inactive trials and training use of remote handling equipment and by using techniques common in other industries, dose uptakes can be kept ALARP.

ACKNOWLEDGEMENT

The Windscale Piles project is funded by the UK Department of Trade and Industry and the Ministry of Defence.

The International System on Occupational Exposure, ISOE Status and Results for 1995

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Introduction

Throughout the world, occupational exposures at nuclear power plants have been steadily decreasing over the past decade. Regulatory pressures, particularly after the issuance of ICRP 60 in 1990, technological advances, improved plant designs, and improved water chemistry and plant operational procedures, as well as other factors, have contributed to this decreasing trend. However, with the ageing of the worlds nuclear power plants the task of maintaining occupational exposures ALARA has become increasingly difficult. In addition, economic pressures have lead plant operation managers to streamline refuelling and maintenance operations as much as possible, thus adding scheduling and budgetary pressure to the task of reducing operational exposures.

In response to these pressures, radiation protection personnel have found that by properly planning, preparing, implementing, and reviewing jobs, occupational exposures can be kept as low as reasonably achievable. To facilitate this global approach to work, sometimes referred to as Work Management, through the exchange of techniques and experiences in occupational exposure reduction, the Nuclear Energy Agency (NEA) of the Organisation for Economic Co-operation and Development (OECD) launched the Information System on Occupational Exposure (ISOE) on 1 January 1992 after a two year pilot program. This four level data base joins utilities and regulatory agencies throughout the world, providing occupational exposure data for trending, cost-benefit analyses, technique comparison, and other ALARA analyses.

The ISOE Structure

The ISOE system consists of four data bases of occupational exposure information. Data is collected via user-friendly computerised questionnaires, and accessed via a Windows-based, relational data base system. The first data base, NEA 1, concerns operating reactors, and those in cold shutdown (stored separately). For each participating reactor various radiation protection performance indicators: total annual collective dose, non-outage annual collective dose, outage annual collective dose divided into 20 job categories and 75 sub-categories, annual collective man-hours and number of workers associated with each job category and sub-category, and annual individual dose distribution are included. Although not all reactors provide data for all categories, all the data provided are updated annually.

The second data base, NEA 2, contains for each participating reactor information concerning methods and techniques used for dose and dose rate control. Primary water chemistry, cobalt replacement/reduction programs, primary water filtering, surface preconditioning, decontamination, work practices, ALARA organisation and management, tools and procedures, and motivation and training practices are listed. The dosimetric effect of each practice is quantified as best possible. This type of information normally evolves rather than changes, thus this data base is updated by the participating utilities on an as needed basis. Information for this data base is still in the process of being collected.

The third data base, NEA 3, contains details on the dosimetric results of specific operations. Items as large as the removal of the reactor temperature detector bypass system, or as specific as reactor vessel head control rod drive penetration inspections have been the subjects of NEA 3 reports. Important radiological aspects of the operation, and the name, address, and phone number of a contact person for further information are listed. The participating utilities are encouraged to complete NEA 3 reports as often as they perform operations with interesting radiation protection aspects.

The fourth data base, NEA D, is not yet operational. This data base will contain information similar to that contained in NEA 1, however exclusively for plants which are definitively shut down or are in some phase of decommissioning. It is hoped that this data base will be operational in early 1996.

Current Status of Participation

As ISOE nears the end of its fourth full year of successful operation, its list of participants continues to grow, and currently includes 59 utilities from 19 countries, and 14 national regulatory authorities. Additional data from some non-participating reactors is collected from published reports, such that the data base now represents

approximately 350 reactors (over 80% of the operating reactors world-wide), including PWRs, BWRs, CANDU reactors, and GCRs. The data base also includes 34 definitively shut-down reactors.

In terms of ties between ISOE and other organisations, the International Atomic Energy Agency (IAEA) cosponsors ISOE for non-NEA Member countries, the European Commission (EC) and the NEA have signed a co-operative agreement such that the ISOE data base now also serves the European Community's data needs, and the Paris Centre of the World Association of Nuclear Operators (WANO-PC) and the NEA have signed a Memorandum of Understanding to assure co-ordination of the activities of the two organisations in the field of occupational exposure. Finally, to administer the collection and distribution of data, a system of Regional Technical Centres has been established. This includes the IAEA for non-NEA Member Countries, the Nuclear Power Engineering Corporation (NUPEC) for the Asian Region, the Centre d'Etude sur l'Evaluation de la Protection dans le Domaine Nucléaire (CEPN) for the European Region, and the University of Illinois for the North American Region.

Thus ISOE has a wide following and is the most complete occupational exposure data base in the world. The value of such a widely used system is its ability to efficiently facilitate the exchange of occupational exposure reduction experiences and practices among participants.

The Use of the ISOE System and Network

There are several diverse ways in which ISOE can be used by its participants. The ISOE System, consisting of the four data bases and their associated software, can be used for statistical and comparative studies, and as a source of good practices and experiences. As a brief example of the type of data available through ISOE, below are listed the average annual collective doses, per reactor, for various countries and reactor types participating in ISOE, as well as a figure showing the progression of various doses and durations associated with steam generator replacements around the world.

The ISOE Network, which consists of all Participating Utilities and Authorities, Regional Co-ordinators for certain countries, and the ISOE Technical Centres, serves as an open line of communication for the real time exchange of data, experiences, policies, practices, etc. Participants interested in the experience of others in specific areas not already covered in the data bases may request that the Technical Centres solicit the needed information. Participating utilities, authorities, and national ISOE co-ordinators are then contacted by the Technical Centres and the resulting information is passed on to the requester, and made available to all other participants.

In addition, ISOE Expert Groups are established from time to time to perform specific studies based on participant's needs. For example, many companies have adopted a very global approach to their work, stressing the importance of approaching jobs from the multi-disciplinary team perspective, and of following jobs completely through the stages of conception, design, planning, preparation, implementation, and follow-up. By focusing such attention on jobs, their successful completion - on schedule, within budget, with a sufficient level of quality, with minimum cost, and with a maximum chance of fulfilling the originally desired goal - can be assured. This multi-disciplinary, start-to-finish approach to jobs can be broadly termed ***Work Management***. To facilitate the application of Work Management principles in the nuclear industry, an NEA ISOE Expert Group is currently preparing a "Handbook of Good Practices". In each of seven areas, this Handbook describe what is currently seen as good practice, and will cite specific case studies to illustrate the techniques used to quantify the impacts of work management actions in that particular area. The areas addressed by the Handbook are; 1. Work Management Policy, 2. Regulatory Issues, 3. Worker Involvement, 4. Work Selection and Planning, 5. Work Preparation and Scheduling, 6. Work Implementation, and 7. Assessment and Feedback.

The Annual ISOE Steering Group meeting includes a Topical Session during which current issues of interest to the participants are presented and discussed. Recent topics have included steam generator replacements, fuel failure, chemical decontamination, and electronic dosimetry.

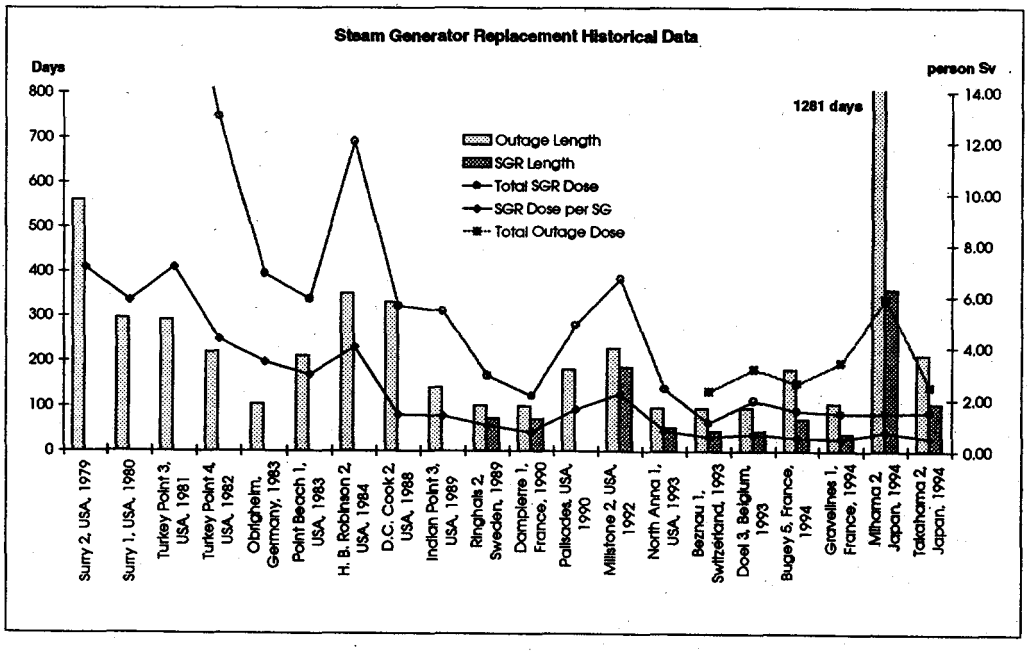
Conclusion

After four years of operation and expanding participation, the ISOE system has entered a period of "usefulness and stability". Continued growth, and efforts by all participants to deliver timely and useful information, will help to assure that ISOE remains an up-to-date conduit for the exchange of occupational exposure experience throughout the world.

Average annual collective dose per operating reactor by type for a number of countries in 1994 (in man Sv)

	Country	PWR	BWR	CANDU	GCR
OECD	Belgium	0.98			
	Canada			1.12	
	Finland	1.17	1.20		
	France	1.74			
	Germany	2.25	2.15		
	Japan	1.07	1.58		0.23
	Mexico		6.03		
	Netherlands	1.82	0.85		
	Spain	1.77	3.94		
	Sweden	0.64	1.71		
	Switzerland	0.79	2.31		
	United Kingdom				0.31[1]
	USA	1.34	3.27		
Non-OECD	China	0.65			
	Czech Republic	0.35			
	Hungary	0.39			
	South Africa	0.81			

[1]: *Some collective dose data from the UK are missing from the ISOE data base.*



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The International ISOE Programme
ISOE Asian Regional Technical Center Activities

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INTRODUCTION

The ISOE Asian Regional Technical Center was established on April 1992 in the Safety Information Research Center of the Nuclear Power Engineering Corporation (SIRC/NUPEC)*1 according to the commission of the Japanese authorities, after the ISOE establishment in January 1992. The participating authorities in Japan are the Science and Technology Agency and the Ministry of International Trade and Industry, and the participating utilities are all (ten) Japanese electric power companies operating the nuclear reactors and the PNC (Power Reactor and Nuclear Fuel Development Corporation). These participants, the Authorities and the Utilities, are actively exchanging with each other dose related operating information. The number of participating reactors is fifty-one; 49 Light Water Reactors, one Gas Cooled Reactor and one Advanced Thermal Reactor (Heavy Water Moderated Light Water Cooled Reactor).

*1 SIRC: The Safety Information Research Center was established on October 1984 as the neutral and public organization, that collects, manages, analyses and evaluates the safety related information using the sophisticated information processing technology, based on the report of the Committee on Information Sophistication of Nuclear Power Generation that was set up in the Agency of Natural Resources and Energy of the Ministry of International Trade and Industry.

ACTIVITIES to NOW

(a) Data Collection

All NEA1, NEA2 and NEA3 data from FY 1990 to FY 1994 were collected from 10 electric power companies and PNC, and transferred to the European Regional Technical Center at the CEPN in France.

(b) Distribution of Information

- Asian Regional Technical Center issued Information Sheets No. 1 and No. 2 in October 1995 that contained the graphs of Japanese occupational exposure trends, by reactor type, and also the periodical inspection dose trends by reactor type.
- Third Annual Report and Information Sheets were translated to Japanese, and distributed to the participating Japanese authorities and utilities.

(c) Software developments for data collection and retrieval

- The Data Collection Software

  The Asian Technical Center developed NEA1 data input module for Japanese data collection with the commercial software (MAGIC) in FY 1992 and modified every year according to the change of the DIF file format.

- The Data Retrieval Software

  The AsianTechnical Center developed the data retrieval system for selected NEA1 data with commercial software of PARADOX for Windows in order to enhance the typical retrieval function and distributed the database system using a Run-Time version to the Japanese utilities for better understanding of ISOE data base.

PROGRAMME of WORK for FUTURE

(a) Participation

The Republic of Korea shows the interest in participating in ISOE and is expected to join in the near future through the Asian Regional Technical Center.

(b) Data Collection and Exchange with the other Technical Center

The Asian Technical Center will collect for every year all of ISOE data from Asian region and transfer the collected data to CEPN with DIF files

(c) Distribution of Information

The Asian Technical Center will issue the Information Sheets to introduce the dose control status, the dose reduction activities and so on inside Asian region. Also, the annual report and Information Sheets from other Technical Centers will be distributed to members and translated to Japanese for the Japanese participating Authorities and Utilities.

(d) Software Development

- The Data Collection Software

  The data input module will be modified every year according to the change of data collection format, the addition of new term or so.

- The Data Retrieval Software

  The data retrieval system for selected NEA1 data will be modified using the commercial software of PARADOX for Windows in order to distribute the more sophisticated analysis tool for utilities.

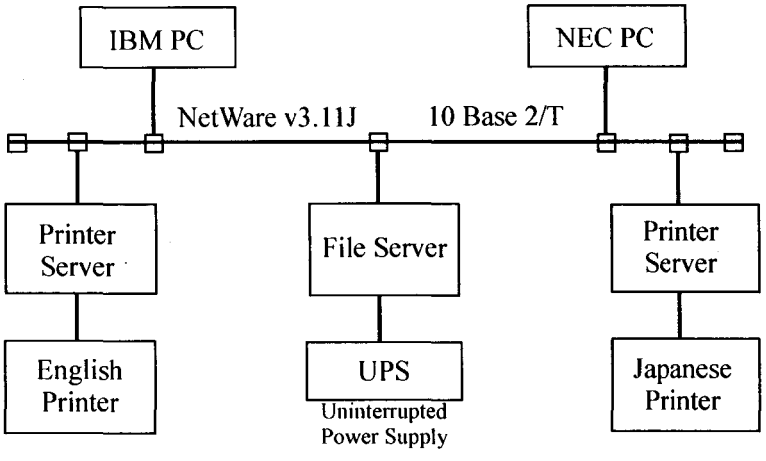

Figure 1. The Computer System of Asian Regional Technical Center

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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FOR OFFICIAL USE ONLY
Abstract No. 90270
Receipt Eingang-TAK
Author 20226 27. Juli 1995
Acceptance
Mini-Presentation

PAPER TITLE THE INTERNATIONAL ISOE PROGRAMME
ISOE North American Technical Centre Activities

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ABSTRACT (See instructions overleaf)

Officially opened in 1993, the ISOE North American Technical Centre has become the focal point for ISOE data collection and distribution for Participants in the United States, Canada, and Mexico. Established at the Nuclear Engineering Department of the University of Illinois, sponsored by the American Health Physics Society's Power Reactor Board, the Technical Centre is overseen by an appointed Board of professional utility radiation protection managers from the United States, Canada and Mexico. This Technical Centre is actively soliciting the participation of all American Utilities (all Canadian and Mexican Nuclear Utilities already fully participate). The Technical Centre is developing new and innovative ISOE products to promote timely information exchange among participants, especially studies on reduction of station dose due to shorter refuelling outages, and analysing the impact of shorter refuelling analyses on station person sieverts. Refinements in work selection and planning derived from European plant experience exchanges have resulted in significant dose and cost reductions in 1994 and 95 in some US plants. For example, in addition to standard ISOE data management, the North American Technical Centre is developing quarterly updates of year-to-date collective exposure, both projected and actual. Other services, such as informal information exchange visits among participating radiation protection managers, have also been accomplished with positive results. This presentation will focus on the current activities of the ISOE North American Technical Centre, and on its plans for future activities.

THE INTERNATIONAL ISOE PROGRAMME

ISOE EUROPEAN TECHNICAL CENTRE ACTIVITIES

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ABSTRACT

The CEPN has been involved from the beginning in the NEA Information System on Occupational Exposure (ISOE). As ISOE European Technical Centre, the CEPN is responsible for the collection and distribution of European data to the various ISOE partners, the collection of data from the other Regional Technical Centres, and the custodianship of the full ISOE data base. This full data base is updated annually and distributed to all the European participating utilities and to other Regional Technical Centres. The European data are also sent to the European Commission (EC) as a contribution to their own occupational exposure data base.

In addition to these activities, the CEPN has developed two computer programmes under Windows to facilitate the management of ISOE data bases. The first software, called ASPIC, allows participating utilities to electronically complete NEA1 and NEA3 questionnaires, as well as to consult the NEA3 data base using key words search routine. This software is available in six different languages. The second programme is a relational data base for the performance of statistical analyses using data from NEA1.

The CEPN also contributes significantly to the efficient flow of information through the ISOE Network. This includes participation in the preparation and distribution of the ISOE Annual Report (which presents and analyses NEA1 data), ISOE Information Sheets (providing short synthesis using NEA1 and NEA3 data), and Technical Reports. Participating Authorities and Utilities have each made information requests to the CEPN concerning dosimetric data, radiation protection experiences, policies and practices. Some examples of these requests, and of the types of data provided, will be presented.

INTRODUCTION

The CEPN has been involved since its inception, with the NEA Information System on Occupational Exposure (ISOE), supporting the OECD Nuclear Energy Agency in promoting the ISOE system and developing computerised tools to accelerate the dissemination of operational experience, and to facilitate the exchange of first-hand information between utilities on specific dosimetric problems and new maintenance operations. Since the launching of ISOE in 1992, the CEPN has acted as the ISOE Regional Technical Centre for the European countries, and has been charged with the operational management of the ISOE data base. In addition, the CEPN contributes significantly to the efficient flow of information through the ISOE Network.

PROVIDING ISOE PARTICIPANTS WITH SOFTWARE

The CEPN has provided ISOE participating utilities with two computer programmes, under Windows, for ISOE data collection, data storage and ISOE data search and analysis. These programmes are also available to participating authorities, but without data base.

The first programme, called ASPIC[1], allows participating utilities to input NEA1 and NEA3 Questionnaires data corresponding to the agreed data collection formats. ASPIC also allows the consultation of the NEA3 data base, by the use of key words, to identify the NEA3 specific reports which are of interest to the participants. This programme is available in six different languages of the European Community (English, French, German, Spanish, Italian, Dutch). The annual collection of NEA1 and NEA3 data, performed using ASPIC, allows the direct transfer of data from the participant to the data base, eliminating the need for manual transfer and the associated time consuming verification step. The ASPIC Software is now used by all the participating utilities in Europe, in Canada, in Mexico and in the ISOE participating Non-OECD Countries. Moreover, the authorities in Sweden, Germany and Spain collect their respective national data using the ISOE NEA1 Questionnaire.

Furthermore, an additional tool is being developed to provide participants with the ability to perform statistical analysis using data from NEA1 Questionnaires. This new software will use a data base management system (DBMS), such as the Microsoft Access DBMS, which allows the exchange of generic features between Users and Regional Technical Centres such as standard data requests or reports, formats, as well as the exchange of data with other commercial products (such as Excel, Paradox, dBase, etc...). It also gives users the possibility of using the programme in a Network Environment, and the benefit of evolutions to come.

MANAGING THE OPERATION OF THE ISOE DATA BASE

The CEPN, as ISOE European Technical Centre, is responsible for the collection of information, quality control of data, storage and protection of information, and retrieval service in Europe. In addition to these activities, which characterise each of the ISOE Regional Technical Centres, the CEPN is also responsible for receiving occupational exposure data from other Regional Technical Centres, and the custodianship of the full ISOE data base. This means that the CEPN has to set-up and maintain an up-to-date whole data base on the various levels of information. This full data base is updated annually and distributed to all the European participating utilities and to other Regional Technical Centres. Moreover, an agreement has been concluded between the European Commission (EC) and the NEA which states that the CEPN is responsible for the collection of NEA1 data from European utilities and for supplying all these data from Europe to the EC as a contribution to their own occupational exposure data base.

The dissemination of information is one of the most important task of the European Technical Centre. For that purpose, the CEPN works in close collaboration with other ISOE Regional Technical Centres in order to guarantee to the participants the access to continuously updated information and to put the updated computer programmes at users' disposal.

FACILITATING INFORMATION EXCHANGES THROUGH THE ISOE NETWORK

The CEPN is deeply involved in the ISOE Network, which facilitates the rapid exchange of operational dose reduction experiences and good practices, and contributes to efficient transfer of information by participating

1 : ASPIC is a Windows based Software System for input and retrieval of data. ASPIC is available, on request, from the Regional Technical Centres or directly from the NEA ISOE Secretary.

actively to the preparation and the distribution of documents such as the annual ISOE Report (which presents and analyses NEA1 data) [1 to 4] or Technical Reports [5]. The CEPN also provides the participants with ISOE Information Sheets. For example, the CEPN provides each year in May an ISOE Information Sheet presenting the first European dosimetric results including data from the previous year for general distribution to all European participants and other ISOE Technical Centres. The CEPN also provides an ISOE Information Sheet, having a restricted distribution to utilities, presenting the NEA3 Questionnaires received during the year. ISOE Information Sheets may also provide summaries of data from both the NEA1 and NEA3 data bases, as the ISOE Information Sheet on steam generator replacements.

European participants interested in the experience of others in specific areas not yet covered by NEA3 Questionnaires, have made requests to the CEPN to solicit the needed information. Both participating authorities and utilities have made information requests concerning data, experiences, policies, practices,... Recent examples of the use of this network system have included a utility's request for information concerning the decontamination of the residual heat removal (RHR) system for the replacement of an RHR heat exchanger channel head; an authority's request for information concerning the dosimetric impact of vessel head inspections in France, Switzerland, Sweden and Belgium; a utility's request for experience in reactor vessel decontamination; a utility's request for information regarding experience in refuelling-pool decontamination and, a utility's request for experience in the repair of fuel storage rack anti-seismic snubbers. In all these cases, the European Technical Centre questioned the ISOE Network and the resulting information collected from participating utilities was passed on to the requester within a very short period , and was then available to all other participants.

CONCLUSION

Since the beginning of the CEPN participation in ISOE, the number of the CEPN members involved in the data base management and communication network is regularly increasing, having reached in 1995 seven different members on part time basis. So far, the European Technical Centre mainly focused its efforts on preparation of the annual ISOE Report, data management (expansion of the ISOE data base to GCR and definitively shutdown reactors) and software development. The NEA3 data base is a way to facilitate efficiently the exchange of experiences and practices concerning occupational exposure reduction among participants. The ISOE network serves as an open line of communication for the real-time exchange of data, experience, policies, practices, etc. Therefore, for the next years, major efforts will be focused on soliciting actively from ISOE Participants an increasing number of NEA3 Questionnaires. For that purpose, the CEPN will select, in collaboration with the ISOE Steering Group and ISOE participants, topics of general interest for the international community, and will make information requests using the ISOE Network. A syntheses of such information will then be made available to participants as ISOE Information Sheets.

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THE INTERNATIONAL ISOE PROGRAMME
ISOE IAEA Technical Centre Activities

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INTRODUCTION

The objective of the sub-programme on Occupational Radiation Protection in the International Atomic Energy Agency, IAEA, is to promote a harmonized approach to optimizing occupational radiation protection by developing guidelines for controlling radiation doses in the workplace and on current occupational radiation protection techniques. A significant part of this programme is the provision of assistance to developing Member States to bring their radiation safety infrastructure to an appropriate level for the usage of radiation in the State.

In consistence with these objectives the IAEA has been involved with the ISOE Programme from its inception and has contributed actively to its growth. In 1993 an arrangement was agreed between the IAEA and the Nuclear Energy Agency, NEA, by which the IAEA co-sponsors ISOE inviting those IAEA Member States which are not members of the NEA to participate cost-free in the programme.

ISOE MEMBERSHIP THROUGH IAEA

The first countries to join ISOE through the IAEA Technical Centre were China, Czech Republic, Hungary and Mexico. While Mexico is now a member of NEA, the utilities in Brazil, China, Czech Republic, Hungary, Slovenia and South Africa, representing seven power plants with a total of 15 nuclear power reactors are now participating through this co-sponsorship. In addition the regulatory authorities in China, Czech Republic and Slovenia take part. Since the value and usefulness of the system is very much dependent on its coverage of the world's nuclear power plants, those from several other Member States are being solicited to join, some of these have indicated their most probable acceptance of the ISOE terms and conditions.

The IAEA Technical Centre acts as representative of its ISOE Participants on the ISOE Steering Group and according to the recently revised arrangement between the NEA and IAEA the participants in ISOE through IAEA will also send an observer (rotating observers) to the Steering Group meetings. The IAEA ISOE Centre is also disseminating information on related topics and meetings among its members and an ISOE Information Sheet has been issued.

MEETING ON TRAINING AND INFORMATION EXCHANGE

In order to provide its members with information on the ISOE project and to enable guided practical exercises with the database software, the IAEA organized a one-week ISOE Meeting in France in December 1994. The practical arrangements were made by the ISOE European Technical Centre at CEPN, with the active participation from the NEA office. The objective of the meeting, which was attended by eight persons representing China, Czech Republic, Hungary and Mexico, was also to give the participants possibilities to exchange ideas and to gain experience on optimization and the implementation of the ALARA principle in French facilities.

The programme of the first three days included, for example: ALARA: Principle and programme implementation; Work management; ALARA and outage organization; ASPIC Presentation - NEA 1 and NEA 3 exercises; Software presentations.

The participants were also invited to the CETIC training centre and the CRUAS Nuclear Power Plant, where they had the opportunity to share feedback experience with their French colleagues.

The initiative to hold this meeting was very much appreciated by the participants and the general consensus was that it was most successful.

FUTURE PROGRAMME FOR THE IAEA ISOE CENTRE

The IAEA Member States participating in ISOE represent four continents, thus geographically more spread out and, culturally, significantly more diverse than participants affiliated to the other Technical

Centres. Special attention therefore has to be paid to the co-ordination of the activities and the communication between the members. The meeting in France in 1994 was one means of improving the contacts between the members.

In order to advise on the programme for the IAEA ISOE Technical Centre a consultants meeting was convened in October 1995. The consultants took into consideration how to improve the implementation of the optimization principle in the ISOE member plants and authorities, the need for quality assurance of the data input into the data base and the different possibilities the Agency has to support their ISOE members through workshops, training, co-ordinated research programmes and publications, also making use of available software programmes and other tools. The group based their recommendations on their knowledge on the present participants in ISOE through the Agency as well as presumptive future members, such as the Russian Federation and Ukraine.

It was concluded that, as far as ALARA culture and implementation are concerned, the situation is very different from one country to another. Some common needs exist, however, and the IAEA ISOE Centre was recommended to take these into account and adapt them to the individual national contexts.

To facilitate the exchange of feed back experience

A very efficient method of sharing past experience is to arrange seminars for those involved in similar activities. In the radiation protection field, in addition to the ISOE topical meetings and workshops, a number of such seminars are regularly organized, e.g by CEC, Westinghouse, Candu Owners Groups as well as BWR and PWR Owners Groups. Other useful meetings are organized on a less frequent basis by, for example, Brookhaven National Laboratory (BNL), Electric Power Research Institute (EPRI) in the US, and National Radiation Protection Societies.

Before the break down of the former Soviet Union similar seminars were arranged regularly for radiation protection staff of the WWER reactors, but these meetings do not exist any more and the Agency was recommended to promote the exchange of experience between its member plants by initiating the formation of radiation protection groups for WWER and, later, RBMK reactors.

The Agency should also facilitate regular participation of radiation protection staff in such types of groups, and encourage participation in international meetings focused on other types of reactors.

The Consultants Group also recommended that specific working groups composed from users of the same type of reactor should be created to discuss the analysis of data and the usefulness of the ISOE network for that type of reactor as well as basic problems in the field of occupational exposure, evaluation methods, calibration procedures, recording systems, etc.

To ensure the dissemination of an ALARA culture in non-NEA Member States

The distribution of information on ALARA theory and practice is a key element in the worldwide dissemination of the ALARA culture. The lack of application of the ALARA principle in the legislation and practice in some countries, as well as language barriers and geographic isolation are the factors preventing general application of proved international practices. Existing literature and other sources of information are frequently not known for radiation protection staff working in nuclear power plants.

The Agency was advised to provide information on relevant publications and promote the dissemination of ALARA culture, for example by inviting managers to an "ALARA awareness day", organizing scientific visits for qualified experts with special emphasis on practical ALARA methods and tools and organizing ALARA courses at regional or national levels. To facilitate training in different native languages and provide participants with radiation protection optimization training tools, the Agency should also organize training courses for ALARA trainers in English.

To ensure the practical implementation of the optimization principle through the use of ALARA tools.

It was recommended that the Agency collects information about existing ALARA tools and organizes a working group in order to evaluate them. Those tools considered as cost effective by the working group should be provided by the Agency to their ISOE participants.

CONCLUSION

By its co-sponsorship of ISOE the Agency enables non- NEA Member States to obtain access to the database on occupational exposure and to exchange feed back experience with radiation protection staff from nuclear power plants all over the world. The distance, both geographically and culturally, between the IAEA Member States participating in ISOE calls for special efforts in the co-ordination of the activities and the communication between the members. There is no doubt, however, that the Agency can, by disseminating information on internationally accepted radiation protection principles, contribute to a significant improvement in the implementation of the ALARA principle in these nuclear power plants.

OCCUPATIONAL EXPOSURE EVOLUTION IN SPANISH NPP'S

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INTRODUCTION

After extremes fluctuations in the early 1980's, the evolution in occupational collective dose in the operating nuclear power plants in Spain seems to be stabilising and lightly decreasing in the last years, within the same range of the average level of the OECD countries.

It is clear that only a well structured, systematic and practical approach to optimization of protection, as a strict selection process and as a "state of thinking", ensures that no important aspects are overlooked and the appropriate decisions are taken to perform the different tasks in conditions to achieve the optimum level of exposure. During the last decade we can affirm that a progressive development of ALARA culture is having a positive influence in Spanish PWR and BWR.

In this presentation the main indicators of the occupational dose trends to assess the present development level of this criterion, the collective dose per reactor, the task related collective dose, individual doses, etc. are reviewed and the key elements that have shown a clear effectiveness to improve the practical application of it, organizational modifications, increasing commitment, global strategies to reduce the source term, increasing presence of robots, etc. are discussed.

COLLECTIVE DOSE

Occupational collective dose per reactor evolution in the operating LWR's in Spain the scope showing in figure 1.

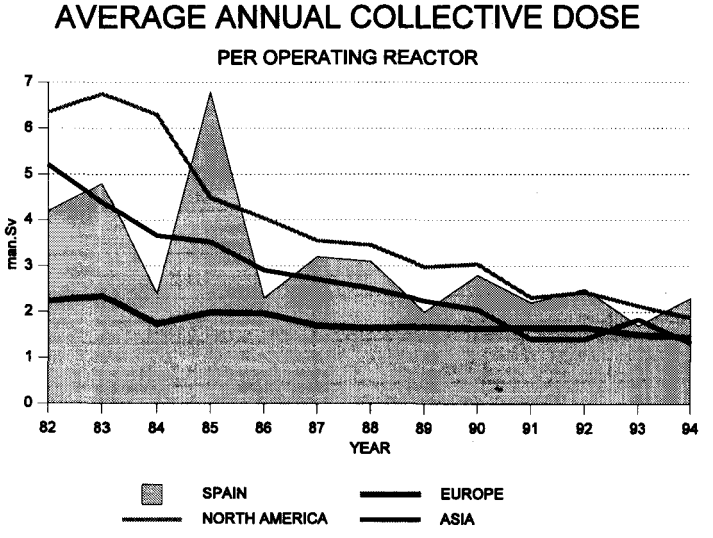

Figure 1.- Annual collective dose per reactor unit for LWRs plants at Spain and different regions.

After extremes fluctuations in the early 1980's, because major modifications to enhance the safety in CN JOSE CABRERA and the recirculation piping system replacement in CN ST^{a} M^{a} GAROÑA took place, the trend in occupational exposure seems to be stabilising last few years within the same range of the average level in the OECD member countries.

Surely, this tendency of the collective occupational dose for Spanish LWR's is however far to reach an ALARA plateau that it is not cost effective to reduce exposure with the current nuclear reactor design and technology.

Comparing the situation for PWR's within the international context, size and age tacking in account, we can see according with the data of the table 1 that while collective occupational exposure at last generation of Spanish PWR's is in a balanced level, otherwise is the behaviour at first generation.

Table 1. Average collective doses in 1993 as a function of reactor size and age for PWR's (man.mSv).

COUNTRY	OLD & SMALL PLANTS	INTERMEDIATE & MEDIUM	MODERN & MEDIUM
SPAIN	3* (01)	1.7 (04)	1 (02)
NORTH AMERICA	1.4 (7)	1.7 (9)	0.9 (7)
ASIA	1.6 (5)	0.7 (1)	0.5 (1)
EUROPE	1.7 (10)	2.3 (18)	-

* Three years average, because the 1993 data is not representative.

NOTE: The values in the parenthesis are the numbers of reactors. The regional data are taken from Ref. 4

Similar terms of comparison can be established for BWR's. Here, the relative situation on occupational exposure is nearly the same for the new plant than for the old one, as can be seen in Table 2.

Table 2. Average annual collective doses in 1993 as a function of reactor size and age for BWR's (man.mSv).

COUNTRY	OLD & SMALL PLANTS	INTERMEDIATE & MEDIUM
SPAIN	4* (1)	4.7 (1)
NORTH AMERICA	3.1 (7)	3.4 (2)
ASIA	2.5 (3)	1 (2)
EUROPE	2.7 (8)	4.7 (1)

* Date of 1994, because the 1993 date is not representative.

NOTE: The values in the parenthesis are the numbers of reactors. The regional data are taken from Ref. 4

TASK RELATED COLLECTIVE DOSE

Operational monitoring systems besides the regulatory individual dosimetry are established in all Spanish LWR's, most of them, except one, are digital and computerising system. This type of dosimetry allows a link to be made between the daily doses received by workers and a set of generic operations taking place during the operation and maintenance of the plants.

Following the recollected data up to date, some jobs involving higher average collective doses in PWR's are represented in figure 2.

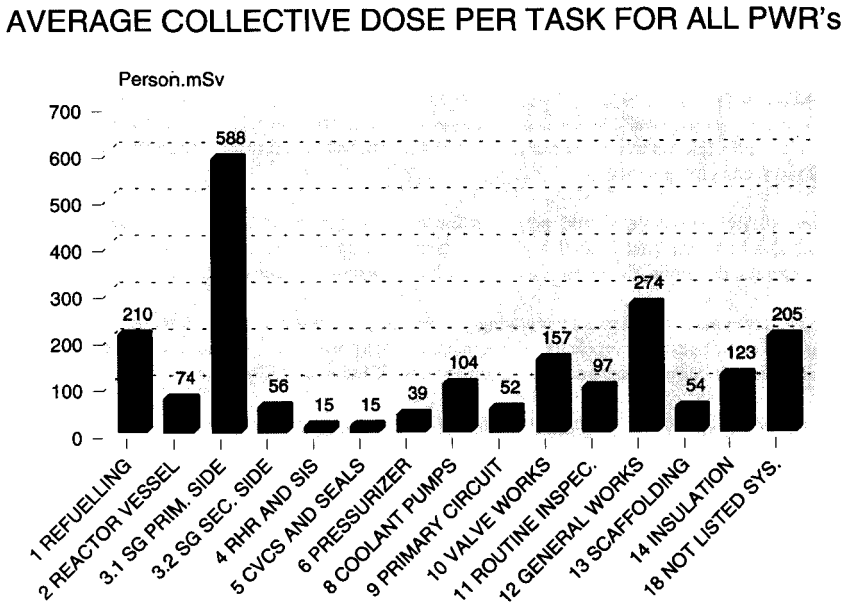

Fig. 2. Average collective dose per task for Spanish PWR's in the period 1988-92.

INDIVIDUAL DOSES

The average individual dose is also a global indicator that is often used to assess the performance of nuclear power plant operation in relation with radiological protection. Only slight variation between the reactor types exists, the international average being about 3 mSv per year. In Spain, this parameter is stabilising on the same order last five years, in 1994 the value was 2.27 mSv. Anyway one should be aware of when using this indicator, which different criteria are used in different countries to record and report doses and it is not a solid basis for firm conclusions concerning the general radiological situation.

In order to approach the previous situation in Spain to face in the early future the new dose limits recommended by the ICRP, more explicit results the evolution of the individual dose distribution, specially to define the size of the critical groups of workers, typically mechanical maintenance workers.

According with the 1994 data, 85% of workers have received less than 5 mSv, only 3% are over 15 mSv and 1.5 % workers exceeding 20 mSv/year, i.e.140 workers over 9,053.

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ALARA application IN SPANISH NPP'S. NEW APPROACH.

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INTRODUCTION

One main objective of all nuclear installations is to underlie the Principle of Optimisation. The focus in the application of this principle varies considerably, from Cost-benefit techniques in the design phases to Commitment, attitude and motivation in operation.

The ALARA principle should permeate the entire organisation of the company during operation. Management should be the first link of a chain originating with explicit corporate commitment, leading to increasingly specific decisions, finally to reach those who are responsible for actual performance of each task.

The safety culture is an element integrating the mentality that is to prevail throughout the organisation. The organisation shall contribute to incorporating the principle of optimisation in all plant operations and all phases of activity. Management's commitment and motivation are transferred to all the workers and organisations involved.

The ALARA commitment naturally extends to all outside organisations.

The ultimate objective is to reduce collective doses through the promotion and enforcing of an ALARA culture and compliance with the operating collective dose objectives.

PREVIOUS PRACTICAL APPLICATION

The optimization criterion was carried out in the 80's in Spanish NPP's trough the "Dose Reduction" programmes, characterized by:

- Responsibility delegated to the Head of the RP Department.
- Specific structures for ALARA reviews of refuelling tasks.
- Occasionally the Plant Nuclear Safety Committee involved in ALARA proposals.
- No systematic approach relating to the source term, design modifications or the role of external companies.

THE STRATEGY

In order to improve the effectiveness of the practical application of the optimisation principle, the CSN established in 1991 a new strategy, in cooperation with the utilities, to change the way of carrying out this criterion from the formerly through the global Approach.

Explicit responsibilities and commitment at certain management level and at key personnel of other sections, the awareness of personnel by ALARA training beyond general radiological protection indoctrination, a specific management structure, the direct implication of contractors companies, the definition of general plans to reduce the source term, these are some main features of the ALARA programs to strengthen their operating efficiency and in which ALARA is regarded primarily as a "state of thinking" that pervade the various levels of management and workforce.

RESULTS

During 1994 most of the plants have changed radically their organization and, what is more relevant, their mentalities to adapt to this approach. The results in collective doses in that period suggest that these organizational changes are not only-formal but effective.

The terms under which these programs are going to be reviewed are commitment, promotion of ALARA culture, administrative system, scope and structure.

COMMITMENT

- Specific ALARA responsibilities.
- ALARA policy and collective dose objectives.
- The principle that responsibility is shared by all.
- The ALARA policy affects contractor companies.

PROMOTION OF THE ALARA CULTURE

- Talks and presentations at plant management levels.
- The ALARA philosophy in the basic training on RP.
- ALARA courses to the design, engineering, mechanical maintenance, RP, etc. personnel.
- Open and general information.
- Dose budgeting for different plant departments and at certain plants also for the different contractors.

ADMINISTRATIVE SYSTEM

- The ALARA organisation and responsibilities included in the Operating Regulations.
- The procedures developing the ALARA program are general administrative procedures.
- The ALARA procedures are included in the QA Manual as auditable documents.
- The drawing up of minutes for meetings held by:

  Owner's NS Committee

  Plant ALARA Committee

  ALARA task forces

SCOPE

The scope of the ALARA programmes includes:

- Normal operating activities and scheduled and non-scheduled shutdown.
- Initiatives aimed at reducing the source term.
- Revision of design modifications having a potential positive or negative impact on the collective dose objectives.

STRUCTURE

- Interdisciplinary company management Committee.
- Plant management.
- Interdisciplinary plant management Committee.

- Head of RP and/or ALARA Engineer or Coordinator.
- Interdisciplinary task performance group.
- ALARA Coordinators.

CONCLUSIONS

In view of the results presented above, it may be stated that we are achieving the objective of undertaking practical application of the ALARA criterion with wider scope, ALARA as one element of safety culture and total quality, based on personnel motivation and involvement, supported by an organisational structure, set up adequate forums for a systematic approach to optimised operation of nuclear power plants.

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THE PRACTICAL ACHIEVEMENT OF RADIOLOGICAL DESIGN CRITERIA THROUGH COST EFFECTIVE DESIGN

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INTRODUCTION

BNFL has for many years been designing, building and operating nuclear facilities at its various sites. The Consultancy Services division is responsible for ensuring that the design of these facilities achieves all appropriate radiological safety criteria. It is essential for the Company's commercial interests that this high level of safety is obtained at minimum lifetime cost. The Company has recently reviewed and revised its internal design criteria for occupational radiation exposures, making them more restrictive following an international trend of reduced dose limits. At the same time, the assessment methodology by which compliance with these criteria is demonstrated at the design stage, has been revised to improve its relevance to the actual operational situation. Thereby, it should be possible to achieve a greater standard of safety whilst keeping costs down.

This paper discusses the achievement, at minimum cost, of design criteria for radiation exposure, by reference to examples of several different nuclear facilities.

DESIGN DOSE CRITERIA

BNFL operates a voluntary annual individual dose restriction for its workforce of 20 mSv per year, with a maximum of 75 mSv over 5 years, for all existing facilities. More recent facilities were designed to lower dose targets, which essentially limit the maximum individual whole body exposure to 15 mSv per year and the average exposure to the workforce on a facility to 5 mSv per year. Following the publication of ICRP60 and other developments, such as the introduction by the UK Nuclear Installations Inspectorate (NII) of the Safety Assessment Principles (SAPs), BNFL have reviewed the design criteria for new facilities. As a result of this review, future facilities are being designed to dose targets that restrict the maximum individual exposure to 10 mSv per year. The design target for average exposure to the workforce is not changed at 5 mSv per year.

In addition, there is always an over-riding requirement to demonstrate that all assessed doses are As Low As Reasonably Practicable (ALARP).

METHODS OF DESIGN DOSE ASSESSMENT

There are many techniques available to analysts in Consultancy Services to ensure that the design criteria for radiation exposure are achieved at minimum cost. These are based around analytical techniques, computational methods and a broad base of experience of the design, construction and operation of nuclear facilities. The experience of individual analysts, coupled with the job control procedures ensure that appropriate tools are used for each task. Invariably, the design assessment includes an estimate of dose, which has two primary functions:

1. to provide, with a very high level of confidence, evidence to Company Safety Committees and external Regulators that demonstrates that the proposed facility can be reasonably operated to achieve relevant dose targets and ALARP requirements;
2. to give a realistic indication of how and where doses will be received within the facility, with the intention of aiding optimisation of the engineered radiological protection and operational dose controls and restrictions.

The first of these purposes is served by producing a demonstrably robust estimate which will, in practice, over-predict the actual doses that will be received in the facility. The conservatisms in this estimate need only be reduced as far as is needed to confirm that the doses will be below the dose target. However, this conservatism

will often obscure the likely actual doses within the facility. If the designer is not careful, large amounts of money may then be spent unnecessarily on engineered solutions to purely theoretical dose problems (for example, shielding may be built to a standard dictated by a theoretical dose that may never occur in practice). More importantly, the designer could be deflected from targeting limited resources on the areas where the greatest real dose, and hence potential benefit, is concentrated. Therefore, the derivation of a more accurate dose estimate may also be appropriate.

DOSE ASSESSMENT FOR A MIXED OXIDE FUEL FABRICATION FACILITY

A new facility for the production of Mixed Oxide (MOX) fuel is currently under construction at BNFL's Sellafield site. Early calculational assessment, along with operational experience from a similar, smaller-scale facility already in operation, indicated that dose uptake would need to be carefully considered during the design process in order that dose criteria would be met, mainly due to the significant neutron component. Consultancy Services therefore initiated dose and ALARP reviews to identify areas where dose could be reduced most effectively. These reviews were performed against the initial design and have been ongoing throughout the project. They utilised the experience of designers, plant operators and Health Physicists as well as Consultancy Services staff, and concentrated on areas of high predicted dose. The outcome led to many significant modifications, including changes in the design, operating and maintenance philosophies such that significant reductions to the dose estimate have been made and the dose criteria will be met.

Specific items covered during the review process were :

- Whether the cost of automating particular high dose operations could be justified;
- Plant and room layouts to reduce the occupancy in areas of high ambient dose rate or to optimise the shielding requirements for such rooms;
- The use of transparent shielding on some gloveboxes to aid in-box visibility and hence reduce the time taken to perform dose intensive operations; removable shielding panels were added where this would lead to reductions in ambient dose rates but still enable access for maintenance operations;
- Reduction in dose for maintenance operations by ensuring bulk sources could be removed prior to maintenance where practicable, locating maintainable items in low dose rate fields (e.g. external to gloveboxes) and designing these to be quicker to maintain or change. This has had the added benefit of reducing the time necessary for maintenance and hence increased the plant's operational capability.

As the MOX facility project is progressing to very tight timescales it has been important in terms of both time and cost to minimise the number of modifications necessary to the facility following completion of the initial design. Hence, by reviewing and optimising the methods of dose control from the preliminary stages of design, before detailed design work was undertaken, Consultancy Services have enabled substantial, but necessary, dose savings to be made in the most cost-effective manner, with the added benefits of improving the actual operability and maintainability of the facility.

DOSE ASSESSMENT FOR A WASTE VITRIFICATION FACILITY

Many new facilities being built today are similar at least in some respects to existing facilities. Where appropriate, therefore, Consultancy Services have been able to take advantage of existing plant experience to enable fit for purpose designs to be developed for our customers. An extreme example of the use of operational plant experience is the design recently of an additional Waste Vitrification facility at Sellafield. As the design of the new facility is based on an existing Vitrification facility, a detailed review of the exposures on the existing facility has enabled optimisation of the radiation protection design for the new facility. The design dose assessment was performed by considering the features of the new facility that would cause dose uptake to be different from that measured on the existing one. Such features include differences in the source data, changes to the design of high maintenance items, optimisation of shielding thicknesses etc. These factors were used to scale the measured doses to produce design dose estimates for the new facility.

DOSE ASSESSMENT FOR A DECOMMISSIONING PROJECT

In addition to new facility design, Consultancy Services are also involved in projects involving the decommissioning of existing facilities. An example of this is the emptying of corroded fuel cladding from a silo storage facility at Sellafield. Structural considerations have placed weight restrictions on the facility which leads to limits on the amount of shielding that can be used. Additionally, the operating areas on top of the silos are in high background dose rate fields due to significant amounts of contamination.

Optioneering studies involving Consultancy Services, Health Physics, plant operators and design office staff have taken place in order to determine the best options for emptying the silo contents. Dose and ALARP have been major considerations in these studies, whilst operational requirements and structural restrictions have also been taken into account. Some of the areas addressed by the studies are considered below. Dose assessments have been performed throughout the design of the plant. Measured dose rate data from the silo facility, and other plant at Sellafield handling similar material, have been utilised in addition to theoretical dose rate assessments to ensure that dose estimates are as realistic as practicable.

- Detailed studies were undertaken to examine options for constructing the silo emptying machine either in-situ or outside the building. The resultant solution ended up with minimum dose uptake and also lowest overall cost by bringing the machine into the building in modular form and constructing on the silo top.
- Complex computer modelling has been used to optimise the effectiveness of shielding on the machine in terms of positioning, thicknesses and use of different materials.
- Dose surveys have been performed in the existing building to identify contamination sources leading to high dose rates. Subsequently, methods of dose reduction have been considered, including removal of redundant or contaminated equipment, the use of localised shielding and decontamination of the building floor.

ASSESSMENT METHODOLOGY DEVELOPMENT

To improve the way in which we produce design dose assessments, BNFL is investing in research and development covering a variety of issues from the collation and use in design of operational data to analysis of the lifetime costs associated with radiation protection. The primary aim of this research is to provide the design analyst with information and analysis tools that will enable him to practically derive a realistic indication of dose. He must be able to produce this estimate speedily and at optimum cost, in order to meet the requirements of the typical design programme, which in turn means that he must have the relevant information readily available to him. Consequently, much of the development is concerned with collating and databasing such information.

The main areas are as follows:

- Analysis of design engineering and operating costs of radiation protection.
- Collation of empirical dose, dose rate and operational data from operating facilities.
- Collation of theoretical dose and dose rate data from historic design analyses.
- Development of methodologies for the use of the collated information.
- Review and development of methods for predicting internal doses from inhaled radio-isotopes.
- Review and development of dose rate calculational methods.

CONCLUSION

The estimation of dose to the workforce is a necessary part of the process of design for facilities handling radioactive materials. At BNFL a range of design dose assessment techniques are used, from theoretical evaluation of calculated dose rates and predicted occupancy durations, to comparison with existing, operating facilities and measured doses.

The production of a realistic dose estimate allows it to be used to optimise the radiation protection design of the facility, and thereby keep design costs to a minimum. Research and development is currently being undertaken to provide tools that will enable the design analyst to produce more realistic estimates within the constraints of the design process.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE SAFETY CULTURE ASPECTS OF MANAGING FOR SAFETY - EXPERIENCE OF A LARGE NUCLEAR REPROCESSING SITE

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The paper will address how elements of a safety culture were identified and reviewed on the BNFL Sellafield Site in the United Kingdom. Sellafield Site is BNFL's largest site with Magnox reprocessing plants, the new Thermal Oxide reprocessing (THORP), waste treatment and storage facilities, and four of the company's Magnox reactors. With reference to the large workforce comprising of such diverse range of activities, the paper will address the dynamic nature of safety culture, both in the formation of new cultures for new plants, and the shaping/development of safety culture within existing teams and their interactions. The paper will describe how the shape of the existing safety culture with its emphasis on radiological safety was established, describing the methodologies employed and the practical considerations of carrying out and interpreting the findings of such studies. The paper will also describe the ways in which that knowledge was disseminated for use by management to reshape our safety culture with the objectives of maintaining a high standard of safety performance (both radiological and conventional), and achieving continuous improvement through and by the workforce.

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FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE OPTIMISATION OF RADIOLOGICAL PROTECTION AT THE DESIGN STAGE OF NUCLEAR INSTALLATIONS: RECENT EXPERIENCES WITH ALARA TOOLS

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2 (see page 7)

ABSTRACT (See instructions overleaf)

Implementing the optimisation principle at the design stage of nuclear installations involves to maintain the collective exposures as low as reasonably achievable keeping at the same time a fair distribution of individual doses among exposed groups (equity principle). This general objective can only be achieved by anticipating from the very start of the design phase, not only the doses that can be received by workers but also how the levels of exposure will be distributed among them.

The assessment of workers exposures heavily rely on the feedback experience which provide necessary data as well as on specific tools to estimate dose rates and to combine all relevant exposure parameters. The detailed radiological analysis of technical options must be done through source-terms identification and assessment, the description of future operating modes, the prediction of incidents, the knowledge of workers locations and physical characteristics of the conceivable biological shielding. Then, all these data must be integrated with appropriate tools to *a priori* determine the most exposed areas, important sources, efficient protections and constraining tasks.

These tools devoted to the evaluation and selection of technical options at the design stage must be compatible with complex sensitivity analyses to properly quantify the efficiency of possible corrective actions of protection (e.g. reinforcement of biological shielding, withdrawal and specific shielding of sources, moving workers away from " hot spots " using robotics tools or remote controls...).

Methods and adapted tools developed during recent years at CEPN to help nuclear designers will be presented in the poster and in the corresponding paper.

Requirements and recommendations for cold shutdowns of nuclear power plants

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Within the framework of the optimisation of the shutdown of units, a procedure for moving to cold shutdown aimed at reconciling the length of shutdowns with dosimetry and the control and limitation of the production of waste is being extended to all 54 PWR reactors being operated by EDF. This procedure, which has been in place since 1984, has developed in the light of the feedback of experience gained.

1 - Aims of the shutdown procedure

The modification of the physical and chemical characteristics of the primary coolant during cold shutdown for refuelling (drop in temperature and pH, going from a reducing to an oxidising environment) leads to a considerable increase in the activity concentration of the primary coolant in terms of corrosion products (see Table 1).

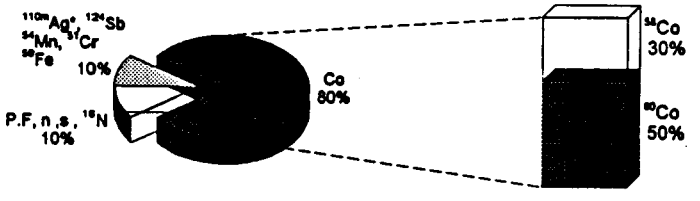
^{58}Co	^{60}Co	^{124}Sb	^{110m}Ag
150	4	10	2

Table 1: Activity values (GBq/t) at the oxygenation peak

In addition, the de-pressurisation of the primary cooling system may result in fission products (xenon, iodine, etc...) passing into the primary coolant and this may pose problems for the protection of workers when the primary cooling system is opened up.

The main objectives of the cold shutdown procedure are as follows:

- to reduce the activity of the primary coolant as quickly as possible and to avoid the contamination of out of flux areas through the deposition of active corrosion products on items of equipment. This is a very important point, since the active corrosion products (^{60}Co and ^{58}Co) are responsible for approximately 90% of integrated doses during maintenance operations (see Figure 1).
- to limit the risk of contamination of the atmosphere in the reactor building during the opening of the primary cooling system (volatile fission products),
- to limit the production of waste and to comply with stack release criteria,
- to facilitate fuel handling operations by ensuring the transparency of the water and by reducing the dose rate at the surface of the reactor cavity.



* impact of ^{110m}Ag varies greatly from unit to unit
n = neutrons, s = active structures

Figure 1: Contribution of corrosion products to collective shutdown dosimetry

2 - Oxygenation of the primary coolant

Since the oxygenation of the primary coolant during cold shutdown for refuelling is inevitable, it was decided to control this operation by provoking it deliberately. The activity is released quickly which enable the primary coolant to be purified efficiently.

This operation has evolved as follows:

- from the start up of unit up until 1994: aeration through scavenging the air of the CVCS tank from a temperature of 120°C
- since 1994: aeration through the injection of hydrogen peroxide at a temperature of 80°C.

In both cases, the temperature is reduced to the oxygenation temperature as quickly as possible (in accordance with technical operating specifications) and continues during the oxygenation phase.

A qualification program has been carried out before the use of hydrogen peroxide in PWR plants, in order to ensure that this reagent did not affect the materials used (Stellites in particular) and that the primary cooling system was not contaminated.

The use of hydrogen peroxide has the following main advantages:

- saving in time: the oxygenation phase last 0.5 hours instead from between 5 hours (lack of fuel cladding defects) and 10 hours (presence of fuel cladding defects).
  Where oxygenation is carried out in the air, the scavenging rate (and therefore the speed of oxygenation) is limited by compliance with stack activity limits (in order to remain below the pre-alarm threshold of $4.10^5 Bq/m^3$)
- improved control over oxygenation, particularly in the presence of fuel cladding defects and therefore improved control over the risk of contamination of out-of-flux equipment by active corrosion products.

3 - Radiochemical specifications adopted and justification

In order to achieve the objectives set out above, thresholds were established for the various steps of the cold shutdown of the reactor.
These limits are set out in Table 2 and the justification for their application is as follows:

- **Hydrogen**
  The objective is to remove hydrogen from the primary cooling circuit in order to prevent dangerous hydrogen-oxygen mixtures.

- **Xenon and Iodine**
  The limits concern ^{133}Xe and ^{131}I and are intended to limit:
  - gas releases,
  - the risk of contamination of the reactor building when the primary cooling circuit is opened.

- **^{58}Co and γ_{total} activity**
  The objective is to ensure adequate purification of the primary coolant and to limit the risk of contamination of out-of-flux equipment by active corrosion products.
  These limits concern ^{58}Co, which is the most abundant radionuclide and γ_{total} activity which enables the presence of other radionuclides, such as antimony and silver, to be taken into account.

SHUTDOWN				
TRANSIENT STAGE	With RCS aeration			Without RCS aeration
	With reactor cavity filling	With RCS opening and without reactor cavity filling	Without RCS opening	
Load reduction	H2 : 5 ml/kg TPN Xe133 < 8 GBq/t I131 < 2 GBq/t			
Borication				H2 : Chemical spec. 2ppm<LiOH<2,2ppm B mini fixed by STE
Aeration	H2<3 ml/kg TPN et H2<2% gas phase of closed vessels Xe133 < 4 GBq/t I131 < 4 GBq/t			
RCS pumps shutdown	Co58 < 80 GBq/t Co58 < 50 GBq/t Υt < 160 GBq/t Υt < 100 GBq/t			
RCS opening	Xe133 < 1 GBq/t I131 < 0,05 GBq/t			
Reactor cavity filling	Co58 < 2 GBq/t Υt < 4 GBq/t			
Temperature increase		H2 fixed by chemical specifications Co58 < 7GBq/t Υt < 14 GBq/t		

All measurements are performed in liquid phase of primary circuit

γt values before RCS pumps shutdown and reactor cavity filling are depending on γt activity in reactor cavity tank

X *Limit*

X *Guideline*

Tableau 2 : Radiochemical limits for cold shutdown of PWR plants

COMPUTERISED DOSIMETRY MANAGEMENT SYSTEMS WITHIN EDF

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EDF, using the ALARA approach, has embarked an ambitious project of optimising the doses received in its power plants. In directing its choice of actions and the effectiveness of such actions, the French operator is using a computerised personal and collective dosimetry management system. This system provides for on-going monitoring of dosimetry at personal, site and unit level or indeed for the entire population of EDF nuclear power plants.

OBJECTIVES

EDF is seeking to improve the personal dosimetry in its power plants, that is to optimise the dose received by all workers, whether EDF employees or those of external contractors, with the greatest benefit going to those works who are most exposed. One of the means to achieving this objective is to reduce the collective dosimetry on sites, which implies the implementation of an experience feedback system: forecasting, monitoring results and analysis of discrepancies.

RESOURCES

To direct its activities and evaluate the effectiveness of such activities, EDF uses systems which continuously update the dosimetry situation of each individual and the collective dosimetry of each site. The doses managed by these systems are the operational doses, of both EDF employees and external contractor personnel, recorded on leaving controlled areas (areas of exposure to ionising radiation).

Given the millions of doses and the tens of thousands of people concerned (6000 to 7000 additional persons per year), the system is fully automated but it is designed so that man remains in control of important decisions and is able to make corrections as necessary. The complete system has two main components; the local system on each site and the national "DOSINAT" system, which consolidates the data from the various local system (See Figure 1). These systems have devices which ensure that only authorised persons have access to the data. They are endorsed by the CNIL[1].

LOCAL SYSTEMS

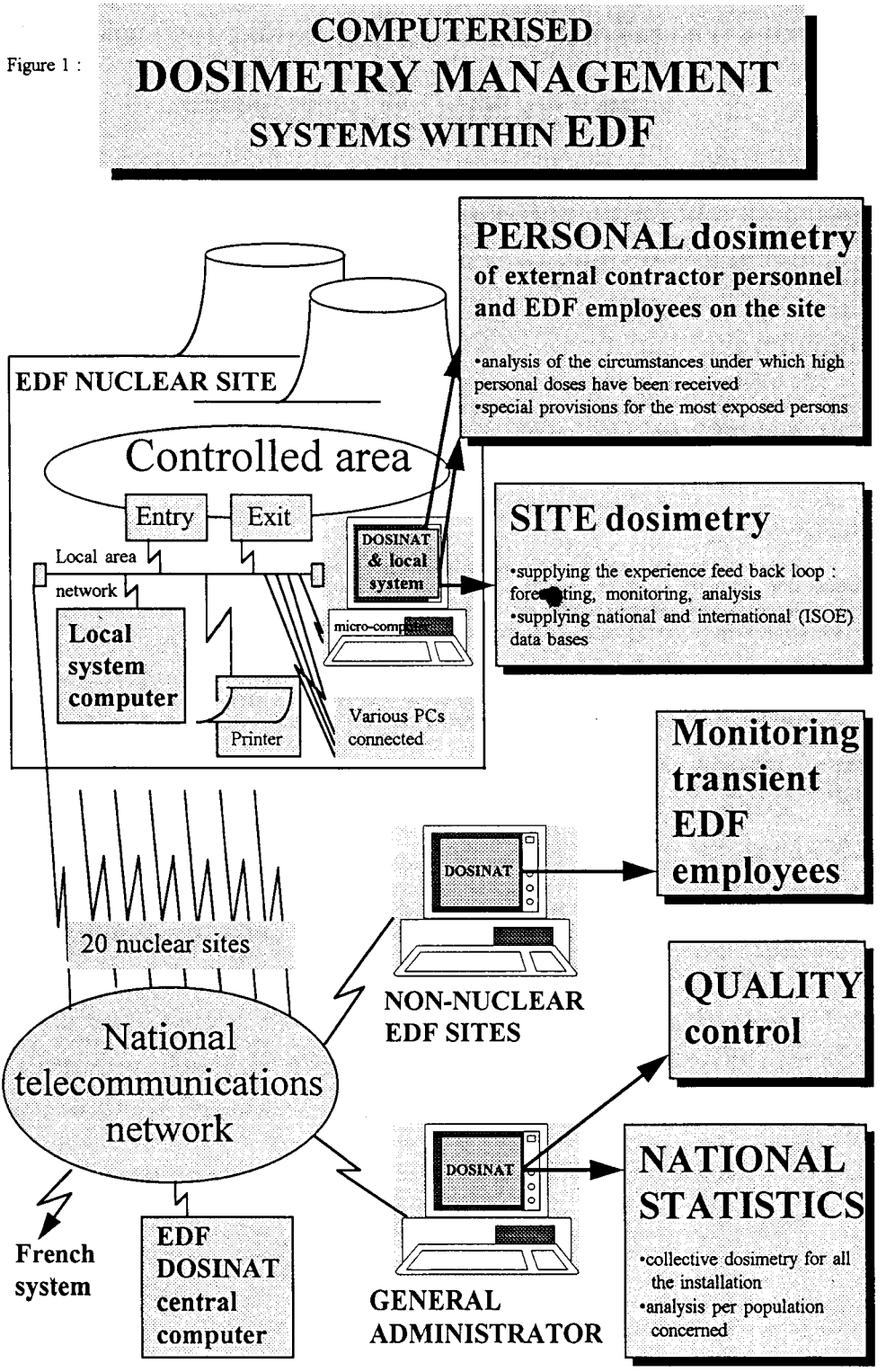
At each entry to or exit from a controlled area, the gates transmit batches of data, known as "movements" to the local system. For a given worker, the "entry movement"-"exit movement" pair represents a dose movement, which takes into account the length of time the work is in the controlled area and which indicates in particular: the identity of the worker, the site on which he is working, the dose he has received, the time of the operation, etc...).

The local system which receives these data is capable of supplementing them with data gathered when the worker arrived at the site concerned, such as: personal details (name, first name, date and place of birth, sex), his employer, his speciality, his training, his qualifications, etc. In addition, the local system receives the personal dosimetry data for the worker over several months from the national system and which is compiled from the data from the EDF sites at which he has worked (in 1994, some 7000 people worked on more than one site).

Using all these data, which are updated continuously, we can monitor the situation of each individual and each site.

[1] (French Commission on Computerized data and Freedom)

Figure 1 :


COMPUTERISED
DOSIMETRY MANAGEMENT
SYSTEMS WITHIN EDF
PERSONAL dosimetry
of external contractor personnel and EDF employees on the site
•analysis of the circumstances under which high personal doses have been received
•special provisions for the most exposed persons
EDF NUCLEAR SITE
Controlled area
Entry
Exit
Local area network
Local system computer
DOSINAT & local system
micro-comp
Printer
Various PCs connected
SITE dosimetry
•supplying the experience feed back loop : forecasting, monitoring, analysis
•supplying national and international (ISOE) data bases
Monitoring transient EDF employees
20 nuclear sites
DOSINAT
NON-NUCLEAR EDF SITES
QUALITY control
National telecommunications network
French system
EDF DOSINAT central computer
DOSINAT
GENERAL ADMINISTRATOR
NATIONAL STATISTICS
•collective dosimetry for all the installation
•analysis per population concerned

For the individuals on the site concerned, we can identify those which have been exposed for the longest period, either during a specific operation or over a longer period. We can therefore develop areas for improving personal dosimetry received during specific operations. We are also able to identify those workers with the highest cumulative doses to direct them towards less exposed activities.

For each site, we are able to monitor the overall dose received more closely, which enables us identify any discrepancy immediately, to analyse the reasons for it and to optimise the end of the operation. Later, we are able to analyse in detail the discrepancies between the forecasts and the actual results. Positive differences may represent good practices and negative differences may direct us to areas in which progress can be made and actions taken. The reference values may also be updated according to events of site.

The site dosimetry calculated by the local systems is used for international (ISOE data base) and national statistics in order to compare units and always to identify good practices and areas in which progress is being made.

EDF's NATIONAL DOSINAT SYSTEM

Each elementary dose in the local system is recopied to the national system, which enables us to have an up-to-date, detailed image of the situation of each individual and all EDF nuclear installations. In order to ensure that an individual is allocated his doses and nothing but his doses, highly sophisticated programmes have been developed to prevent two files being created for a same person or a dose being allocated to the wrong person. This includes phonetic recognition of names, for example.

In so far as individuals are concerned, the national system has the same features as the local system to which it transmits data for several months. In addition, a unit can consult the DOSINAT file of a person coming to work on that unit or inform other units of the provisions it had to take, after consultation, for a worker, where the other units did not carry out the same checks when the person concerned arrived.

All doses received in EDF installations will be transmitted by DOSINAT to the French system, which is being set up, to supplement the doses received in other environments.

Finally, statistics can be established for all those populations which can be derived from data concerning workers, employers or the type of work carried out. We can also obtain an overview of the number of individuals for each annual dose level and decide on improvements to be made from one year to another.

For EDF nuclear installations, we can monitor the evolution of overall dosimetry in relation to the targets which have been established, thus enabling the Nuclear Power Plant Management to take the actions required to correct the situation.

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DOSIMETRIC OPTIMISATION OF WORKSITE INVOLVING THE INSTALLATION OF VATS CONTAINING HIGHLY ACTIVE EFFLUENT

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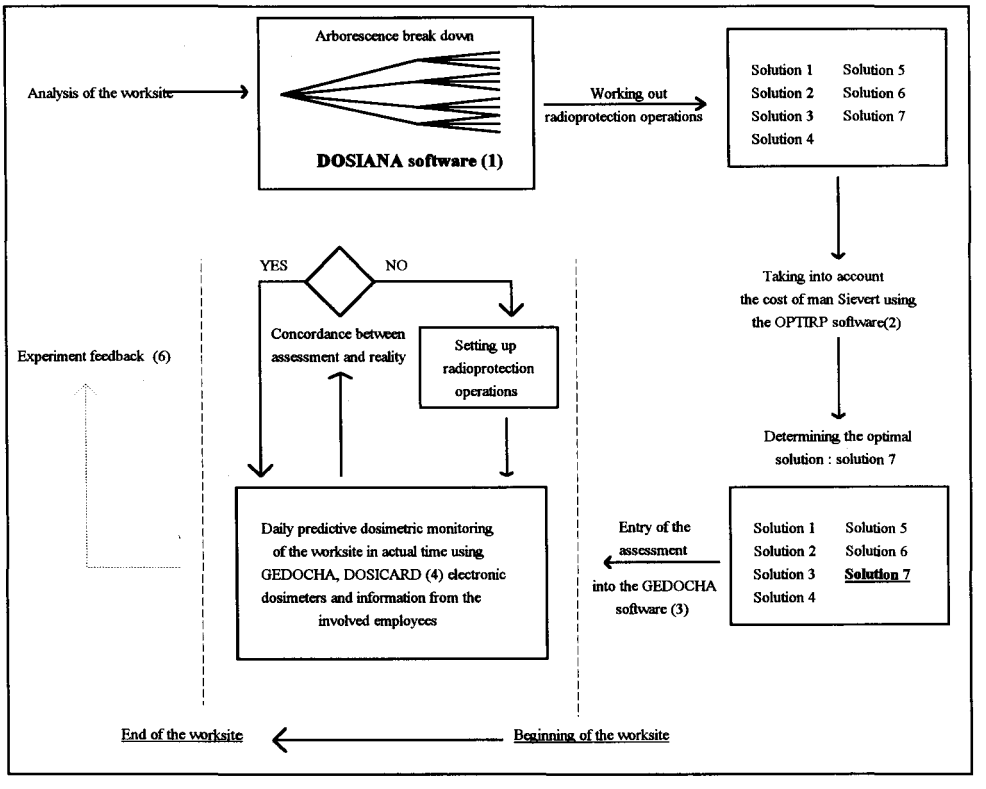
[1] Commissariat à l'Énergie Atomique , Fontenay-aux-Roses, France

Introduction

Within the framework of safety improvements at the CEA, CEA-FAR, concerned to formalize the ALARA initiative, has carrried out for information and training purpose and to create awareness a dosimetric asssessment of the worksite where new storage vats for highly active effluents are to be installed.

This study has led to the following organizational chart:



The numbers in brackets relate to the following informations

(1) DOSIANA software

This software developed by the Centre d'Etude sur l'Evaluation de la Protection dans le domaine Nucléaire (CEPN) (Center for the study of Safety assessment in the Nuclear Field), allows to create with the help for data collected on site, an arborescence specific to the worksite by capturing the values which constitute each elementary task, that is :

- lenght of time
- received dose
- number of employees involved in this task
- working outfit
- category of worker

The second stage was to study the radioprotection operations likely to reduce the doses received by the various employees.

After ruling out a number of financially unfeasable solutions such as robotization, seven propositions was retained.

(2) Cost of man.Sievert

Seven propositions, each modifying the initial arborescence, were made to compete with each other.

The optimization was carried out by considering the cost of man.Sievert according to the following criteria :

The optimization was carried out by considering the cost of man.Sievert according to the following criteria:

- 100 $ per man.Sievert for the employees having received doses between 1 and 5 mSv
- 1400 $ per man.Sievert for the employees having received doses between 5 and 30 mSv

The OPTIRP software of the CEPN brought out the unquestionable superiority of the seventh solution. This arrangement involves the highest investment expenditure but reduces dramatically the operation cost due to faster and more effective work resulting in a 16,1 % saving in collective dose as well as financial saving.

(3) GEDOCHA software

The Service for the Protection against Radiation has developed the GEDOCHA software under Windows environment (GEstion DOsimetrique des Chantiers de Haute Activité : Dosimetric Management of Highly Active Worksites) allowing to establish from the capture of the received doses, of the hours'work and the identity of the employee a series of statistics and indicators in connection with the assessment.

The software indicates (in phasis, sub-phasis and elementary task) the received dose at time t with respect to the estimated overall dose, as well as the amount of hours's work with respect to the estimated overall length of time. A simple indicator is given :

$$\left(\frac{Number\ of\ hours'work}{Estimated\ amount\ of\ hours} - \frac{Received\ dose}{Estimated\ dose}\right) \times 100$$

This indicator is predictive. Indeed, it doesn't reflect an established fact, but an inadequation between assessment and reality in the event the worksite went on operating under the same conditions.

Positive, this indicator shows that the assesssment over estimated the dose (or underestimated the amount of hours); negative, it shows the the assessment underestimated the dose (or overestimated the amount of hours).

Therefore, these indicators allow to make radioprotection measurements while work is in progress in order to fulfil dosimetric objectives.

(4) DOSICARD electronic dosimeter

Each emloyee was given an electronic dosimeter (DOSICARD from NOMATEK) able to memorize the daily doses over a three month period and equipped with an alarm indicating trespassing of the limit. These wear-resistant devices, the size of a credit card, measure reliable doses at the microsievert level.

(5) Information of employees

Any radioprotection measurement involves providing information to all employees, with different skills and coming from various companies.

In order to do this a function for issuing daily "Computerized monitoring cards" was included in the GEDOCHA software. These cards tell the employee working in the frame of an elementary task the maximal dose to be received in order to fulfil the dosimetric objectives. Associated to the dosimetric measurement badge with digital reading and to trespass limit in the badge, reset each day for this objective, the computerized monitoring card increases each employee's awareness of the importance of radioprotection on such worksites.

(6) Experiment feedback

The experimental feedback highlighted various important points :

- the deep involvment of the Radiation Protection Department of CEA/FAR combined to computerized monitoring and a reliable measurement system allows detailled knowledge in actual time of the work in progress as well as an extremly accurate historical analysis.

- the most difficult parameter to assess remains the estimated amount of hour's work. Therefore improvement of the program consisted in adding a menu allowing easily such modification.

Conclusion

The approach used for this worksite is global. Technics used were all complementary, ensuring constant elaboration, expemriment follow-up and feedback of a worksite at a relatively low dosimetric cost (an estimated 36 men.mSv brought down to 30 men.mSv through implementation of the ALARA principle)

This type of global conception of radioprotection involving all the employees (head of project, project managers, companies, radioprotection employees...) which today proves its worth on a modest worksite must now be extended to worksites of a broader scope (several hundreds of men.mSv) where fulfilment of the dosimetric objectives is a major stake.

ALARA PROJECT

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THE INITIAL CONTEXT

EDF's nuclear power stations were built with provisions being made, as from the design stage, to limit radiation sources and the results observed over the first ten years (annual collective dose and dose per unit of less than 2 m.Sv) were considered to be very good. However, these results began to deteriorate from 1988 onwards. At the same time, considerable progress was being made by other generators of electricity, who were achieving results which were better than those achieved by our later units. Furthermore, radiological protection standards are being revised and personal dose limits will soon be lowered.

A DETERMINED POLICY

A policy aimed at reducing doses has therefore been implemented for the benefit of all EDF personnel and external contractors with the objective of reducing personal dose dispersion - the even-handed aim of ICRP recommendation 60 - and collective doses in order to achieve the following:

- providing all workers with the same level of dosimetric protection, whatever their employer,
- achieving collective dose results similar to those of the best operators throughout the world in the operation of nuclear reactors.

This has led to the establishment of dose targets and decision making criteria (value of the collective dose unit to be avoided). It is not a question of reducing doses at any price, but rather in carrying out a cost/benefit analysis of the possible solutions.

ENCOURAGING RESULTS

The decision to adopt this ALARA approach at national level has been extended to all nuclear installations over the last few years.

Working groups for specific areas and led by various local and national committees, have been able to reverse the trend noted after 1988 and the results for the years 1992 to 1995 prove this.

TODAY

In order to return the Nuclear Operating Division of Electricité de France to the ranks of the world's best operators in terms of collective and personal dosimetry, a national "ALARA" project was set up to co-ordinate a certain number of actions, which are aimed at achieving the following objectives:

1. To define a common language, methods and tools for all nuclear installations in order to achieve a level of standardisation in this field.
2. To make the main levels of management aware of their responsibilities for what is at stake in terms of dosimetry and to encourage them to establish objectives in order to move from a resource driven approach to a result driven approach.

3. To bring external contractors into this overall process.

4. To improve the most costly maintenance sites in terms of doses.

5. To reduce specific sources of exposure.

6. To reinforce the organisation and exploitation of experience feedback in this field and to modify the existing information processing system accordingly.

1) Defining a common language, methods and tools

The implementation of a policy of optimisation, in accordance with the ALARA concept, requires a priori management of exposure levels. In other words, we have to be capable of anticipating and forecasting where, when and how given tasks will be subject to exposure.

Therefore, the first objective of the ALARA Project is to define the common language, methods and tools which are required to operate the ALARA optimisation loop at nuclear installation level and involving all Operational Units.

The main areas of activity concern the following:

- defining the structure for the gathering and detailed analysis of dose data in order to be able to combine them with specific operations and levels of responsibility,

- standardising the establishment of gamma environment mapping in order to transpose the reference doses and to render the results objective, thus facilitating comparison and analysis,

- to establish a dosimetric reference for the main maintenance or retrofit sites, according to type of unit shutdown and according to type.

- to adopt common or compatible methods and tools.

2) Responsibilities

The establishment of objectives is an effective management tool. This is one of the key aspects of the ALARA concept. The initial actions set out above, will greatly facilitate the following, be it at national or local level:

- the commitment of various levels of management (client and prime contractor) to their responsibility for making dosimetry forecasts. Complying with, such commitments necessarily result from the desired actions: consideration, analysis, preparation.

- the examination of results, the analysis of discrepancies and finally the implementation of the necessary corrective actions,

- the gradual introduction of performance evaluation techniques in this field.

3) Bringing external contractors into the overall approach

Approximately 80% of doses received in nuclear installations directly concern the personnel of external contractors and a strategy must be developed to bring such contractors into the EDF collective and personal dose reduction and control project.

For this, various improvements to partnership relationships are being considered and are subject to actions at both national and local level. They are intended to define: the requirements set out in agreements, the possible use of contracts in this field, the dosimetry targets specified in orders, training activities, the involvement of contractors in the experience feedback process and evaluation criteria.

In addition, European regulations are being revised to include ICRP recommendation 60 which, in addition to the annual dose limit of 50 mSv, sets of 5-year dose limit of 100 mSv (or an average of 20 mSv per year).

Without waiting for this, and without endangering the jobs of those concerned, the contractors most concerned will consulted from between now and 2000, in order to achieve the objective of nobody receiving more than 20 mSv in any 12 months.

4) Optimisation of the most costly sites in terms of doses

At the same time as the structural actions taken and as a sine qua non, some twelve maintenance sites or fields of activity will receive special attention. They alone represent 70% of the dose received in all nuclear installations and the optimisation of these sites should achieve a reduction of such dose of 15 to 20% in the medium term.

In this case, it is a question of implementation or continuation of such optimisation, to establish an initial dosimetry reference (reference doses, mapping, operating conditions, good practices) and to pass on the benefits to all nuclear installations.

Included among these operations, we should mention maintenance works on steam generators, operations concerned with the opening and sealing of reactor vessels, maintenance on valves and fittings, non-destructive testing, decontamination and cleaning operations as well as operations concerned with auxiliary systems.

5) Reducing specific sources

Certain components in contact with the primary coolant tend to encourage the appearance of oxides or particles which, when they are active, are deposited evenly on the walls of pipeworks and vessels where "hot spots" may be created.

These components include stellites, materials which are used in valves and fittings for example, and they are responsible for a significant share of the doses. A specific aspect of the Project is concerned with the replacement of the worst components (3 mSv/year for all installations) and to forbid their use in future installations.

In addition, some ten nuclear units are affected, to greater or lesser extent by "hot spots". Their impact on dosimetry justifies the taking of curative actions to clean these units, not forgetting monitoring and preventive operations in unaffected units.

6) Improving the organisation and exploitation of experience feedback data...

The feedback of experience gained on work sites is essential. There can be no significant progress made without ordered communication and the exchange of information concerning good and bad practices.

Accordingly, the existing information and experience feedback system will be improved through the Project by establishing compendiums and data bases and by setting up networks of opposite numbers and colleagues.

In addition, the activities of the Equipment Division during the design stages of future nuclear power stations (and the modification of power stations in service) should lead to an improvement in the choice of materials and of the environment and therefore working conditions during maintenance operations.

This requires EDF to make adequate operating feedback information available. The setting up of an efficient and durable organisation is a long term investment which should be made.

... and developing the information processing system

Finally, the requirements which emerge from the various aspects discussed above must be taken into account in order to develop the "site dosimetry" information processing system into more advanced system.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
(Instructions for preparation on reverse)

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Abstract No.
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Author
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PAPER TITLE PANTHERE V1 SOFTWARE : A tool dedicated to the prevision of the dose rates

AUTHOR(S) NAME(S) Ph. RIDOUX

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2 (see page 7)

ABSTRACT (See instructions overleaf)

Simulation of dose rates with appropriate softwares is an important issue when implementing the ALARA process because one of the steps of the ALARA approach is predictive.

Moreover, in the prevision of a task-related dose, the dose rate parameter is more important than other parameters because it does have an influence on working conditions and even on the feasibility of the task.

The PANTHERE V1 software that will be in demonstration is caracterized by its ability to deal with complex situations (geometry, materials, sources). Many existing softwares offer the same ability with the drawback of a laborious data entry task and management of results. PANTHERE V1 benefits from the help of built-in CAD as an interactive means to manage both entry and results. 3 D visualization is avaibable.

The specifications for the development have been elaborated on the basis of the results of a working group gathering different kinds of users.

The final aim of PANTHERE V1 is to contribue to the standardization of the ALARA Approach within Electricité de France.

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Abstract No.
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PAPER TITLE ORGANISATION OF NUCLEAR FACILITIES FOR AN ALARA MANAGEMENT OF EXPOSURES DURING OPERATION AND MAINTENANCE : A SYSTEM APPROACH

AUTHOR(S) NAME(S) SCHIEBER Caroline

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2... (see page 7)

ABSTRACT (See instructions overleaf)

The implementation of optimisation of radiation protection during operation and maintenance of nuclear facilities implies to adopt a management system of exposures based on an 'a priori' knowledge of interventions, in order to be able to anticipate the level of collective and individual exposures. It is therefore necessary to set up an organisation allowing an efficient collection, analysis and circulation of information. The elaboration of such an organisation can be facilitated by the use of a 'system approach' which allows to structure the action fields by a description of the various components interacting in the management of the facility. When this approach is applied to the management of exposure, four 'systems' can be addressed :

- the 'Aims System' describes the objectives set by the facility in the aim of optimising radiation protection,
- the 'Piloting System' describes the persons in charge of the objectives, their responsibilities, the decision scheme and the program elaborated in order to reach the objectives,
- the 'Information System' describes the relevant data for optimisation and their treatment (methods and tools used for the collection, the storage, the control, the analysis and the diffusion of data),
- the 'Human Factors System' analyses the relationship between individuals, their coordination, their motivation as well as their education.

Based on French and international experience, the paper proposes, for each system, the organisation, the methods and the tools which seems relevant for the optimisation of radiation protection implementation, and points out the interactions between the various systems.

ELECTRICITE DE FRANCE

St-LAURENT des Eaux Nuclear Power Plant

Jean-Marc WILLEMENOT **(33) 54 44 84 13**

December 22, 1995

IRPA 9 **1996 International Congress on Radiation Protection** **Vienna**

OPTIMIZING RADIOLOGICAL PROTECTION IN STEAM GENERATOR REPLACEMENTS

Abstract n° 90858

A large number of nuclear power operators worldwide have met difficulties with their steam generators through corrosion of the internal pipes. Electricité de France has not escaped this problem. In four nuclear power stations, we have replaced twelve steam generators by cutting the primary pipes, involving significant exposure for the nuclear workers.

In order to reduce the individual and collective doses, EDF developed a general action policy, applying the ALARA concept during the entire phases of the first operation in 1991, on the DAMPIERRE NPP.

The other Steam Generator Replacements (SGRs) managed in FRANCE and abroad, showed the continual decline of the total exposure received during these operations.

PLANT	YEAR	FORECAST	RESULT	RESULT BY SYSTEM (3 SG / nuclear unit)
DAMPIERRE	1991	4.5 man.Sievert	2.13 m.Sv	0.71 m.Sv
BUGEY	1992	2.6	1.55	0.52
GRAVELINES	1994	1.8	1.45	0.48
ST LAURENT	1995	1.43	0.91	0.30

For the St LAURENT SGR, we have made a breakthrough, reaching one of the best results in the world. In particular, we have reduced by one third of our forecast dose.

The main reasons for the importance of the reduction in our forecast dose are the followings:

* the very detailed preparation of each elementary task, one year before, by working groups consisting of multidisciplinary members. This fine preparation allowed us to reduce the work duration in the exposed areas by 10 % in comparison with the previous SGRs.

* the optimized organisation of radiological protection with one person on shift, on each loop, with the unique responsability for radiological protection and industrial safety; and that during the entire SGR operation in the Reactor Building.

* a very intensive training and information was given to the subcontractors in order to motivate every one of them to reduce the collective and individual doses.

* the quantity (about 100 tons) of lead shielding brought into operation contributed to reduce the collective dose.

* a fine forecast of received dose in each task gives good conditions of real time follow up, and allows to compare the results with the dose forecast, in order to immediately carry out the corrective actions if necessary.

* the firm will of the source reduction was carried out through a long purification in spite of the impact of the outage duration.

* after the prototype development period, by our main contractor FRAMATOME, we took benefit of the normal industrial process applied on the french standardized nuclear network, by using the same tools and equipment on the industrial scale; the St LAURENT Steam Generator Replacement was the fourth in France.

The long-term production of nuclear-powered electricity requires the conditions of a good relationship of confidence with the external environment. The demonstration we can manage a large operation while continuously reducing the worker expositions plays a part in the fulfillment of these conditions.

IRPA9
1996 INTERNATIONAL CONGRESS ON RADIATION PROTECTION
APRIL 14-19, 1996
VIENNA, AUSTRIA

ANALYSIS OF ACTION TO TURN THE TREND OF INCREASING COLLECTIVE DOSES IN SWEDISH LWR'S

Lars A Malmqvist, Thommy Godås
Swedish Radiation Protection Institute

Poster No.: ***P52-21***

ABSTRACT

Since the first nuclear power reactor in Sweden, Oskarshamn 1 was started in 197 l and further on with a total reactor program of 12 reactors (4 NPP sites) the occupational collective dose have been less than the ambition level of 2 manSv per GW installed capacity up to the year 1991. This rather favourable picture was changed for the BWR's in 1992. The trend and some of the forecast discussion shows an dose increase for the BWR's, while the PWR's after 1989 have shown a decreasing dose trend. Increasing safety requirements resulting in extending inspection programs, ageing reactors are factors also contributing to this. The changed situation has called for the establishment of more extended ALARA programs for the LWR's. In April 1993 SSI started a development program in dose reduction. The purpose was to identify the various reasons for the increasing dose levels, study the expected dose- and dose rate- trends for coming years and to advice on concrete actions to reduce occupational doses in the long time perspective. In an other action the SSI has alert the plant management on the situation by direct ALARA inspections where the different components in the radiation protection programme at the various plants has been focused at. In the paper, results from these analysis are presented and expected actions to be taken are discussed.

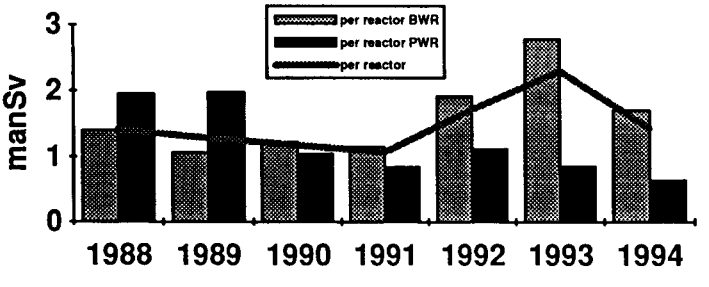
INTRODUCTION

Occupational dose reduction is important not only for the health and safety of the workforce but also because the associated requirement for a good management system enhances safety, quality and reliability of the installation and thus the economy of the plant. Indeed, during the eighties and beginning of the nineties progress has been made and occupational doses have decreased in most countries but unfortunately this is not the situation in Sweden. We have learned that there is no time for complacency because as plants become older there is a general tendency of increased maintenance and repair requirements. Moreover, the ICRP Publication 60, which recommends more stringent dose limits, further draws the attention to the exposure of workers and consequently to ways of reducing such exposure.

TRENDS IN OCCUPATIONAL DOSES

Traditionally occupational radiation doses in the Swedish nuclear industry have been low seen in the international perspective. The decrease in doses we saw in the end of the eightees and in the beginning of the ninetees (figure 1), didn't give any motivation for implementing new ideas with regard to lower the doses.

Figure 1



Unfortunately the situation changed radically 1992. From the positive trend up to 1991, we found the collective dose for 1992 to be "all time high", 20,5 manSv and it become even worse in 1993. In 1993 the total collective dose reached the level of about 28 manSv, i.e. 2.8 manSv per installed GWe, which in fact exceeds the planning level of 2 manSv per GWe which SSI has required as an average for 5 consecutive years.

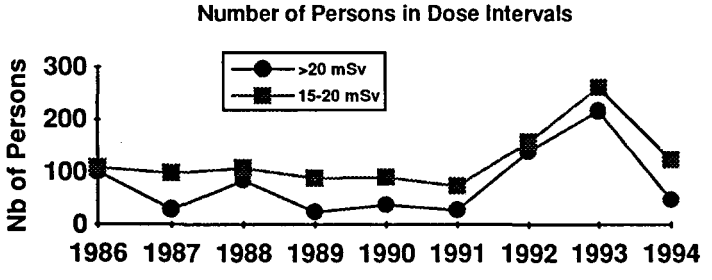
In July 1992, a safety valve in the automatic depressurization system at Barsebäck 2 opened inadvertently at 30 bar and blew steam to upper drywell causing a simultaneous clogging of both trains of the emergency core cooling system. Five of the Swedish reactors, those with external recirculation pumps and small strainer areas, were later that year taken out of operation due to this incident.

At four of the five reactors a decision was taken to replace most of the fibre insulation with metallic insulation. At the fifth reactor fibre-glass insulation was chosen. All of them increased their strainer areas.

From a radiation protection point of view, we have had some unsatisfactory experience from the use of metallic insulation, it requires some time-consuming and troublesome handling.
We have also seen a considerable increase in the individual doses to insulator personnel. Therefore, we had some doubts in the whole operation.
However, the replacement took place and it was not a success neither from a radiation protection nor a technical point of view.
Lack of planning, wrong drawing support combined with an extremely tight time schedule gave a collective dose of approximately 7 manSv in total for the four reactors installing metallic insulation.

Also, the annual individual doses have increased during the same years but are still well below the dose limits (50 mSv for any single year and 100 mSv as a total for five consecutive years). Moreover, the annual average dose for all the work force are below the ambition level of the SSI of 5 mSv per year. However, for some groups the average individual doses have exceeded that level. The number of persons with annual doses of more than 20 mSv was in 1993 about 200 but the number decreased to less than 50 for 1994 (see Figure 2).

Figure 2



REASONS FOR INCREASING DOSES

Collective doses at nuclear power plants are caused by the works that are needed to be carried out i areas where the dose rates are elevated. The dose rates are in general higher the closer to the reactor and its primary systems you are. The five oldest BWR-reactors are here of particular interest in that their recirculation loops require significant inspection and test activities causing important doses to the personnel. In PWR similar problems exist as regard their steam generators.

Works

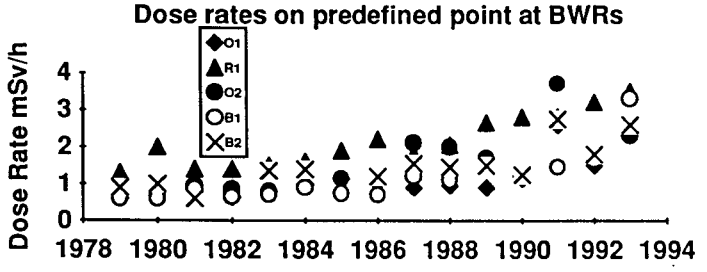
Due to significant amounts of work, the doses have increased considerably during 1992 and 1993. The increasing doses can partly be explained by the fact that some of the reactors are ageing thus requiring significant maintenance and repair works. Increasing safety requirements resulting in extending inspection programs are also contributing to this. In particular, a significant safety related event happened in 1992, when some insulation material was fed into the inlets of the safety injection systems causing risk of clogging . This event led to repair and modification works at all the BWRs of similar design leading to collective doses of about 7 manSv for the five reactors concerned.

Dose rates

Erosion and corrosion of base material in the reactor systems mean that large amounts of corrosion products are fed into the reactor. An important part of them are deposited onto the fuel, activated and thereafter spread in the reactor systems.

Activated corrosion products and in particular Cobalt-60 are the main source of the radiation fields in the nuclear power plants and thus to the resulting radiation doses. Cobalt is one element in stainless steel. The most important contribution comes from stellite which amounts up to 60 percent cobalt in the hard facing alloy, common in valves. Also, fuel failures are causing an increased spread of cobalt-60 from the fuel to the reactor systems. This phenomenon is presently studied in more detail, but it is already now evident that increased attention to fuel failures are needed also from the occupational exposure point of view.
To get a measure of the evolution of the dose rates in the reactor systems, the nuclear industry is measuring regularly the dose rates at a number of places in the reactors. One such series of measurements is shown in figure 3, from which it is evident that the radiation levels at some places in the primary reactor systems are increasing with time. The curve shows the dose rates at all the reactors with external recirculation loops.

Figure 3



At the Radiation Protection Institute (SSI), we cannot accept a prolonged negative dose trend, and therefore, we have worked hard to find countermeasures to turn the trend.

Using our research funds, we started a significant development program in the field of dose reduction. The Swedish "reactor maker" ABB-Atom was on one behalf studying the reasons for the increasing dose levels, estimating the expected dose situation during the years to come as well as giving advice on concrete actions to reduce occupational doses (project DORIS, DOse Reduction In Swedish BWRs).

In the new regulations (SSI FS 1994:2) on occupational exposure, new requirements were included. First of all we decided to introduce a new individual dose limit, 100 mSv in 5 consecutive year in addition to the annual individual dose limit which is 50 mSv.

We have also required an extended education and training program in radiation protection, addressed especially to foreman and team-leaders, working for the utilities as well as for contractors. We believe that this program will increase the understanding and motivation of the personnel to more heavily engagement in dose reduction.

Additionally, we believe in an ALARA, or work management approach, i e where the utilities systematically review their strategy towards radiation protection and develop goals in the area of occupational doses.

In the regulation mentioned above, we also require that each utility have to prepare an ALARA program. These programs shall contain objectives and dose targets for the short and longer terms, discussions on the basic considerations behind the choice of such objectives and targets, dose reduction plans (source and exposure time reductions to be considered) and ways to monitor, follow up and analysing experience. Finally, the plans shall contain programs for education and training of the workforce as well as the organisational aspects related to all the above.

SWEDISH NUCLEAR POWER - DOSES, DISCHARGES, POLICIES

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ABSTRACT

Occupational doses were low but increased 1992-1993. After corrective action, doses are now low again. In spite of relatively large discharges, doses to the public are only a few μSv, except at Ringhals with doses around 0.05 mSv after fuel damage. Emergency preparation includes a revised strategy for early measurements and upgrading of IT equipment.

INTRODUCTION

The Swedish nuclear power programme comprises 12 light water reactors (9 ABB Atom BWRs, 3 Westinghouse PWRs) at 4 sites, started between 1971 and 1985. The oldest reactor, Oskarshamn 1, has been shut down for repairs 1993-95, but is expected to be restarted early in 1996. These NPPs produce about half of the electricity in Sweden with an energy availability factor of ~75-90 %. Some years, utilisation has been about 10 % lower than availability, due to well-filled water reservoirs for hydroelectric power and low power demand in mild winters.

After the TMI (Harrisburg) accident in 1979, it was decided politically that nuclear power would be phased out by 2010. However, few tangible actions have been taken, and a parliamentary inquiry in 1995 suggests that Sweden should begin shutting down NPPs within a few years, but that it would be prohibitively expensive to complete the phaseout already by 2010.

Dose limits conform with ICRP 60 (1). There is also a supplementary lifetime dose limit for workers of 700 mSv. A supplementary limit of 180 mSv at age 30 was removed in 1994 when the ICRP 60 limit, 100 mSv/5 years, was enacted. Public doses around NPPs must not exceed 0.1 mSv annually. Collective doses to NPP workers should not exceed 2 man Sv/GW, and public collective doses should not exceed 5 man Sv/GW. In 1994, this ambition level for occupational doses was made more stringent as a planning limit per site and 5-year period.

OCCUPATIONAL EXPOSURES

Data on the current nuclear power programme in Sweden, and average effective doses due to external exposures, are given in Table 1. Broadly speaking, individual and collective doses

Table 1. Occupational exposure to external radiation at Swedish NPPs, 1985 - 1994

	Reactors in operation [1]	Energy generated [2]	Installed capacity	Workers w/ doses >0.1 mSv	Annual individual mean dose for:		Annual normalised collective mean dose per:		
					utility	contractors	reactor	GW a	GW [3]
(Year)	(Number)	(GW a)	(GW)	(Number)	(mSv)	(mSv)	(man Sv)	(man Sv)	(man Sv)
1985	12	6.7	9.5	4 674	1.8	2.6	0.9	1.7	1.2
1986	12	8.0	9.5	5 688	2.1	3.4	1.4	2.1	1.8
1987	12	7.7	9.7	5 815	1.9	2.9	1.3	2.0	1.5
1988	12	7.9	9.7	5 976	2.2	3.1	1.5	2.2	1.7
1989	12	7.1	9.9	5 449	2.0	3.2	1.3	2.2	1.6
1990	12	7.4	10.0	5 138	2.1	3.1	1.2	1.9	1.4
1991	12	8.6	10.0	4 742	2.1	3.0	1.1	1.5	1.3
1992	12	7.0	10.0	5 999	2.3	3.9	1.7	2.9	*2.1*
1993	11	6.7	9.6	6 574	2.4	4.9	2.5	4.1	*2.9*
1994	11	8.0	9.6	5 886	1.8	3.4	1.6	2.2	1.8

Notes: 1) Oskarshamn 1 temporarily closed for major refurbishment 1993-1995
2) To convert from gigawatt years to terawatt hours, multiply by 8.766
3) Ambition level ≤ 2 man Sv/GW; from 1995, this is a planning limit per site and 5-year period

are low to intermediate in international comparison (2). However, in 1992 and 1993 doses to Swedish NPP staff increased sharply. Safety-related rebuilding of the oldest five BWRs after the so-called strainer clogging incident at Barsebäck in 1992, and major refurbishment of the oldest plant, Oskarshamn 1, were important but by no means sole contributors to this development.

The resulting individual and collective doses still were not, in themselves, very high, and no dose since 1981 exceeds 50 mSv in a year. The normalised collective dose exceeded the regulatory level of ambition somewhat in 1992 and 1993, but no formal rule infractions occurred. Nonetheless, the Swedish Radiation Protection Institute (SSI) found the new trends highly disturbing and reacted strongly. A detailed analysis of the trends and their causes, and a description of the actions taken to offset the trends and their results, are given in the accompanying paper by Malmqvist and Godås (3).

All doses causing concern were due to external exposures. Consistently, very few employees are subject to any intake of radioactive material. Table 2 demonstrates the very limited exposures from intakes.

Table 2. Occupational exposure due to intake of radionuclides at Swedish NPPs, 1988 - 1994

(Year)	Workers with intakes >0.005 ALI (Number)	Range of individual ALI values (min - max ALI)	Maximal committed effective dose (mSv)
1988	5	0.005 - 0.009	0.5
1989	1	0.005	0.3
1990	0	-	-
1991	6	0.005 - 0.095 a)	4.8 a)
1992	0	-	-
1993	5	0.005 - 0.048	2.4
1994	7	0.006 - 0.014	0.7

Note: a) Probably an overestimate due to insufficient surface decontamination before measurement, since a second measurement 6 days later indicated only 0.002 ALI

EXPOSURES OF THE PUBLIC: DISCHARGES FROM NORMAL OPERATION

Calculated annual doses to critical groups and normalised global collective doses from discharges of activity into the environment are shown in Table 3. The collective doses are

Table 3. Calculated annual individual and normalised collective doses to the public from discharges of radionuclides from Swedish NPPs, 1988 - 1994

	Calculated annual effective dose to critical group (mSv)[1]				Global collective doses from ^{14}C releases (man Sv/GW) [2,3]				Global collective doses from all other releases (man Sv/GW) [2]			
(Year)	Ringhals	Oskarshamn	Barsebäck	Forsmark	Ringhals	Oskarshamn	Barsebäck	Forsmark	Ringhals	Oskarshamn	Barsebäck	Forsmark
1988	0.014	0.012	0.002	0.001	2.9	6	6	6	<0.01	0.12	0.03	0.01
1989	0.013	0.003	0.003	0.001	2.9	6	6	6	<0.01	0.02	0.07	0.04
1990	0.013	0.003	0.001	0.001	2.9	6	6	6	<0.01	0.01	0.01	0.01
1991	0.013	0.002	0.002	0.001	2.9	6	6	6	<0.01	0.01	0.02	0.01
1992	0.012	0.001	0.001	0.001	1.9	3.6	3.3	4.6	0.02	0.01	0.01	<0.01
1993	0.028	0.001	<0.001	0.001	1.9	3.3	3.5	4.8	0.09	0.01	<0.01	<0.01
1994	0.043	0.001	0.001	<0.001	2.9	6	6	6	0.14	0.01	<.001	<0.01

Notes: 1) Relevant dose limit = 0.1 mSv
2) Regulatory ambition level for doses due to the sum of all discharges = 5 man Sv/GW
3) Calculated on basis of installed capacity except 1992-93 (with unusually long outages) when based on generated electricity

computed using a truncated 500-year time integral. They are dominated by ^{14}C, the discharges of which are calculated only (not measured).

In general, calculated individual doses remain relatively low, although fuel damage at Oskarshamn 2 in 1988 and a more serious case at Ringhals 1 (BWR) in 1993 are very evident in the Table. Furthermore, the PWRs at Ringhals cause higher calculated doses due to ^{14}C because of lower smoke stacks. The regulatory level of ambition for the collective dose, ≤5 man Sv/GW, is often exceeded somewhat by the calculated ^{14}C discharges.

Whilst computed doses are low to moderate, the underlying activity discharges are in some cases fairly high compared to similar reactors in other countries (4), to some extent reflecting large recipient capacities. This could raise questions about the application of Best Available Technology (BAT) under an ALARA regime.

Admittedly, environmental monitoring indicates relatively minor contamination outside the NPPs. Samples of various plants taken from the marine environment in the immediate vicinity of the plants show elevated radionuclide concentrations which are usually well correlated with reported discharges. For ^{60}Co, these may reach some kBq/kg dry weight for the most exposed organisms (algae scraped from plates 200 m from the discharge pipes). In terrestrial samples, concentrations are very low except for contamination due to the Chernobyl accident (9). Nonetheless, the fact that relatively big discharges cause small doses to humans under the prevailing conditions also pinpoints the question of protection of the environment: does protection of man always automatically provide fully adequate protection for other species, as postulated by ICRP?

EMERGENCY PREPAREDNESS

An extensive programme from 1981, in response to TMI, was further augmented in 1990 as a result of Chernobyl. One of the early components of the measurements foreseen in case of an accident in Sweden comprises indication measurements, performed by specially trained firefighters at predetermined locations around NPPs. Several rounds of exercises have shown that it is unwise to complicate such early actions by trying to deploy available staff at selected locations and times expected to be particularly exposed. Instead, SSI now has the strategy that all locations should be manned and measurements start as soon as possible in an emergency.

Within SSI and in the counties concerned, computer equipment and software are being upgraded. This is not primarily to do with dose calculations, for which fairly simple existing tools are perfectly adequate in the early emergency phase when a paucity of measurement data will in any case limit the precision of prognoses. Instead, the tools which really need to be state-of-the-art deal with communication, record-keeping, and data storage, handling and retrieval. In this context, harmonisation of software tools in different countries and at the various NPPs is important.

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ASSESSMENT OF THE INTERNAL CONTAMINATION OF NUCLEAR POWER PLANT WORKERS IN FINLAND

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INTRODUCTION

Two nuclear power plant (NPP) sites with two reactors each are located in Finland. At Loviisa the reactors are of the PWR (VVER) type and at Olkiluoto of the BWR (ABB Atom) type. Finnish NPP surveillance regulations require that doserate, air and surface contamination be monitored and that partial-body monitors (also thyroid monitors) be available at the plant site for direct measurements of internal contamination. The risk of NPP workers becoming internally contaminated is highest during the annual refuelling and maintenance outage. During the outage period, a group representing workers with the highest risk of internal contamination must be measured with a system giving nuclide specific results. For this purpose, whole-body counter measurements are performed by the Finnish Centre for Radiation and Nuclear Safety (1).

MATERIAL AND METHODS

The body monitors detection limits of some thousands of becquerels at the NPP:s are used especially for internal contamination control of outage workers. For whole-body counter measurements a mobile unit with chair geometry and one HPGe detector or a stationary scanning type counter equipped with NaI(Tl) and HPGe detectors in Helsinki are used (detection limits ~ 50 Bq for ^{137}Cs)(2). With the mobile whole-body counter at STUK a small group of workers varying from 30 - 130 persons, representing different types of job involving risks of internal contamination is routinely measured once during the annual outage and once during normal operation each year and always if internal contamination is suspected. Each worker showers and changes into clean clothes before the measurement. The measured body content of radionuclides is considered to be internal although part of it might be skin contamination. In the mobile system the measurement time is usually 1000 s and in the stationary system about 1800 s.

After incidents involving risks of internal contamination workers are monitored and further investigations are performed if decontamination procedures are not effective enough. In such cases the stationary scanning counter in Helsinki is used for whole-body counting.

RESULTS

Varying but small amounts of activation and corrosion products have been detected (e.g. ^{51}Cr, ^{54}Mn, ^{58}Co, ^{59}Fe, ^{60}Co, ^{95}Zr, ^{95}Nb, ^{110m}Ag, ^{124}Sb and small amounts of ^{131}I), mainly during outage periods. Figure 1 illustrates two contamination cases at the NPP:s in Loviisa and Olkiluoto. In the present paper all detectable body burdens ($\geq$ 100 Bq) are classified as contamination (Fig.2). The mean body burdens have usually been < 1 kBq (3).

The body burdens of persons working together have been found to vary much thus showing the importance of radiation protection instructions and of good individual working habits. Some profile scan measurements have been done in order to gain more information on the distribution of radionuclides in the body and the retention pattern.

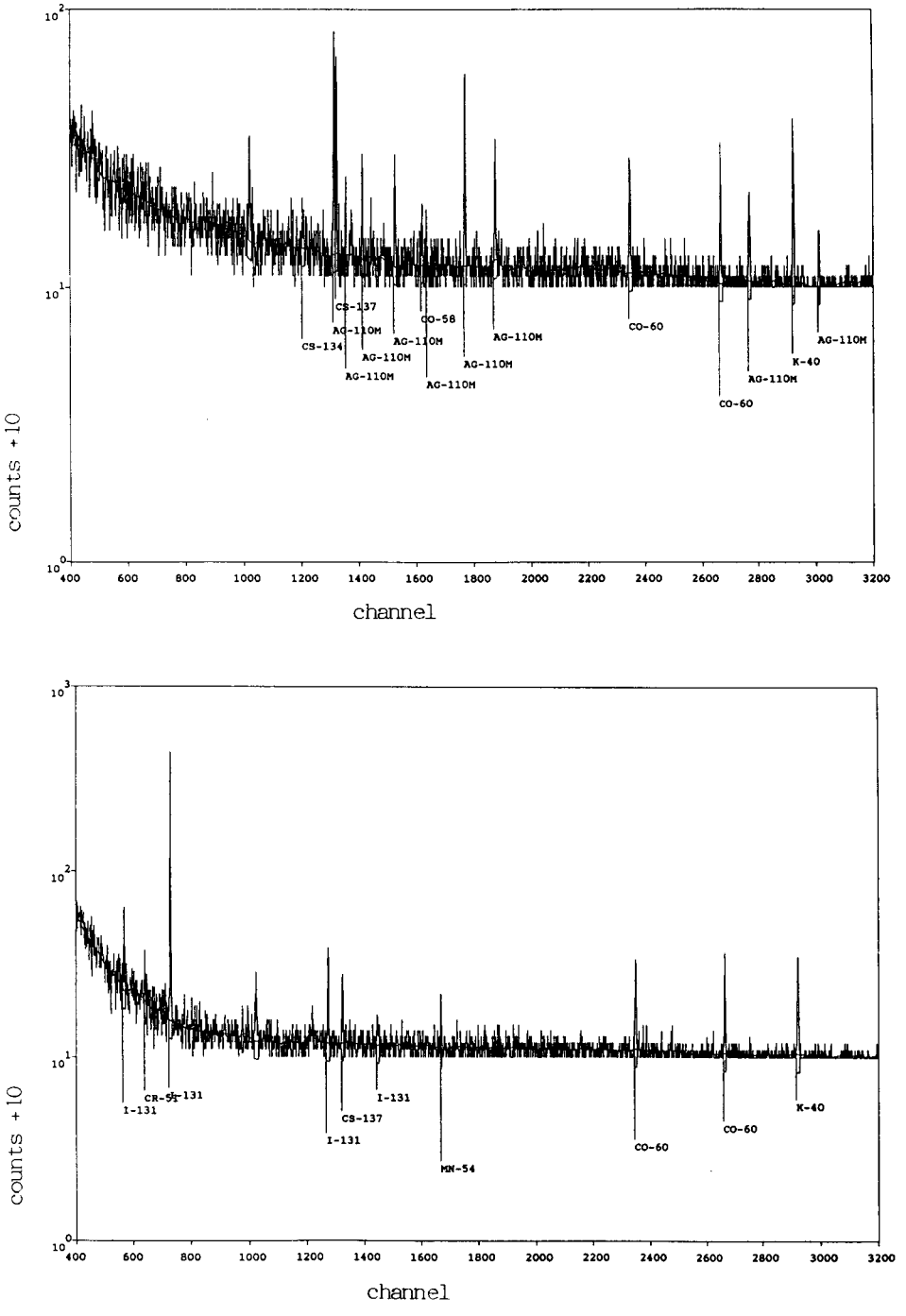

Figure 1. The upper spectrum was measured at the PWR NPP (100 Bq ^{58}Co, 400 Bq ^{60}Co, 1000 Bq ^{110m}Ag) and the lower spectrum at the BWR NPP (1200 Bq ^{51}Cr, 100 Bq ^{54}Mn, 500 Bq ^{60}Co, 2400 Bq ^{131}I) during the annual outages in 1994. The radiocesium originates in the Chernobyl accident.

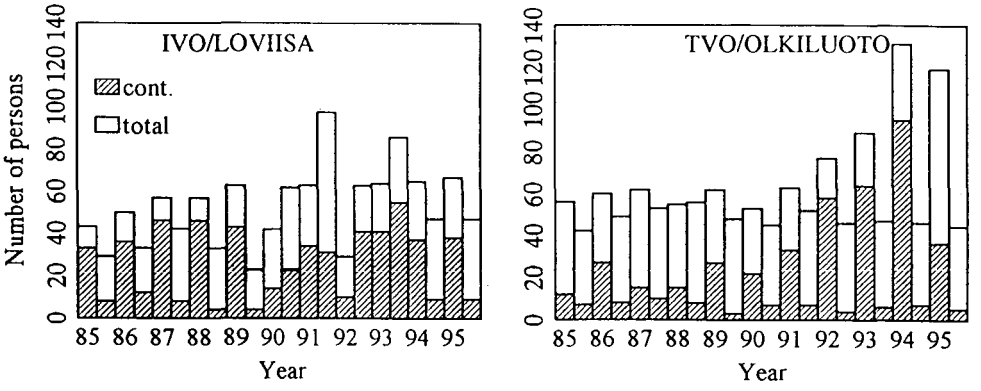

Figure 2. The number of persons measured at both NPP:s from 1985 forwards. The bar to the left represents the outage period each year. All detectable body burdens are classified as contamination (≥ 100 Bq).

DISCUSSION

A typical occurrence for VVER reactors is the appearance of ^{110m}Ag in workers, as also reported by Ormai et al. from Hungary (4). No ^{110m}Ag was detected in workers from the BWR NPP in Finland, although it was detected in the primary coolant. In terms of radiation protection ^{60}Co is the most important radionuclide.

Contamination levels depend greatly on the type of repair and maintenance work done during the outages. Experience has shown that, if the internal effective dose approaches 1 mSv, an incident has occurred. In an outage, usually no more than 20 % of the whole-body counted workers receive a dose of > 0.1 mSv, which is the minimum recording level in Finland. If the body burden of a worker is considerably higher than the mean body burdens during an outage, this person will be remeasured and the radiation history established so that the radiation dose can be estimated more accurately. If possible, such persons are measured repeatedly during a longer period of time to enable even more accurate estimation of the half-lives of the radionuclides in question.

In addition to dose control, whole-body counting aims to monitor radiation hygienic conditions in work areas and especially to motivate workers to pay close attention to their own working habits. The whole-body counter measurements help to keep the doses as low as possible. They are also important for reassuring workers that working conditions are satisfactory.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Lifetime doses of nuclear power plant workers in Finland

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2 (see page 7)

ABSTRACT (See instructions overleaf)

In Finland, there are four nuclear power plants at two sites, commissioned between 1977 and 1981. A central registry for occupational doses of all radiation workers has been maintained from the year 1965. The computerized registry allows easy follow-up of personal doses, including lifetime and 5-year doses.

In this paper the results of a study of lifetime doses of workers are presented. The results include a table of distribution of lifetime doses and data on the age and work classification of the persons with highest doses. Also distribution of 5-year average in the periods of 1985 - 1989 and 1990 - 1994 are presented.

RADIATION PROTECTION IN TVO AFTER YEAR 2000

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INTRODUCTION

What is the optimum level of radiation protection in nuclear power plants? Is it still adequate if radiation protection is realized according to ALARA? In most cases I would say no if we are seriously thinking about the acceptability and the future of nuclear power.

Although TVO´s annual radiation doses are very low worldwide, on average 0,74 Sv per unit, they are still too high and must be decreased. There are two ways to do it, either by minimizing the radiation sources or by making people work in such a way that they get less doses. In TVO we are doing both. There is a DOSE-project, the aim of which is to decrease the cobolt-60 content in the reactor water by all available means. This paper only discusses how we must take the human factor into consideration in minimizing personal doses in the future.

INDIVIDUAL RESPONSIBILITY

Given the chance, a worker will look for the most dose-efficient method of doing his work. I am convinced that personnel safety will certainly not be at risk if we allocate responsibility more to the individual.

Who is more interested in the person´s own radiation doses than the person himself? Nobody! A radiation protection organisation is more interested in seeing that personal dose limits are not exceeded or that collective doses are on acceptable level, but an organisation does not have the capacity, and maybe not even an interest, to concentrate on a single person's low doses. That being the case, the person himself is his best radiation protection man, and the only one to take the whole responsibility for his own safety. Let us make it possible for all nuclear power workers to take care of their own radiation protection. This will take a lot of radiation protection training.

TRAINING

A refuelling outage involves about 1000 workers from some 100 different contractor companies. These workers receive about 70 - 80 % of the total annual dose. If the number of companies were reduced to 20 - 30, arranging company-specific radiation protection training would be more practical. With company-specific training we can concentrate on radiation protection matters that are important to that specific company. Electricians have different problems than mechanicians.

Training alone is not enough to keep the contractor companies' knowledge of radiation protection on a high level. Each company should nominate a person who is responsible for the communication between the power company and the contractor company on radiation protection questions. He would be responsible for disseminating all information which is sent to the company to all its employees. He would thus act as a company's own radiation protection manager.

VERSATILITY OF SKILLS

One way of reducing doses is to reduce the number of people working in a controlled area. By teaching people to do a series of operations rather than just one specialist task, the number of people can be reduced. For example, a mechanician can insulate components, make simple scaffoldings and take care of the cleanliness of the work site in addition to his mechanical work. Doing the work in this way, it will become more flexible and also more interesting and more challenging to the employee. A motivated employee also takes good care of his own radiation protection.

GENERAL CLEANLINESS AND ORDER

Cleanliness and order go hand in hand with good radiation protection. That is why the responsibility for the cleanliness of the working environment should also be passed more towards the individual. Contamination cannot be seen by eyes but by keeping surfaces clean and the work site in order, so that there are no unnecessary objects lying around, you can be quite sure that there is no contamination. A high level of cleanliness and order also makes the work site pleasant and safe. Working in such environment is faster and more fluent.

DEVELOPMENT OF RADIATION INSTRUMENTS

As responsibility for radiation protection is given to those who do the work, new demands will be placed on the measuring instruments. They have to be light and simple to use. One push button (on/off) is enough. All other buttons are unnecessary and make using the instrument too difficult. The accuracy of the instrument does not need to be very good; 1 mSv/h is enough. Analogical display is better than digital. The rough range of dose rate is more important than the exact value in numbers. Besides the dose rate instrument, the employee has to have an electronic dosimeter with a digital display and sound alarm to be able to monitor his doses.

ONLY BY MEASURING CAN YOU FIND OUT WHICH DOSES YOU SHOULD CONCENTRATE ON REDUCING

COMPUTER AIDED RADIATION PROTECTION SYSTEM AT TOKAI REPROCESSING PLANT

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INTRODUCTION

Radiation control for workers and workforce has been carried out strictly and effectively taking into account ALARA principle at Tokai Reprocessing Plant(TRP) which has treated about 860 tons of irradiated fuels by now since 1977. The outline of radiation control method at TRP has already been described in (1), (2) and (3). This paper briefly describes our experiences and the capabilities of Radiological Information Management System (RIMS) for the safety operation of TRP, followed by radiation exposure control and activity discharge control as examples .

OBJECTIVES AND SCOPE

The main plant construction started in 1971 and was finished in 1974, since then waste treatment facilities and waste storage facilities were successively constructed together with locating the safety control rooms at several facilities separately. On the other hand various kinds and levels of radioactive materials are treated in TRP and the radiation fields are monitored for the purpose of avoiding excessive exposure of workers and of confirming that working environment is satisfactory for operations. Therefore all the data are to be centralized in the central safety control room for the effective radiation protection.

SYSTEM DESCRIPTION AND CAPABILITIES

The RIMS, which consists of several functions such as a radiological monitoring at workplaces, a radiological data base of past experience, a management of radiological work, etc., was designed and set up at the safety control rooms to reduce exposures associated with routine and special repetitive maintenance operations in TRP. The schematic diagram of RIMS is shown in Fig.1.

(1) Radiological Monitoring at Workplaces

The radiological monitoring at workplaces of TRP is conducted by using about 400 channels of continuous on-line monitoring equipment such as area monitors for gamma-ray and neutrons, dust monitors for alpha and beta particles, exhaust monitors for krypton-85, iodine-129,131, etc. which are located at each spot. The signals of all the detectors are centralized to the health physics panels in each safety control room and then are linked to the computer system of RIMS through the panels of the central safety control room. And to support these on-line monitors, manual radiation monitoring such as area survey of external radiation, airborne radioactive materials and surface contamination is carried out periodically. Dose equivalent rates at fixed points in each facility are measured with survey meters daily, weekly and monthly. Airborne radioactive materials in workplaces are weekly sampled by the air sampling system and then measured by low background proportional counters to estimate the concentration of radioactive materials in workplaces. Surface contaminations at fixed points in the workplaces are estimated by smear sampling daily and weekly. All these data except surface contamination are input to the computer of RIMS. There is no need to record surface contamination which is normally less than the detection limits except for the special maintenance period because the restricted area of TRP is essentially well controlled based on the principle of "Non-contamination". The off-line survey data are input to the RIMS through handy-terminals or radioactive measurement instruments.

By using the RIMS, each data is displayed in case of need, for example, every minute, hour, day or week.

The data are easily compared to the authorized limits to confirm the state of the facilities. If dose equivalent rate or concentration of airborne radioactive materials were raised over the control levels, they warn workers to evacuate from the room immediately.

(2) The Radiological Management System and Radiological Data Base of the Past Experiences

In case of the work done under high dose rate or in contaminated area such as the work in a cell, a special radiological work plan needs to be authorized by the director of TRP after checking by the staffs of radiation protection section and the safety superintendent. Advice on radiation protection is given to the plan in consideration of minimization and optimization of exposure to workers. On this occasion the staffs of radiation protection section refer to the past similar experiences from view points of the working process, the radiation monitoring programme, the choice of radiation protection apparatus, the education and training, the maximum individual dose, the total collective dose, the emergency planning etc.. Especially the prior check of the radiation levels and the procedures of the work is very important. So far these performance checkings have been done by the human sea tactics, and so it takes a lot of time and the products of the past experiences have not always been put to practical use. The RIMS includes many experiences of the past special radiological works, and the required data are gained quickly as factors of date, working room number, working unit, working group's name, species serial number and so on. During the work the exposure doses to workers are collected and added by every individuals and are compared with the planned exposure dose by using the alarmed personnel dosimeters and the RIMS. After the work the analysis report of the work is written and is input to the RIMS for the next similar works.

(3) Information on the Present Conditions of Plant Operation

The RIMS includes operational conditions such as shearing, dissolution, clarification, extraction, Pu purification, Pu concentration, acid recovery distillation, and so on which are needed now and then.

CONSEQUENCES OF RADIATION CONTROL

The number of workers have increased in keeping with increase of facilities. Higher collective dose equivalents, by way of example, 1.7 man-Sv in 1983, 2.0 man-Sv in 1984 and 4.5 man-Sv in 1988 were resulted from the large maintenance works such as the repair of dissolvers, the replace of acid recovery evaporator and so on. Under normal plant operation the average collective dose equivalent is about 0.5 man-Sv in a year and the average exposure of an individual is about 0.1 mSv recently. These values are mainly the external exposure because the internal exposure have rarely occurred.

CONCLUSION

By operating the RIMS, the conditions of workplace such as dose equivalent rate and air contamination are easily and rapidly grasped to take prompt countermeasures for radiological protection, localization and elimination of contamination, and also the past experience data are properly applied to new radiological works to reduce exposures associated with routine and special repetitive maintenance operations at TRP. Finally, authors would like to emphasize that the form and system for radiological control of reprocessing plant has been established throughout our 15-year-experience at TRP.

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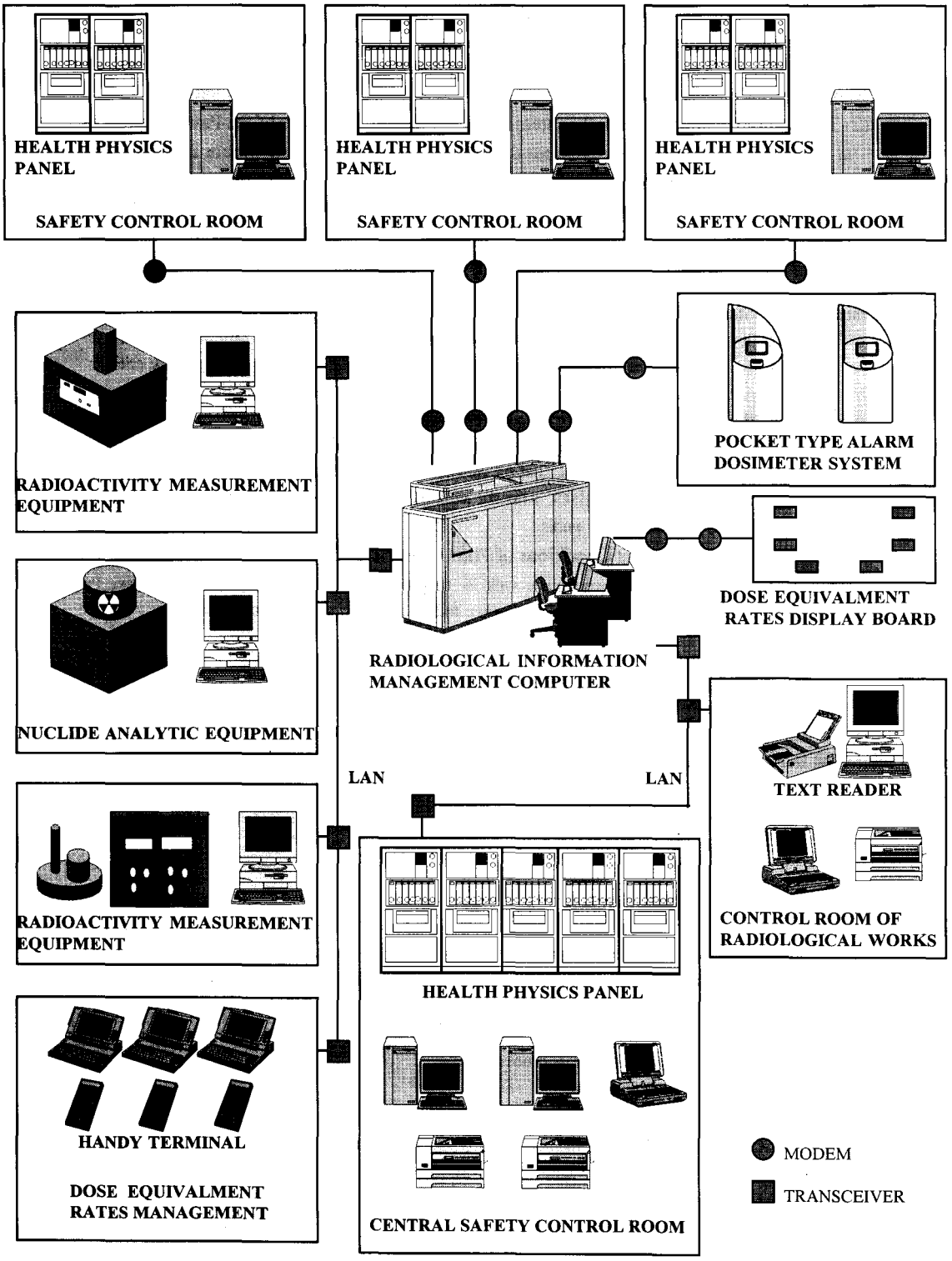

Fig.1 Schematic Diagram of the Radiological Information Management System (RIMS)

TREND OF COLLECTIVE DOSE AND DOSE REDUCTION MEASURES OF MITSUBISHI ELECTRIC CORPORATION WORKERS IN NUCLEAR POWER PLANTS

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("MELCO" hereafter)

ABSTRACT

MELCO has supplied the reactor instrumentation control system, reactor coolant pump motors, turbine generator and central control system for the pressurized water type nuclear power plant. For the legal periodical inspection and repair work, MELCO has also received orders for the periodical inspection for 23 power plants (including 4 plants under construction) of 5 electric power companies, and executed the inspection work from the view point of preventive maintenance.

The annual dose for MELCO's workers is liable to be decreased in spite of increased number of plants. The dose for new plant in particular is 50% or less as compared with that for conventional plant. This is because the measures taken for the conventional plant against the dose reduction is reflected upon the new plant.

The dose reduction measures are taken for each system for which order was received. Such measures are mainly intended to improve the work procedures and equipment for reduction of work time in the radioactive area and to arrange the working process, so as to perform the work in such period when the dose level at the working environment is low.

To enhance the workers' consciousness for reduction of dose, MELCO provided the workers with dose predictive training, and let them aware of such items known at the tool box briefing(TBM), which could realize the dose reduction for workers. MELCO has been positively promoting the activity to arrange the desirable work environment for extermination of 3Ks (*Kiken, Kitsui, Kitanai*) or 3Ds(dangerous, difficult, dirty) including protection against radiation in corporation with electric power companies.

INTRODUCTION

The number of workers engaging in nuclear facilities for MELCO's nuclear work to talled to approx. 300 (net) annually, and the number of workers engaging in such work, including the employees of cooperative companies totalled to approx. 1,500 (net) annually.

The paper is intended to report the results of investigation/analysis on the annual dose(1983~1994) for employees of MELCO at time of periodical inspection from the view point of dose reduction, and to introduce the technical method for dose reduction and contents of education/training for workers.

TRENDS

1. The annual collective dose(PWR) for MELCO's workers(including workers of cooperative company) during period from 1983 to 1994 tends to be decreased inspite of increased number of power plants.
   1) The annual dose for MELCO's worker per each periodical inspection of PWR plant is liable to be decreased.
   2) The reason why the dose for MELCO's worker was increased in 1993 is that the

attached work (replacement of steam generator) was executed.
(dose for work concerned: 443 man.mSv)

2. The main exposure for MELCO's workers is that received during periodical inspection, or from the replacement work of reactor coolant pump motor. The transition of dose during latest 5 years is liable to be decreased.

METHODS

The basic principles for dose reduction measure are to secure the proper distance from the radiation source, to reduce the time required for work, to shield and remove the radiation source. The effect of dose reduction is depending on how these four principles are fulfilled by the electric power company, contractor(who plans the work) and workers in cooperation. As a result of such activities, the dose for MELCO's workers during periodical inspection of nuclear power plant is liable to be decreased.

1. The methods for those who plan the dose reduction measures will include the improvement of equipment/facility, adjustment of work implementation process, and improvement, education and training of work procedures, etc.
   1) The improvement of equipment/facility will include the modification of power cable connection from conventional type to plug-in type, installation of lube oil filling and drainage equipment for lube oil, installation of lifting device for motor terminal box, and is intended to shorten the time required for work.
   2) The adjustment of work implementation process, for instance, is so planned as to be performed when the water is filled in the secondary side of steam generator within the loop room as much as possible.
   (When the water has already been filled in the secondary side of S/G, the water will serve as shielding substance, causing the environmental dose rate to be decreased by more than 50%.)
   3) The education/training is an essential item from the view point of radiation protection, as it helps to enhance the worker's consciousness against dose reduction.
   For the education/training, MELCO provides the basic education/practical education on the radiation for 7 hours, to learn the knowledge on radiation, influence of exposure, dose limit, etc.
   For dose reduction, MELCO provides the dose prediction training(HYT), using the illustrations, to minimize unnecessary radiation exposure.
   Such training was evaluated for effect in accordance with the result of questionnaire conducted on the workers, which proved that some effect was brought about, though it was difficult to evaluate it quantitatively.
   For dose prediction training, MELCO gathered the workers in charge and selected the leader to make up a small-member group, to provide the dose prediction training, using the illustrations simulating the site for each group
   In addition, MELCO provides a reading-out of instruction manuals prior to the work, or the TBM on the very day of work.

2. The measures to be taken by the electric power companies will include the purification of primary cooling material stained due to oxidation operation of

plant, removal of high dose rate piping, etc.
When the radiation source is removed, the dose rate for entire plant is decreased, causing the dose against the worker of contractor to be decreased.
Such measures are expected to provide the greatest dose reduction effect.

3. The education/training discussed above is intended to allow the workers to learn the knowledge on the basic principles for dose reduction, to be conscious of dose reduction,
and further to utilize such knowledge during the process of work.
Enhancing the consciousness of worker against the dose reduction will help to eliminate unnecessary radiation exposure of each worker, and repeated practice will reduce exposure.
The attitude of each worker concerned with the radiation protection will be an essential item to secure the radiation safety.
It is also essential for each worker to take the dose reduction activity as his own activity, and to let him have such consciousness as to be taking part in the activity.
Although it is difficult to calculate the dose reduction quantitatively for trial, the dose reduction of 30% is thought to be an effect of such activity.

CONCLUSION

The paper mainly discussed the dose reduction measures from the standing point of a contractor.
It is thought that the great effect will be assured for dose reduction by through the work by three parties of electric power company, contractor(who plans the work) and workers in cooperation.

It is also necessary to minimize the exposure risk of workers by reflecting the ALARA spirit(which ICRP recommends), upon the local equipment/facility design/layout and workers, to minimize exposure risk of workers.
It is important to enhance the consciousness of workers against the dose reduction as well as to study the technical dose reduction measures.
Each nuclear power plant takes the radiation as a main factor of countermeasure against 3Ks, and has been carrying on the dose reduction measures together with the electric power company.

The dose reduction will be an important problem to be settled, for those who are engaged in the radiation protection. The authors are intended to positively promoting the subject in the future.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Aplication of ICRP 60 and its Impact on Dose Regulation in NPP of Slovak Republic

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2. (see page 7)

ABSTRACT (See instructions overleaf)

This year the new Slovak legislation on radiation protection, based on ICRP 60 recommendations, has been prepared. The individual dose trends for different professional groups of nuclear power plant workers for last few years are sumarised. In this conection the attention has been given to the professional groups and activities with potential high radiation risks. The results show that some problems with new dose concepts complying can arise.

FIELD MEASUREMENTS OF BETA-RAY SPECTRA INSIDE NUCLEAR GENERATING STATIONS USING A SILICON DETECTOR COINCIDENCE TELESCOPE

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INTRODUCTION

Beta radiation is now recognized as a significant radiation safety problem in the personal dosimetry of weakly penetrating radiations. The use of "thick" dosimeters (eg., typically 100 mg cm^{-2}) for beta dosimetry requires, therefore, the application of beta correction factors which correct for the under-response of these dosemeters to low-energy betas. The estimation of these beta correction factors requires in-turn some knowledge of the beta-ray energy spectrum impinging on the dosemeter. In acknowledgement of this requirement, and to improve the accuracy of beta ray personal dosimetry in their nuclear power installations, Ontario Hydro initiated a study in the 1990s aimed at the characterization of beta radiation fields in CANDU reactor typical working conditions.

In the first stage of this effort, a two-detector coincidence telescope, based on a thin (97 μm) silicon detector (dE detector) and a thick (2 cm) plastic scintillator (E detector), was used (1). The use of a 97 μm thick silicon detector allowed us to achieve a lower energy coincidence threshold of 125 keV. This telescope was used to measure approximately thirty beta radiation fields in CANDU nuclear generating stations (2). Monte Carlo calculations were then used to estimate beta correction factors for the LiF-TLD elements in the Ontario Hydro TL dosimeters (3). Due to the poor energy resolution and noise characteristics of the plastic scintillator, the spectrometer coincidence efficiency decreased rapidly below 500 keV reaching a value of only approximately 15% at 125 keV. These low energy characteristics were judged insufficient, since the Monte Carlo calculations (3) showed that it is exactly the energy region below 500 keV for which the electron depth-dose distributions vary extremely rapidly, thereby contributing to the large and uncertain values of the beta correction factor. Furthermore, the measured beta spectra, corrected crudely for the loss in coincidence efficiency below 500 keV, showed an approximately exponential electron fluence intensity that increased with decreasing electron energy, implying that the deduced beta correction factors could be very significantly underestimated due to the uncharted electron energy region between 60 keV (the minimum electron energy which penetrates to a skin depth of 7 mg cm^{-2}) and 125 keV. Other serious difficulties arose when measurements were attempted in areas, for example a reactor containment vault, which required encapsulation of the spectrometer in plastic to prevent radioactive contamination. The encapsulation in plastic without adequate ventilation or cooling led to an increase in the temperature and leakage current of the front silicon detector, resulting in a rapid increase in noise level and consequently to an uncontrolled increase in the chance coincidence rate.

Based on these considerations a two- or three-element detector coincidence telescope based exclusively on silicon detectors, and cooled thermoelectrically when encapsulated was constructed and tested with the following significant advantages.

SYSTEM PERFORMANCE

1. The lower energy coincidence threshold (front dE detector, A = 40 μm) is 70 keV (60% efficiency) compared to 125 keV (15% efficiency) for the silicon/plastic

spectrometer. Slightly higher but still improved coincidence thresholds are obtainable with 72-μm and 97-μm thick detectors.

2. The coincidence efficiency decreases from 100% for electron energies below 100 keV to 250 keV depending on the choice of thickness for the front detector A compared to 500 keV for the plastic spectrometer. For the 40-μm detector the coincidence efficiency is 100% over the entire range of energies above 100 keV. For the thicker dE detectors the loss in coincidence efficiency is far more moderate below 250 keV than the loss in coincidence efficiency for the plastic spectrometer.

3. An important figure of merit (FOM) of the spectrometer is the rejection factor against photons, R_p, where

$$Rp = N_1/(N_t - N_c). \qquad (1)$$

N_1 is the singles counting rate in the thick energy detector , N_t is the total coincidence counting rate, and N_c is the chance coincidence rate. R_p is defined in this manner for a pure incident photon fluence and is essentially a measure of the reduction in the photon count rate due to operation of the coincidence requirement.

The photon rejection ratios are also significantly improved compared to the silicon/plastic spectrometer. For ^{133}Ba, in the two-element mode, the photon rejection ratio is 1600:1 compared to 360:1 for the silicon plastic spectrometer. In the three-element mode, the photon rejection ratio for ^{60}Co photons is approximately 2000:1 compared to 225:1 for the plastic spectrometer.

The beta-ray spectrometer described herein, incorporating photon rejection ratios of up to 2000:1, represents a significant improvement in the "state-of-the-art" in beta ray spectroscopy in the intense photon fields often encountered in reactor radiation environments. The spectrometer has been used to measure over 40 field spectra in CANDU reactors (5) ands the details of these measurements and the calculated beta factors using depth dose distributions based on Monte Carlo calculations for these spectra are described in the following.

BETA-RAY MEASUREMENTS

A total of forty two measurements were carried at three sites operated by Ontario Hydro, the Pickering, Bruce B and Darlington Nuclear Generation Stations. The measurements were carried out on various fuelling machine and boiler room components and smears from various working areas, in various detector configurations and at various source-detector distances. These sources are representative of components that working personnel come into direct contact during clean-up and maintainance procedures. The spectra are generally featureless with the relative intensity of the electron fluence increasing roughly exponentially with decreasing electron energy. The maximum energy varies between 1200 keV and 3500 keV, with an average energy of approximately 550 keV. This behaviour is apparently typical of a mixed fission-product spectrum. The threshold energy of the various spectra varies between 70 keV and 150 keV. Filtered spectra, through various layers of protective clothing, show no significant changes in shape in the important low-energy region, the main effect of the filtration being a reduction in the maximum energy of the spectra. Thus, although low-energy betas in the primary spectrum are absorbed by the protective clothing, multiple scattering and attenuation of the higher-energy electrons replenishes the supply of low-energy electrons, resulting in an essentially unaltered spectrum in the low-energy region. This observation goes against the common dogma that protective clothing reduces superficial dose via absorption/removal of low-energy electrons. This will be true only in the absence of high-energy electrons which continually replenish the supply of low-energy electrons.

CALCULATION OF BETA CORRECTION FACTOR

Beta correction factors were calculated for all the measured beta-energy spevctra, for both the 100 mg cm-2 and 240 mg cm-2 skin and extremity dosimeters, in both perpendicular and isotropic geometries. It is interesting to compare the average beta correction factors obtained using the silicon spectrometer with those obtained using the plastic spectrometer. The beta factors are increased by 19% for the skin chip and by 11% for the extremity dosimeter respectively, indicating that low-energy electrons (between 60-125 keV and not detected by the plastic spectrometer) lead to a measureable, but not dramatic, increase in the beta correction factors.

Averaging over all the beta correction factors for the skin dosemeter yields 2.33 ± 0.83 compared to the current value of 1.74 used by Ontario Hydro. Averaging over all the beta correction factors for the extremity dosemeter yields 3.73 ± 1.38, compared to the current value of 2.0 used by Ontario Hydro. Our measured values are, therefore, 34% (skin chip) and 87% (extremity dosemeter) higher than the current beta correction factors used by Ontario Hydro in their dosimetry program. It is intuitively more reasonable to expect that the working place environment (and averaged over the movements of the radiation worker) produces an effective near-isotropic radiation field for a "chest-worn" or an extremity dosimeter. In this case the more appropriate beta correction factor for the "skin" chip would be 2.73 ± 0.77 and for the extremity dosimeter 4.42 ± 1.17. These values are 57% and 120% greater, respectively, than the current values used by Ontario Hydro.

CONCLUSIONS

The very large spread in beta correction factors clearly indicates the need for a dosimetry system based on thinner detectors for both the "skin" chip and the extremity dosemeters. With the currently used thickness of 100 mg cm^{-2} and 240 mg cm^{-2} used by Ontario Hydro, the appropriate beta correction factors vary by approximately one order of magnitude depending on the source of the radiation field, the angular distribution of the radiation field, the source - TLD distance, etc.... The only practical method to reduce these large uncertainties is to adopt detector thicknesses as low as possible and still consistent with other dosimetric requirements such as minimum detectable dose, ruggedness, etc... To demonstrate the benefits of decreased thickness we have calculated the beta correction factors for representative and extreme spectra for 40 and 20 mg cm^{-2} chips. These particular thicknesses were chosen to correspond to commercially available LiF-TLDs. The benefits of decreased thickness are dramatic. The average value of the beta correction factor decreases from 4.80 for a 240 mg cm^{-2} chip to 1.29 for a 20 mg cm^{-2} chip. In addition, the spread in beta correction factors also dramatically decreases, from 4.80 ± 2.1 (44%) for the 240 mg cm^{-2} chip to 1.29 ± 0.1 (8%) for the 20 mg cm^{-2} chip. Clearly a 20 mg cm^{-2} chip of high sensitivity is the optimum choice for superficial dose estimation due to beta rays.

ACKNOWLEDGEMENTS

This research was supported by the CANDU Owner's Group (COG) Health and Safety Program.

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DOSIMO

A NATIONAL COMPUTERIZED SYSTEM FOR MONITORING OPERATIONAL RADIATION EXPOSURE

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GIIN General Secretary

INTRODUCTION

In parallel to the expansion of the number of French nuclear power plant units in operation, maintenance actions have multiplied, which has meant calling upon the services of a growing number of increasingly specialized workers. It has therefore proved necessary to reinforce the radiation dose rate surveillance of these workers.

As a result, certain companies decided to set up their own occupational radiation dose monitoring system, in addition to mandatory monitoring by the OPRI[1], using dosimeters, generally electronic or thermoluminescent film badges, supplied by the subcontractor companies or nuclear facility operators. This enables acquiring fast and accurate knowledge of the radiation doses received by the workers.

For this type of surveillance to be fully efficient, a data centralization system was required, able to provide frequent, even daily readings if necessary, of the dose received during the current month or for any period of time, up to the sum of the doses accumulated over five years.

A first step forward has already been taken by EDF with the implementation of the Dosinat computer application, which centralizes data on the occupational radiation exposure of EDF agents and subcontractor company personnel in all French nuclear power plants.

The difficulties encountered in monitoring transient workers confirmed that a monitoring system not restricted to nuclear power plants alone was necessary (see the Birraux report, the CSSIN working group, etc.).

This is why EDF and the GIIN, joined by the CEA, Cogema and the DGA[2], decided to set up a system covering all subcontractor company personnel in the nuclear field. This system, called Dosimo, is managed by the GIIN.

The regulations in force reinforce the interest of the Dosimo system, which is expected to extend throughout Europe, then worldwide.

Euratom directive No. 90/641, published in December 1990, specified the responsibilities of nuclear facility operators and companies employing personnel in the nuclear field, with respect to radiation dose rate monitoring. The draft revisions of Euratom directives Nos. 80/836 and 84/467, which set basic radiological protection standards, confirms these responsibilities and integrates the recommendations of CIPR publication No. 60, leading to a decrease in the

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[1] OPRI: Office de Protection contre les Rayonnements Ionisants (Bureau for Protection from Ionizing Radiation).

[2] CEA: French Atomic Energy Commission.
Cogema: Compagnie Générale des Matières Nucléaires.
DGA: Délégation Générale à l'Armement (an agency of the French Ministry of Defense).
GIIN: Groupement Intersyndical de l'Industrie Nucléaire (Nuclear Industry Trade Group).

allowable dose rate limits applicable to workers, and the implementation of measures to reduce workers' individual doses in accordance with the ALARA[3] optimization principle.

This evolution makes it necessary for both nuclear facility operators and subcontractor companies to have rapid, accurate, and complete knowledge of the doses received by personnel. The Dosimo system fits perfectly into the framework of these new requirements.

OBJECTIVE

The objective of the Dosimo system is to provide a close follow-up of workers' occupational dose rates.

To do so, Dosimo gathers the readings of occupational dose rates:

- Measured by nuclear facility operators using the Dosimo system during operations performed by subcontractor company personnel,
- Transmitted by the subcontractor companies employing the personnel in other cases.

Via Dosimo, this information is at the disposal of:

- Subcontractor companies, as concerns their own employees,
- The employees themselves, and
- Nuclear plant operators, for all subcontractor company personnel.

It will be made available as needed to company doctors, via employers.

FUNCTIONALITIES

Management of non-dose rate related data:
A set of functions managed by the Dosimo system enables identifying all personnel, nuclear facility operators, and subcontractor companies.

Management of dose rate data:
A set of functions enables:

- Manually or automatically downloading of the doses received by workers,
- Preserving monthly readings for five years for each worker,
- Reading the dose rate received by each worker for the current month, up to the dose accumulated over five years,
- Guaranteeing that confidentiality rules are respected.

ORGANIZATION

Dosimo inputs:
Subcontractor companies transmit identifying data on their companies and personnel concerned, and notify the departure of the latter from these companies.

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3. ALARA: As Low As Reasonably Achievable.

Nuclear facility operators enter the readings of individual workers' occupational dose rates into the system; they are responsible for the data they enter and they alone are authorized to make changes. If need be, the sucontracting companies complete these data with readings of doses received offsite, or abroad.

USE OF THE DATA

The information gathered can be used:

- By the representative of the company holding the password. The password only gives access to data concerning company personnel. The company is thus informed of the doses received by its personnel, and can compare the results with set objectives, estimate the efficiency of its radiological protection measures, and use this experience feedback to prepare subsequent operations.

- By persons designated by the nuclear facility operator, holding the password. These persons have access only to data on fully identified workers.
  The data enables the operator, in collaboration with the subcontractor company, to implement preventive measures for the operation envisaged, taking into account the occupational dose situation of the workers, and thus optimize radiological protection (ALARA).

- By the GIIN, which manages the system. The GIIN:

  - Receives requests for connection to the system,
  - Issues the passwords,
  - Ensures data consistency and contributes to solving anomalies,
  - Ensures proper operation of the system, and
  - Is authorized to answer any request concerning Dosimo.

CONFIDENTIALITY RULES

Approval of the Dosimo system was received from the CNIL (National Committee on Information Systems and Personal Freedom) on December 21, 1993. In this framework, a worker, on request from the GIIN, can obtain information concerning himself.

Employers are bound by a charter to respect the rules concerning the confidentiality and use of the data supplied by Dosimo.

Nuclear facility operators and subcontractor companies undertake to limit access to Dosimo data to specifically designated persons, on a "need to know" basis.

COMPARISON OF NEUTRON DOSE MEASURED BY ALBEDO TLD AND ETCHED TRACKS DETECTOR AT PNC PLUTONIUM FUEL FACILITIES

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INTRODUCTION

Power Reactor and Nuclear Fuel Development Corporation(PNC) has fabricated Plutonium and Uranium Mixed OXide(MOX) fuel for FBR MONJU at Tokai works. In this site, PNC/Panasonic albedo TLDs/1/ are used for personal neutron monitoring. And a part of workers wore Etched Tracks Detector (ETD) combined with TLD in order to check the accuracy of the neutron dose estimated by albedo TLD. In this paper, the neutron dose measured by TLD and ETD in the routine monitoring is compared at PNC plutonium fuel facilities.

DOSEMETER DESIGNS

Figure 1 shows a cross-sectional view of PNC/Panasonic albedo TLD. This dosemeter consists of eight TLD elements, four of them for beta/gamma ray and the other for neutron. Element 1 is $^{7}Li_2{}^{11}B_4O_7(Cu)$ for compensation of gamma component, element 2 and 4 is $^{6}Li_2{}^{10}B_4O_7(Cu)$ to mainly measure thermal neutrons of the external fields and albedo neutrons respectively. Neutron dose H is given by the next equation,

$$H=Kn \cdot Tn=Kn \cdot (T4-T1-C \cdot (T2-T3)) \quad (1)$$

where T1,T2,T3 and T4 are thermoluminescence readings in unit R, calibrated to ^{137}Cs exposure, and Kn is the conversion factor from albedo neutron reading, Tn, to neutron dose. Energy and angular response of this dosemeter has been investigated by mono-energetic neutron fields/2/, moderated neutron fields, and also calculated by Monte-Carlo method. The conversion factor Kn is very energy dependent and appropriate value must be used to application areas. Default value of Kn for routine monitoring was determined experimentally by the comparison between TLD and neutron dose equivalent detector, "rem counter", in the typical workplace.

In this comparison study, we used Neutrak badge commercially available from Nagase- Landauer Ltd. as Etched Tracks Detector(ETD). This detector material is poly-allyl diglicol carbonate (trade name CR-39) and has a dimension size 10 x 5 x 1mm and high density polyethylene film(1mm) is attached in front of etched layer. Neutron dose is assessed by counted etch-pits N multiplied by conversion factor.

RADIATION FIELDS

The dominant nuclides which cause external exposure in the plutonium facility are ^{238}Pu,^{240}Pu and ^{241}Am. These emit gamma-ray(low energy X-ray), spontaneous fission neutrons and (alpha,n) reaction neutrons. Glove-box is shielded by polymethyl methacrylate(PMMA) and/or lead-contained PMMA, total thickness is about 50mm. Neutron energy spectra at various points in the workplace are measured by INS type multi-moderator ^{3}He detector/3/ and calculated by unfolding method. Average neutron energy in the typical workplace is about 1MeV, and the neutrons with energies above 100keV dominates approximately 90% of total neutron dose.

COMPARISON OF THE NEUTRON DOSE

Total 1,200 workers wore albedo TLD combined ETD at the fabrication process of MOX fuel from April, 1993 to March, 1994, and significant neutron dose were measured by TLD and/or ETD of 413

workers. Figure 2 shows the comparison between the albedo neutron reading Tn and counted etch-pits N. Superimposed lines present the maximum/minimum ratio obtained by the in-the-field experiments, in which albedo TLD and ETD are mounted on the water phantom installed in the typical workplace. The relation between N and Tn obtained by the combined dosemeter worn by workers agrees very well to that by experiments. Number of etch-pits is proportional to albedo reading and the average of N/Tn ratio is 0.3 with a standard deviation of 30% except the low dose range(below the detection limit, 0.2mSv). Figure 3 shows frequency of the quotients of the neutron dose measured by ETD and TLD, and the ratio are centered about 1.0 and almost within a margin of factor 2.

Though these dosemeters have a quite different energy response(TLD is sensitive to slow neutron, ETD fast neutron only), N/Tn ratio is constant. This results mean that the neutron has a uniform energy distribution in the typical workplace and TLD doesn't need any corrections, i.e. energy dependence at different workplace, for assessment of neutron dose.

CONVENTIONAL CALIBRATION METHOD

We have investigated the relation between N/Tn ratio and Kn at various neutron fields. Figure 4 shows the relation between the N/Tn and Kn. Though N/Tn ratio is observed in the range 0.3 to 10 over the average neutron energy range 0.5 to 5MeV, Kn is proportional to N/Tn with the next relation.

$$Kn=Ka\cdot 13.7\cdot(N/Tn)^{0.92} \qquad (2)$$

where Ka is the angular dependence correction factor since these two dosemeter has a different angular response. Factor Ka varies from 1.0(perpendicular) to about 2.0, depending on incident angle, but the appropriate value of Ka is found to be fixed as 1.5 not to underestimate the neutron dose in the stray neutron fields encountered in ordinary work.

Therefore we propose the conventional calibration method for albedo TLD corresponding to the ratio of measured value by TLD and ETD. Using ETD as relative neutron monitor to study average N/Tn ratio in the typical workplaces, we can classify the types of neutron fields, where special Kn value is needed, and easily get appropriate Kn. As example, if the average neutron energy in the workplace is lowered by additional shielding around glove-boxes and N/Tn ratio changes remarkably, we may re-evaluate Kn value using equation (2).

CONCLUSION

The summary of this comparison study of TLD and ETD in the PNC plutonium fuel facilities are as follows.

(1) Neutron dose measured by albedo TLD has a good agreement with that by ETD.

(2) The ratio of N/Tn ratio obtained from two dosemeters worn by workers is almost constant in the workplace. Any location-dependent variations of N/Tn ratio are negligible except for special work conditions.

(3) The ratio of N/Tn ratio is an ideal index for Kn and we can easily get appropriate value of Kn using that ratio.

(4) Use of ETD as relative neutron monitor for personal monitoring, we can easily re-evaluate Kn.

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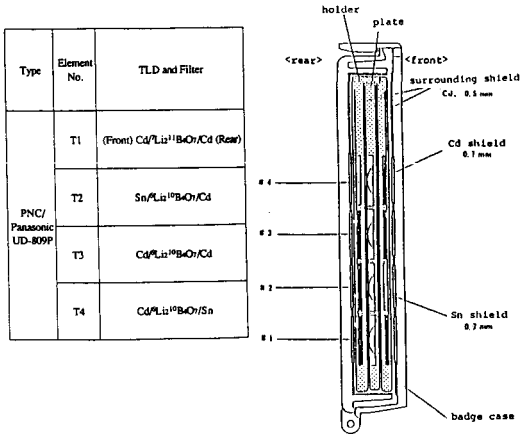

Type	Element No.	TLD and Filter
PNC/ Panasonic UD-809P	T1	(Front) Cd/^{7}Li$_2$^{11}B$_4$O$_7$/Cd (Rear)
	T2	Sn/^{7}Li$_2$^{10}B$_4$O$_7$/Cd
	T3	Cd/^{7}Li$_2$^{10}B$_4$O$_7$/Cd
	T4	Cd/^{7}Li$_2$^{10}B$_4$O$_7$/Sn

Fig.1 Cross-sectional view of PNC/Panasonic albedo TLD (type; UD-809P)

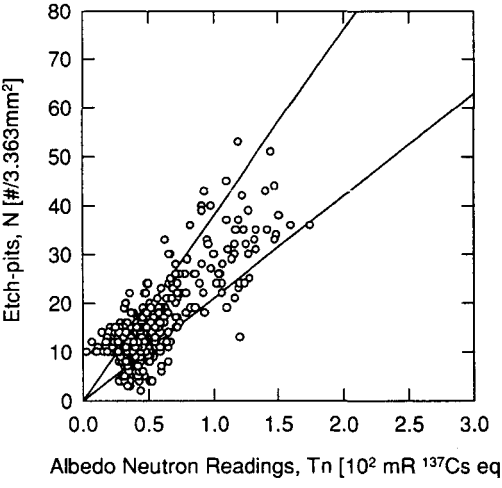

Fig.2 Comparison between the albedo neutron reading Tn and counted etch-pits N
(circle; worn by workers from April,1993 to March,1994
solid line; maximum/minimum ratio obtained by the in-the-field experiments)

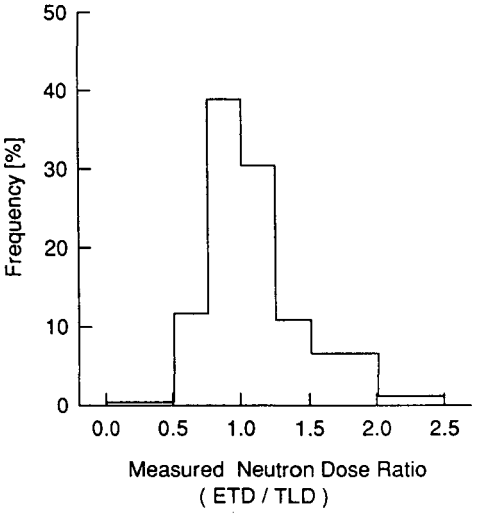

Fig.3 Frequency of the quotient of the neutron dose measured by ETD and TLD

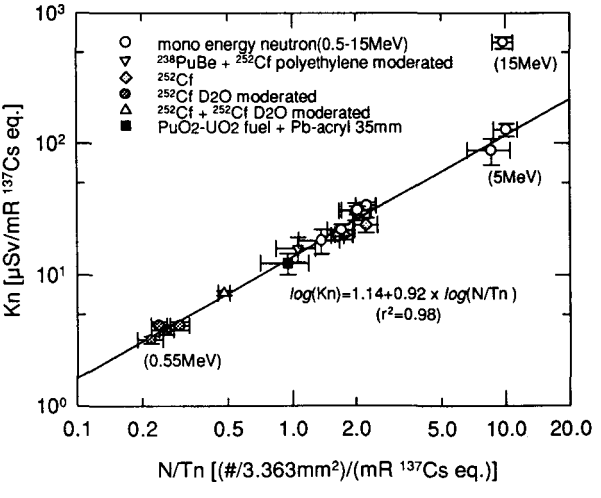

Fig.4 Relation between N/Tn ratio and Kn at various neutron fields

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Abstract No. 80203
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PAPER TITLE Radionuclide correlation in aerosol effluents from NPP V1 Jaslovské Bohunice in Slovakia.

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MAJOR SCIENTIFIC TOPIC NUMBER 5.2 (see page 7)

ABSTRACT (See instructions overleaf)

The aerosol samples from ventilation stack of NPP V1 Jaslovské Bohunice have been collected for the long time. The sampling is carried out by aerosol sampler (Model Sierra Misco 8080) with a flow control system. The device is placed at the bottom of the stack, where the air from all ventilation ducts is already mixed. The air volume throughput for one filter is several thousands of m^3. Sensitive gamma spectrometric analysis of aerosol samples is performed using germanium detectors. Radiochemical analysis followed by beta, resp. alpha spectrometry has been carried out, too. About 20 radionuclides with activities in the range 10^{1}-10^{-5} Bq/m^3 was studied. The essential part of total volume activity of evaluated radionuclides is connected with ^{110m}Ag, ^{58}Co, ^{60}Co, ^{54}Mn and ^{51}Cr.

From the large file of experimental data obtained in the years 1985-1995 correlation between individual radionuclides was performed. The aim of the study is to find out easily measurable "marker" radionuclides which will enable either to estimate the contribution to the dose from not easily measurable radionuclides or to perform more sensitive analysis of samples indicated by higher activity of marker radionuclides. The correlation coefficients were determined for the all combinations of evaluated radionuclides and they were compared with critical values of correlation coefficient for selected level of significance. Correlation was studied from the point of view of reactors' operation mode. Its variation in time was studied, too.

EXPERIENCE ON STATISTICAL ANALYSES OF OCCUPATIONAL DOSE DISTRIBUTIONS IN NPP EBO JASLOVSKE BOHUNICE

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INTRODUCTION

We analysed the basic characteristics of occupational effective doses, that are:

- the average annual effective dose. E,
- the annual collective effective dose, S,
- the fraction of workers exposed to annual dose exceeding 5 mSv, NR_E,
- the collective dose distribution ratio, SR_E, for E exceeding 5 mSv,

collected ad NPP A1, V1, V2 and EBO, (as the whole), in the site Jaslovske Bohunice, by several statistical methods of dose distribution [1]. The doses were summarized during the years 1990 to 1994.

For the assessment of above mentioned parameters three different dose distribution models were used:

a) direct method (DIR), using the distribution of workers by dose ranges. The midpoint of each interval has been used as **geometric mean** of dose ranges [2],

b) log-normal distribution of doses (LN), with parameters α, β estimated by regression analysis of the function:

$$\nu_i = \alpha + \beta \ln(E_i),$$

where ν_i is the quantil and E_i is the upper bound of i-th dose range.

c) hybrid log-normal distribution (HLN) with parameters α, β, γ, estimated by regression analysis of the function:

$$\nu_i = \alpha \ln(E_i) + \beta E_i + \gamma$$

Values of quantils, ν_i were calculated by standarts statistical methods [3]

DATA USED FOR ANALYSES

TABLE 1 NPP JASLOVSKÉ BOHUNICE

Slovak republik		V1	V2	A1	EBO
Reactor	Typ	VVER 230	VVER 213	HWR	ALL
Nr.of reactors	[1]	2	2	1	4
Install capacity	[GW]	0.88	0.88	a)	1.76
Energy generated	[GWa]	0.62	0.62	a)	1.30

a) reactor in decommissing

TABLE II. DISTRIBUTION OF WORKERS BY THE ANNUAL DOSES IN NPP EBO IN THE YEARS 1990 - 94

E [mSv.a-1]	1990	1991	1992	1993	1994	1990-94
< 0.1	1433	1430	1145	1018	1342	6368
< 0.5	420	471	731	722	647	2991
< 1	168	212	363	319	220	1282
< 2	195	183	339	324	220	1261
< 5	197	225	386	353	260	1421
< 10	67	168	215	237	87	774
< 15	10	40	54	36	8	148
< 20	4	10	8	5	1	28
< 30	2	2	1	0	2	7
ALL	2496	2741	3242	3014	2787	14280
MON [a)]	1063	1311	2097	1996	1445	7912
S [man.Sv]	1.83	3.14	4.53	4.18	2.19	15.87
E_{AVG}(ALL)	0.73	1.15	1.40	1.39	0.79	1.11
E_{AVG}(MON)	1.72	2.40	2.16	2.09	1.51	2.01

[a)] Persons over 0.1 mSv per year

RESULTS

TABLE III. AVERAGE ANNUAL DOSES (E_{AVG}), NR_E AND SR_E ESTIMATED BY VARIOUS DOSE DISTRIBUTION MODELS

E_{AVG} [mSv.a^{-1}]						
DATA	1.72	2.40	2.16	2.09	1.51	2.01
DIR	1.71	2.38	2.13	2.08	1.53	1.99
LN	1.61	1.94	1.75	1.67	1.38	1.63
HLN	1.68	2.26	1.95	2.05	1.42	1.88
NR(E>5mSv) [%]						
DIR	3.3	8.0	8.6	9.2	3.5	6.7
LN	1.8	2.7	3.3	3.0	1.7	2.1
HLN	3.4	6.4	6.9	7.8	3.0	5.7
SR(E>5mSv) [%]						
DIR	39.0	60.7	52.4	52.8	35.1	50.2
LN	32.8	38.4	31.5	31.2	27.1	27.2
HLN	43.8	54.8	47.1	45.4	36.1	46.2

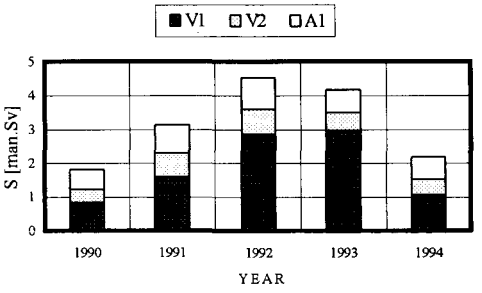

Figure 1. Collective doses in the NPP V1, V2 and A1 (1990-94).

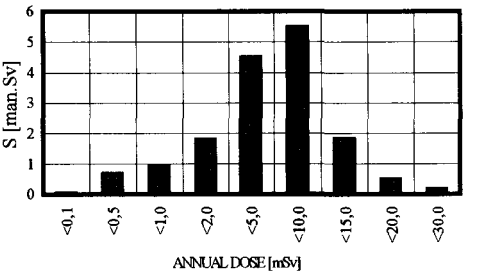

Figure 2. Distribution of collective doses according to the range of individual dose(1990-94).

CONCLUSION

Our experience with the statstical analyses of occupational doses in NPP EBO confirmed that:

- The assessment of annual individual and collective doses by using three methods of dose distributions shows small differences. The average values of annual effective doses for monitored persons are in the range of 1.4 to 2.0 mSv. LN distribution show lower values.
- The LN distribution method gives significantly lower values of NR_E and SR_E ,
- Direct method and HLN distribution model provides practically the same results.

According to Basic Safety Standards [4] we suppose that 5 mSv could be an investigation level (instead of until now used three tenths of the basic limit, that is 15 mSv). This value is exceeded by less than ten percent of workers.

The fraction of collective doses due to the effective doses higher than 5 mSv, reaches the value of about 50%..

In the light of the given results one can see that the introduction of the reduced basic annual limits of effective dose (20 mSv/a), will not create a serious problem by organizing the work in NPP EBO.

The fraction of the ''lost'' collective efffective doses, with respect to the chosen recording level of 0.1 mSv , does not exceed 2%.

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SAFETY CULTURE ASPECTS OF MANAGING FOR SAFETY - EXPERIENCE OF A LARGE NUCLEAR REPROCESSING SITE

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INTRODUCTION

The Nuclear Industry is going through turbulent times both in terms of public acceptance and business issues. Safety is one area which impacts on whether the business is allowed to continue, and how an organisation organises itself. The need to cut costs to make nuclear power a viable energy resource, has forced the Nuclear Utilities to review manning policies, and management style, and in particular how to maintain safety standards during a period of change, and ultimately support continuing improvement of standards.

The shrinking workforce requires a new style of management, one that depends more on the people of the organisation taking responsibility for safety at all levels of the organisation. Not only personal safety but the safety of their colleagues, general public and the environment. With reduced number of people, and 'delayering' of an organisation, managers have to switch to a facilitating or co-ordinating role, as the controlling role e.g. close supervision, is no longer possible or effective. This often requires a culture change. In the case of BNFL the changes were not confined to the recent past, but began in the 1970's with the civil service culture, moving to a project culture in the 1980's with the building and commissioning of major projects, and needing to change to an business/operations culture to secure work for the company into the 21 Century.

The safety culture of an organisation is indivisible from the company culture, each aspect of a culture influences the whole and so the balance between business, safety and quality, has to be managed.

BNFL provides a full fuel cycle service to nuclear power plants, and associated services to many national and international organisations. The following notes are taken from the work carried out in the company, and mostly at the Nuclear Reprocessing and Waste storage Site at Sellafield, based in the North West of England. Following the recent re-organisation, the site now employs 6,200 people and has a further 1,500 contractors working on construction activities on the site. Activities on the site range from remote handling to hands on tasks, involving highly active materials to low level waste. These activities take place in old plant (> 30 years old) and new plant, and within the full life cycle of plants i.e. design, build, commission, operate, modify, decommission/re deploy, care and maintenance. Multiple types of tasks under many different conditions.

JUDGING IMPROVEMENT IN SAFETY CULTURE

In order to measure or judge improvement in anything, the starting point has to be determined. The two questions "What have we got?" and "Where are we ?" have to be answered before sensible major programmes for improvement can be formed and implemented. Without these answers, any action will be blind and some may cause the opposite effect to that desired. Any technique(s) used for judging Safety Culture must be able to identify :

1. *Perceptions - What are the work force's opinions on the safety of their plant? Do they feel safe?*
2. *Attitudes- how do the workforce see their and others responsibility for safety? Are they proactive or are they passive? Is safety at work somebody else's job?*
3. *Strengths and Weaknesses - What do they do well? How do they compare with companies with 'good' safety performance?*
4. *Beliefs vs. Behaviours -Is what they say what they do?*

Once these factors have been identified management can then plan for change and monitor any change. At BNFL Sellafield, two methods were developed to identify "What have we got?" and "Where are we now?" .

Sellafield Safety Attitude Survey

This was designed to identify the Safety Culture on the Sellafield Site and was planned in 3 stages.

1. *Focus group- 50 people were brought together as a discussion group to identify the topics and issues of main importance to safety on the Sellafield Site. A questionnaire for a much larger sample of people was developed.*
2. *Pilot Study - 161 people answered the questionnaire, and the technical validity of the questions, and the administration logistics were tested and improved.*
3. *Full Survey -During Autumn 1991, 5295 questionnaires 172 attitude questions were completed. These were analysed by Professor T Lee of St Andrews University.*

There were many detailed findings giving the strengths and weaknesses of the work force's safety attitude, and the results also gave indication where effort should be targeted. The survey also identified attitudes helping BNFL to develop its safety culture and contributing to improving safety performance, and those that were not.

Sellafield Safety Culture Review

A complementary method for judging safety culture was developed by BNFL using the foundation of IAEA's INSAG -4 *(ref. 1)* and the UK Government's Advisory Committee on Human Factors 3rd Report "Organising for Safety" *(ref. 2)*. The review was developed and applied for the managers of 3 plants who were uncertain as to the true state of safety attitude on their plants, and how they influenced their workforce's safety culture.

Targeted interviews using a question framework developed from the above texts were used by the interviewers as a guideline to structure the interviews. Selection of personnel interviewed (both individuals/groups, shifts/days) were based on the role of the personnel within the plants, and included service groups, the plant's independent safety advisors and the Government Nuclear Installation (NII) inspectors directly involved in the inspection of the plant. The latter were included, as some measure of behaviour modification became apparent as the perceived result of contact with the inspectors.

The interviews identified the perceptions and attitudes associated with the plant's safety culture, which were then analysed to identify where they matched, conflicted, or were discontinuous. This process identified dominant attitudes and areas of conflict, along with the strengths and weaknesses forming their Safety Culture. From this, the plant developed a programme, building on their strengths to eliminate or neutralise their weaknesses, using participative team methods to engender ownership of plant and performance.

Attitude Survey and Safety Culture Review are complementary. They can be used selectively to address different aspects of a organisation's Safety Culture. The Safety Attitude survey, once developed, is easy to apply, and the analysis has a statistical basis so that confidence in the results can be judged readily. It gives a good overview of the dominant safety culture present in a population, and identifies the sections or profile of the workforce holding a dominant attitude. The Safety Culture Review looks at Safety Culture within a Plant's management system. Trained and experienced interviewers are required, and the analysis of the data is more complex. However it is able to follow the issues through and identify root causes of difficulties, and identify the possible root of the attitudes expressed. Judgement of behaviour is also able to be incorporated.

THE BUSINESS CASE FOR PRO-ACTIVE SAFETY MANAGEMENT

For an organisation to use resources to develop or improve their safety culture, tangible benefits must be evident. Traditional benefits of a good safety culture are : **Reduction of costs** - *e.g. associated with waste, accidents, staff turnover, equipment* and **Legal Imperatives**- *e.g. costs of insurance, civil action, prosecution.*

However, a more positive business case for a good safety culture can also be made with:

The Enhancement of Business Opportunities

A good safety culture can gain a company advantages when bidding for work. The improvement of quality and reliability, and associated gains in efficiency and effectiveness, can be directly related to having a good safety culture within an open management system.

A good safety performance is also an assisting factor for good industrial relations, and for promoting a culture of excellence. A company who wish to recruit quality personnel also require a good image. This in turn will assist the company to maintain a high public and business profile as a quality company. Potential customers prefer to do business with a non-controversial company.

The moral imperative

All industry has a responsibility to the people they employ, the general public, and the environment they can affect. Without this sense of responsibility there would be no control over actions by companies or individuals within the company. Damage due to irresponsibility cannot be totally quantified, and society places a value on morality. A good safety culture is one way of promoting and maintaining the sense of responsibility within a company's workforce, without the need for restrictive, punitive and costly management systems.

SHAPING AND CHANGING A SAFETY CULTURE

BNFL had to identify what culture they needed to develop a successful and smaller workforce for operating or decommissioning their existing plants, and for the new workforce that was to operate their new Thermal Oxide Reprocessing Plant (THORP). Workforce reorganisation required a clear statement of values and behaviours before team building could commence. The shaping action was developed and installed locally, for experience had shown that Central or HQ 'initiatives' can be forced through and led by the senior management, but eventually the initiatives can fail due to lack of local support. In a large organisation different sub-cultures require different shaping processes to move towards the desired culture. Within BNFL, the common theme was participation and involvement of all the workforce, led and supported by senior management. By promoting ownership of locally developed and installed shaping processes, the resultant improvements are normally faster and are more likely to endure.

All of the shaping processes identified to the workforce the behaviour demanded by the company. Once benefits to the individual are identified e.g. dose reduction, job security, attitudes are more likely to change. These shaped attitudes however, require commitment by management, consistency of application of behaviour demands and the benefits to individual and peer group identified and publicised.

One immediate benefit is that once an attitude change has been achieved, peer group pressure becomes a strong reinforcer i.e. "the way we do things around here" which then sustains the behaviours you require. This is a lot more effective and cost effective than continuous managerial pressure and policing.

Involving all persons in the workforce uses all the skills and ideas of the workforce, and improves communication allowing a quicker response to circumstances. There is a need to develop management to be receptive to workforce ideas, and comments, so that ideas are freely offered. Management in this environment are also more likely to be told of problems and their consequences earlier, before damage to people, plant or organisation occurs.

In summary, development of safety culture has a lot to offer an organisation in terms of safety, quality and productivity. The final step of 'empowerment' can only be achieved by trust in fellow worker's attitudes, and abilities, and sustained by continued empowerment even under difficult conditions. Improvement of the safety culture has to build on the existing strengths of the organisation and recognise the associated weaknesses. Once an improvement programme has been derived, management and workers must have patience with the process, and aim to monitor movement at regular intervals. In a large organisation progress will not be uniform across all departments, and actions successful in one sub-culture may be unsuccessful in another. Identifying visible successes and publicising them will encourage management and workers, and serve to silence persons in the organisation who wish to maintain the status quo. For an existing organisation, a step by step approach is a less painful method for change but ultimately an organisation cannot afford to carry 'passengers'. By involvement, the workforce can start to participate, by participation ownership develops, and finally by true empowerment, actions involving safety, production and quality are put in place at all levels of the organisation, naturally, efficiently and continuously.

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1 Introduction

BNFL's Thermal Oxide Reprocessing Plant (Thorp) reprocesses uranium oxide fuel assemblies which have been irradiated in thermal reactors in the UK and overseas. Plans for the plant were first announced in 1974. Application for planning permission was submitted in 1977, and government permission to construct the plant was granted after the Windscale enquiry in 1977. The plant was given the license to start active commissioning in Head End in early 1994, and then in Chemical Plants in late 1994. Presently the whole of the process is being challenged in a planned commissioning strategy which will last into 1996.
Thorp is designed to reprocess the spent oxide fuel into uranium trioxide (UO_3) and plutonium dioxide (PuO_2).
The Thorp complex can be essentially broken down into three distinct areas:
Thorp Receipt and Storage
Provides pond storage for fuel awaiting reprocessing in Thorp.
Head End
Fuel is transferred from Receipt and Storage into the Feed Pond where it is monitored to check fissile content, burn up and cooling time. The individual fuel assemblies for LWR fuel, or cans in the case of AGR fuel, are transferred onto the shear elevator and carried up to the shear cave. The fuel is sheared into small lengths to optimise the dissolution of the fuel inside the cladding. The sheared fuel and cladding debris is directed via a chute into one of three dissolvers, each with a nominal 1.8 teU capacity and dissolved in 8M nitric acid for approximately 16 hours. The cladding hulls are retained in a removable basket and sent for encapsulation. Insoluble fission products and fine particles of cladding are removed by centrifugation. Clarified dissolver solution is then accounted for by measurements taken for volume, mass and isotopic composition. Following this, the solution is transferred to buffer storage tanks and fed onto the Chemical Separation area.
Chemical Separation
The liquor is transferred to the Chemical Separation area where it undergoes first cycle separation in pulsed columns to separate out uranium, plutonium and highly active fission products. After product removal, the remaining liquor is stripped of solvent and sent for evaporation and thereafter vitrification. Separated plutonium and uranium streams then undergo purification and product finishing to PuO_2 and UO_3 respectively prior to containerisation and storage in dedicated facilities.

This paper will discuss Radiation Protection Experience during the commissioning of Thorp Head End and Chemical Plants, and will look at issues such as plant design and its effect on contamination control, the routine survey philosophy, the commissioning strategy, dose control and environmental discharges.

2 Plant Design

The Thorp design relies on biological shielding in the form of bulk concrete for the protection of personnel from radiation exposure. All shielding assessments for Thorp Head End and Chemical Plants have been based on PWR fuel with an irradiation period of 40 Gwd/t, a rating of 40 Mw/t and a cooling period of 5 years.
Many other factors have been built into the shielding design of Thorp, eg all walls are much thicker than required purely for shielding purposes because of structural and seismic requirements, all shielding calculations have assumed that vessels and pipework are full of active liquor, and all sources of potential radiation exposure have been diverted away from main occupancy areas as far as practicable.
Unrestricted access areas such as corridors and offices, are designed to a target dose rate of 1 μSv/hr, as compared with the legal limit for these areas of 7.5 uSv/hr. (See Fig 1 *Classification of Areas*).
The Thorp design has also aimed to reduce radiation exposure and occupancy times in higher potential radiation areas, by use of remote control, remote manipulators, remote reading of instruments and gamma interlocks.
Processes requiring direct operator intervention have been eliminated as far as practicable, by such measures as the installation of ejection systems instead of pumps, automatic liquor sampling, pneumatic transfers of samples to areas for analysis and remote filter changing within filter caves.
As an aid to the design of localised shielding, and to control radiation exposure, a system of area classification has been adopted in Thorp, according to expected/potential radiation and contamination levels in each area.
The classification system is comprised of 5 zones, R1-R5 radiation and C1-C5 contamination. (See Fig 1 - *Classification of Areas*).
The radiation and contamination classifications of an area are carried out independently of each other, against such criteria as expected and potential radiation and contamination levels in the area, occupancy time in the area and the type of work to be undertaken in the area.
Thorp is designed so as to restrict the maximum individual whole body dose, ie the sum of effective dose equivalent from external radiation, and committed effective dose equivalent from internal radiation, to less than 15 mSv/year, (the legal limit being 50 mSv/year), with the worker group average dose not exceeding 5 mSv/year.
Whole body dose in Thorp is comprised almost entirely of external radiation exposure. The effect of internal exposure is minimal due to the design of the plant in terms of contamination control, ie Ventilation, C1-C5, Sub-changerooms, Containment etc. Internal exposure therefore, is not routinely assessed on Thorp, but carried out on a campaign basis only.
The Thorp design target for extremity dose is that any individual should not exceed 300 mSv/year, (the legal limit being 500 mSv/year) with a worker group average not exceeding 200 mSv/year.
The integrity of plant shielding, confirmation that the plant is operating within target radiation and contamination levels, and that operational radiation exposure is minimised, as well as multitudinous operational parameters, will be tested throughout the period of Level 5 commissioning.

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3 Commissioning Strategy

The overall strategy for the commissioning of Thorp, defined the need for 5 levels of commissioning, where the plant and process were challenged with water, simulants, chemicals and following the designation of the plant as a controlled area, uranium and ultimately irradiated fuel in 'bands' of increasing irradiation, band 1 being the least irradiated and band 4 being similar to reference fuel. (See Fig 2 - Fuel Bands).

The objectives of Level 5 active commissioning are to demonstrate the safety and technical acceptability of Thorp operations, test all related and supporting systems, and confirm radiation and contamination control are as per design targets.

Commissioning is undertaken using System Performance Demonstration Definition Documents (SPDDDs), which identify the intent, objectives, pre-requisites, resources, material and sequencing of activities associated with the performance of a particular test.

Some radiological Level 5 commissioning tests include; trend monitoring of aerial discharges, routine monitoring of plant radiation, contamination and airborne activity levels and confirmation that they are within design targets, and demonstration that occupational exposure to personnel is within design targets.

To date, all radiological SPDs carried out have confirmed that the plant is functioning well within design targets, and no anomalies have been reported.

4 Contamination Control Philosophy

The contamination control philosophy for Thorp is based on controlling contamination in such a way as to keep the normal operating areas free from loose surface contamination. This philosophy was adopted to ensure that the routine issue of personal air samplers, for the assessment of internal dose, was not required. In order to achieve this position, the average surface contamination levels must not exceed about 0.004 Bq/cm^2 (alpha) which is two orders of magnitude more restrictive than the traditional "non-active area target".

To this aim, the plant has been designed into segregated areas C1 to C5 (see Fig 1 *Classification of Areas*). Where C1 is the lowest, ie inactive and C2 areas (operating areas) are comparable with these, up to C5 where there is the greatest possibility of contamination to be generated. The plant has been designed so that the air flows throughout the building from C1 through to C5, with the ventilation, filtering and extract systems becoming more involved the closer they are to the sources of contamination.

For management purposes, the whole of the plant has been classified as a Controlled Area. Entry to the plant is via a main changeroom, which contains lockers, change facilities, showers and monitoring facilities. Plant personnel wear factory basic clothing and shoes which are removed on exit at the barrier in the main changeroom.

The barrier against the spread of contamination is defined at the C2/C3 sub change facility, where there is provision for a change of outer clothing, ie C3 coveralls, gloves, boots and monitoring facilities. As the "C" category is increased so the protective clothing and respiratory requirements are increased.

Appropriate engineering consideration has been given at the workplace to contain both alpha and beta contamination both during design and operation of facilities. The operating philosophy has been to engineer out contamination hazards, via containment such as gloveboxes, ventilation and local barriers at the job.

Activity-In-Air Monitoring

In order to constantly monitor the working area breathing zone for the presence of airborne activity, Thorp has a system of approximately 750 installed activity in air monitors. These instruments are all connected via the Radiological Protection System (RPS) to the Central Control Room (CCR). If an instrument detects airborne activity it alarms locally to warn the local area operators, and in the CCR to enable a response to the alarm to be formulated. There are 3 levels of alarm which are dependant on the amount of activity present in the air.

There are specific monitors which are designated as zonal evacuation monitors. Certain combinations of these monitors in alarm will instigate a zone or plant evacuation.

5 Routine Survey Philosophy

In order to maintain the long term operability of Thorp, it is essential to ensure that radiation and contamination levels of the plant remain within the Thorp target levels as discussed previously (See Fig 1 *Classification of Areas*).

A routine survey schedule was implemented both during Level 4 and Level 5 commissioning which covers every area of Thorp, and identifies the type of radiological survey to be undertaken, ie radiation and/or contamination, the survey frequency (based on occupancy time), the radiological classification and the activities undertaken in the area.

The routine survey philosophy for Thorp defines four main survey types;

Control Monitoring - These general radiation and surface contamination surveys are undertaken in all unrestricted access areas such as corridors, offices and workshops, ie C2 areas.

First Level Plant Status Surveys - are undertaken to warn plant operators of any significant increase in radiation and contamination levels that require immediate remedial action. These surveys focus on areas where radiological problems would be readily detected, such as sub changerooms.

Second Level Plant Status Surveys - are carried out in an audit capacity in all unrestricted access areas, at least once a year.

Audit surveys - are carried out in all key operating areas at least once a year to confirm that radiological control is being maintained, and also to identify any trends in radiological conditions.

6 **Dose Control Philosophy**

The Thorp target for whole body dose is 15 mSv per year for an individual, and 5 mSv for a worker group average.
Personnel working within the Thorp controlled area are designated classified persons in accordance with the Ionising Radiation Regulations 1985, and as such must wear a personal dosemeter as approved by an authorised dosimetry service.
The routine dosemeter for Thorp is currently the film badge, issued on a monthly basis, with quarterly issue being considered for some itinerant worker groups. In addition to routine dose monitoring, dose control is required to run in parallel to ensure that dose targets are not exceeded and that exposure is ALARP. Dose control is carried out by the issue of Merlin Gerin Personal Alarmed Dosemeters (PADs). These PADs are issued on an individual or job specific basis, where there is the potential for enhanced radiation exposure. These are normally issued where gamma doserates are expected to exceed 25 μSv/hr. A guide dose of 50 μSv per day has been adopted in Thorp to maintain dose targets, however, higher doses can be authorised by management on a case by case basis.

Extremity doses are measured in Thorp using Thermoluminescent Dosemeter (TLDs), which are issued when doserates exceed 75 μSv/hr beta gamma. Plant and Operations have been designed such that extremity, internal and neutron exposures are less than one tenth of the relevant dose limit, and therefore, routine assessment is not required.

7 **Environmental Impact of Thorp Discharges**

At the Windscale enquiry (1977), BNFL stated a strategy for the management of effluents, placing specific limits on Thorp discharges:-

1 No member of the most highly exposed (critical) group(s) should receive an annual committed effective dose equivalent >50 μSv from aerial *or* marine discharge routes.
2 By the time of Thorp operations, the sum total of annual exposure (CEDE) from the whole of Sellafield (including historical effects) operations should be less than 500 μSv pa.
3 Radiation exposure to the public should be ALARP (As Low As Reasonably Practicable).

During normal operating conditions (ie design flowsheet 1200 tonnes PWR pa) annual discharges have been calculated to result in critical group doses of ~20 μSv via the aerial pathway and ~24 μSv via the marine pathway. Both are clearly within the 50 μSv limit.
Recent calculations also show that mid way through the base load period (ie Thorp maximum throughput at design flowsheet) total site discharges, including Thorp, will result in critical group doses of ~60 μSv (aerial pathway) and ~190 μSv (liquid pathway). Again, these are clearly within the 500 μSv limit.
Figure 3 shows the aerial discharge authorisation limits for the most significant isotopes for a reprocessing rate of up to 100 tonnes pa and the actual aerial discharges Jan-Dec 94.
The environmental impact of the 1994 discharges is 0.38 μSv, predominantly due to the Isotope I129.
The total annual liquid discharges (Jan-Dec '94) are presented in Fig 4. *Annual Liquid Discharges Jan - Dec 1996.*
The impact of the 1994 liquid discharges is 1.54 μSv, dominated by Co60 and Cs137.
Discussion
To date, the arrangements implemented in Thorp to ensure contamination control and dose reduction, have proven to be successful.
There have been no indication of any contamination migrating from C3/5 areas into the C2 operating areas, with all operating areas remaining at the Thorp target levels. This implies that contamination control measures at the sub changeroom barrier, and the ventilation system are functioning correctly and efficiently.
Only 1% of the total number of routine surveys undertaken in 1994 were found to be above the Thorp target levels, however these were exclusively in sub changerooms, where potentially contamination items, eg coveralls and boots would not be unexpected. These contamination levels were quickly returned below action level by decontamination, or removing the source. No areas of gross surface contamination, or high radiation areas have been found whilst carrying out routine surveys in operations areas.
The shielding of the plant has been tested both routinely by way of regular radiation surveys, and also in support of SPD commissioning checks upon the first challenge to an area with irradiated fuel.
No shielding weaknesses have been discovered, and all operating areas are well below the target of 1 μSv/hr. These SPD checks will be repeated as commissioning progresses to the next fuel band.
The average whole body dose in Thorp during 1994 was 0.93 mSv, well below the Thorp worker group average of 5 mSv. No single individual who had been employed exclusively in Head End or Chemical plants received a whole body dose greater than 2 mSv, again well below the 15 mSv whole body dose limit for an individual in Thorp.
The Thorp extremity dose target for an individual is 300 mSv. No individual exceeded 1% of this target, an indication that remote working and decontamination prior to commencing a task, are both common practices.
All dosimetry campaigns to date, internal, neutron and extremity, have shown no need for systematic assessment of these doses.
During 1994, the aerial discharges from Thorp were well below the authorised levels, with liquid discharges also being very low.
As commissioning progresses towards irradiated fuel of a similar magnitude to reference fuel, aerial and liquid discharges, as well as radiation and contamination levels on the plant will be continually monitored. Shielding checks will continue and dose reduction will be an ongoing trend.
The engineered design of Thorp, together with the Radiological Protection Systems and promotion of good working practices will successfully support the operation of Thorp into the 21st century.

RADIOACTIVE GASEOUS EFFLUENTS MONITOR UPGRADE AT THE BRUCE A NUCLEAR GENERATING STATION

N. Sion

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INTRODUCTION

Ontario Hydro is committed to the effective management and control of the radioactive gaseous effluents from the ventilation pathways within the limits established by the Atomic Energy Control Board (AECB), and within the design and operating targets associated with the ALARA (As Low As Reasonably Achievable) policy.

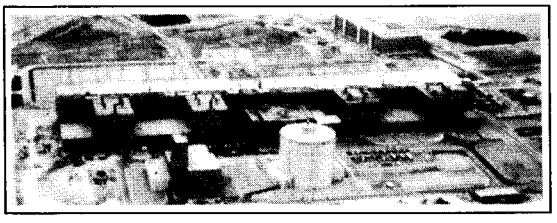

Fig. 1 Aerial View of Bruce A Nuclear Generating Station

The Bruce A nuclear generating station, fig. 1, is located on Lake Huron and is a 4-reactor power plant of 740 MW(e) net each. A major overhaul and upgrade of the radiation monitoring system has just been completed resulting in a system that effectively complies with current Canadian regulatory requirements.

THE MONITORING SYSTEM

Fig. 2 is a much simplified schematic of the Stack Monitoring flow diagram. In essence, the monitor draws a representative isokinetic sample conforming to ANSI 13.1 by a probe located in the stack above the station roof to minimize non-uniformity of the flow profile. The flow is maintained isokinetic and the sampling lines above the station roof are heat traced to forestall potential condensation.

The sampling stream then divides into two. The smaller stream passes through a desiccant which removes any water vapour for subsequent analysis for the presence of tritium. The larger stream is filtered to remove airborne particles and is monitored with a beta detector. The flow continues through a charcoal filter to remove gaseous halogen aerosols in the presence of an energy window to detect the presence of Iodine-131 photopeak. The sample is then pumped through a 7.5 litre Marinelli chamber where it is monitored for noble gas by a 2 channel NaI detector. The Marinelli chamber has a second port through which a germanium detector can be inserted for isotopic identification. Periodic purging of the chamber is performed to obtain updated background.

The monitoring system was supplied by Sorrento Electronics of San Diego whose microprocessor would continuously collect data and transmits hourly to the Control Room. Local data storage, with local alarms and displays are in place should communication between the microprocessor and the Control Room be disrupted.

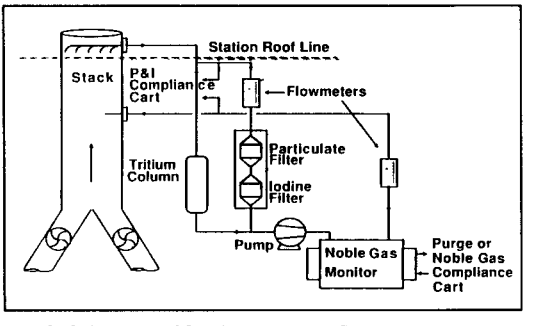

Fig. 2 Schematic of Stack Monitoring System

FUNCTIONAL REQUIREMENTS

The monitoring of the gaseous effluents is based on the most restrictive radionuclides via I-131 for radioiodines, Sr-90 for particulates both measured in curies and predominant matrix of noble gas mixture measured in γCi-MeV, Ref. 1 as well as tritium in water vapour form, measured in Ci.

The monitoring objectives are

For Compliance: where the continuous monitoring demonstrates compliance with regulatory limits and operational targets - by laboratory analysis

$$\text{Lower Limit} = \frac{\text{Concentration of 1\% Weekly Operating Target (WOT)}}{\text{Mean Weekly Flow at the Sample Point}}$$

Upper Limit = Concentration of $\frac{\text{100\% Weekly DEL}}{\text{Mean Weekly Flow at the Sample Point}}$

For Control monitoring: where on-line detectors continuously monitor the sample so that an operator can take corrective action to ensure limits are not exceeded.

Lower Limit = Concentration of $\frac{\text{0.1 Weekly Operating Target (WOT)}}{\text{Mean Weekly Flow at the Sample Point}}$

Upper Limit = Concentration of $\frac{\text{25 Weekly Operating Target (WOT)}}{\text{Mean Weekly Flow at the Sample Point}}$

Radiological releases are reported in Derived Emission Limits (DEL), or fractions thereof. The DEL is the amount of a radionuclide, or group of, which if released from the entire station in one year, would result in a member of the public at the station boundary receiving a whole body maximum permissible dose (500 mR/yr - 5 mSv/yr). The Ontario Hydro self-imposed target limit is 1% DEL (5 mR/yr - 50 μSv/yr).

Fig. 3 The Stack Ventilation Exhaust Fans

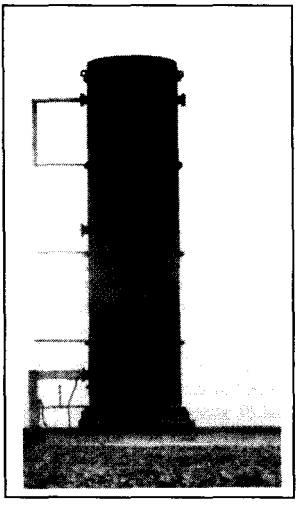

Fig. 4 Stack Probe Location Above Roof

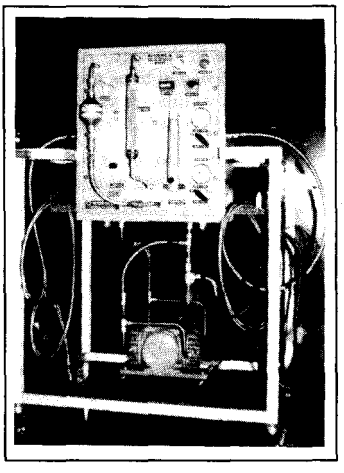

Fig. 5 The Particulates, Iodine and Tritium Compliance Monitoring Cart

For Bruce A the above translates to releases in table 1.

RADIONUCLIDE	1 DEL Ci/week regulatory compliance upper limit	1% DEL = WOT Ci/week Weekly Operating Target	1% WOT Ci/week compliance lower limit	0.1 WOT Ci/week control lower limit	25 WOT Ci/week control upper limit
H^3	$1.95\ E^{+5}$	$1.95\ E^{+3}$	$1.95\ E^{+1}$	1.95^{+2}	$48.75\ E^{+3}$
I-131	0.618	$0.618\ E^{-2}$	$0.618\ E^{-4}$	$0.618\ E^{-3}$	$15.45\ E^{-2}$
NG (Ci-MeV)	$1.29\ E^{+5}$	$1.29\ E^{+3}$	$1.29\ E^{+1}$	$1.29\ E^{+2}$	$32.25\ E^{+3}$
Particulates	1.38	$1.38\ E^{-2}$	$1.38\ E^{-4}$	$1.38\ E^{-3}$	$34.5\ E^{-2}$

Table 1: Regulatory and Target Release Rates

Out of Bruce A's nine stacks, the stack having the lowest activity concentration is shown in Table 2 which becomes the Minimum Detectable (MDA) to be measured.

STACK	RADIONUCLIDE	COMPLIANCE		CONTROL	
		Lower Limit	Upper Limit	Lower Limit	Upper Limit
Non Contaminated	H^3	0.418 $\mu Ci/m^3$	4.18 $\mu Ci/m^3$	4.182 $\mu Ci/m^3$	1.05 mCi/m^3
Stack	I-131	1.33 pCi/m^3	13.25 nCi/m^3	13.3 pCi/m^3	3.31 nCi/m^3
Reactor	NG	0.277 $\mu Ci/m^3$	2.77 mCi/m^3	2.766 $\mu Ci/m^3$	0.692 mCi/m^3
Building	Particulates (Sr-90)	2.96 pCi/m^3	29.59 nCi/m^3	29.6 pCi/m^3	7.40 nCi/m^3

Table 2: Target Minimum Detectable Activities

AREAS OF IMPROVEMENT

Stacks, fig. 3: The original regtangular ducts were replaced with circular stacks, the flow profile improved and the velocity probe was located to give an accurate average of stack velocity.

Relocation of Sampling Probe and Velocity Probe, fig. 4: The original sampling probe was some 8m above the roof and close to the edge. Any maintenance to be done required scaffolding and was hazardous to staff. The probes were lowered to be about 1.53m (5 ft) above the roof. This was about 3 stack diameters above the last confluence of ventilation streams. This was accepted since the flow profile was steady; representative sampling and averaged flow velocity were attainable, Ref. 2.

Leak Detection: Two mass flowmeters were added to the inlet and outlet of the sampling line to the Sorrento monitoring system; and usually read the same for no in-leakage or out-leakage. Useful for after cartridges are changed, and for pressurization/vacuum tests of the sampling system.

The Particulates and Iodine Compliance Monitoring Carts, fig. 5: Two mobile carts, each of which can be plugged into the sampling line as needed when the installed system was down for maintenance or during equipment failure, enhanced the reliability and availability requirements.

The Noble Gas Compliance Monitoring Carts, fig. 6: Each of the two carts has a High Purity Germanium (HPGe) detector; its liquid nitrogen cryostat, power supply, multi-channel analyzer, PC with screen display, printer, and software for data collection and analyses, isotope identification, quantification and trending. Its HPGe detector can easily fit the installed Marinelli chamber's second port. They are used when the installed noble gas system is unavailable, or when isotope identification is required.

Meeting Unavailability Requirements: A reliability of >99% as specified in Ref 3 is being met by the ongoing availability of the mobile carts figs. 5 & 6.

Interfacing with the Control Room: The installed monitoring system alarms in the Control Room on high release rates and on system malfunction. Emissions in %DEL rate/h are also displayed. This accelerating trend gives an operator advance warning of a potential problem before the alarm limit is reached.

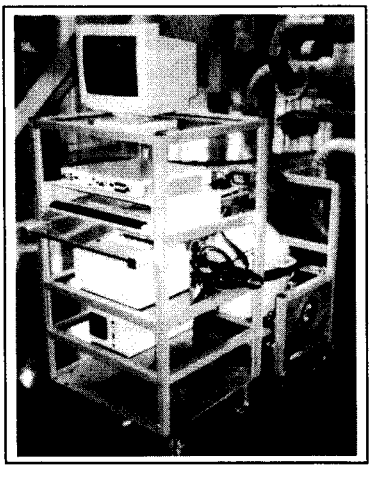

Fig. 6 The Noble Gas Compliance Monitoring Cart

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AN IMPROVED SAFETY SYSTEM OF A RADIATION FACILITY FOR INDUSTRIAL STERILIZATION

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Abstract - Modernization of the original safety system of the radiation facility for industrial sterilisation at the "Vinča" Institute of nuclear science is described. In order to improve radiation safety of the facility, the latest recommendations and requirements of IAEA have been implemented. The concept and design of the modernised system are presented. The new elements of the safety system are described and the improvements achieved by means of this modernisation are pointed out.
Key words: safety system, radiation facility, ^{60}Co.

1. INTRODUCTION

The gamma irradiator of the Institute of nuclear sciences "Vinča" performs the industrial process of sterilization of single use medical appliances, food, pharmaceutical, cosmetic or other products or raw materials. The sterilization is carried out by ionizing radiation from a ^{60}Co source. Currently a $9.25 \cdot 10^{15}$Bq (250,000 Ci) source is being used. The facility has been designed to accommodate safely the source of maximum activity of $3.7 \cdot 10^{16}$Bq (1,000,000 Ci). When the facility is not operating, the source is submerged into a water storage pool 6.5 m deep. The design of the facility and the principle of laying down the source classify it as the IV group of facilities for industrial sterilization [1].

In order to ensure a high level of radiation safety the facility is equipped with an autonomous safety system. In accidental situations, or in cases of failures of some of its elements, this system discontinues the irradiation process and returns the source to fully shielded position.

The original safety system has been ensuring a safe operation of the facility both for the personnel and for the environment since 1978. Having in mind this long period of intensive use of the facility, with the aim of increasing the radiation safety, and to upgrade the properties of the safety system to meet the latest requirements and recommendations of the IAEA [1,2], a modernization of the original safety system has been undertaken.

The accidental conditions causing the trip of the safety system could be due to either radiation or radiation-free events. The original safety system is activated by a power failure as a radiation-free event or by a fall out of the ^{60}Co source from its container or by inadvertent entry of a person through the product entry or exit ports as radiation events. The reaction of the safety system to any system malfunction or to a detection of accidental conditions is returning the source to the fully shielded position.

2. THE MODERNIZATION OF THE SAFETY SYSTEM

The original safety system consisted of: two ionization chamber measuring channels for the measurements of radiations in the irradiation chamber and in the maze, pressure mats at each product entry or exit ports of the maze, one emergency cable and emergency stop buttons located in the irradiation chamber, a safety delay timer allowing the irradiation chamber inspection in the course of preparations for start-up, and an electromagnetic lock on the door of the chamber.

The ionization chamber situated in the irradiation chamber serves to detect the position of the source. While the source is in the exposed position no person can have access to the irradiation chamber. The other ionization chamber situated in the maze serves to detect accidental carrying the source out of the irradiation chamber. A failure of either of the two channels will cause a safety system trip and return of the source to the fully shielded position.

Through the product entry and exit ports the products are transported to and out of the irradiation chamber by the product conveyor placed in aluminium containers called tote boxes. At both entry and exit ports pressure mats have been installed in order to register any inadvertent entry of personnel. Any excitation of these pressure mats will discontinue the irradiation process and the source will be returned to the fully shielded position.

In the course of preparations for the start-up the operator inspects the irradiation chamber during the time allowed by the safety delay timer and locks the door of this chamber by the key A. Once the safety system has been activated, this key is used to initiate the irradiation procedure. In case that during preparations someone was left behind in the irradiation chamber or in the maze, an emergency stop cable located around the walls of the irradiation chamber and of the maze and emergency stop buttons are available for stopping the process and returning the source to the fully shielded position. The key X is in possession of the manager of the facility and it is used for disabling parts of the safety system in the process of reconfiguration or replacement of the source. These processes are carried out following the written administrative procedure.

For the purpose of routine controls of radiation levels on the premises a portable radiation monitor is available

to the personnel. Despite the written procedures and requirements set for the use of this monitor, in the accidents that have occurred in similar installations throughout the world the portable monitor was out of order or had not been used at all [3].

The basic requirements set to the modernization of the safety system were related to the extension of the detection capabilities of both radiation and non-radiation causes of accidental states retaining, at the same time, the concept, operating modes, and all elements of the original safety system. In addition, a better presentation was required of the state of the system, of the causes of its operation, and the possibility of connecting this system to a cental computer was required.

The modernized system was extended to include detection of fire in any part of the installation and detection of earthquakes. The extension also covered the detection of a rupture of the encapsulation of the source ^{60}Co and the addition of a photoelectric system for controlling the product entry and exit ports. These extensions have been materialized by introducing into the safety system the following elements [4]:

- the electronic control system of the product entry and exit ports,
- detector of contamination of the storage pool water,
- detector of fire,
- seismic detector,
- the portable radiation monitor indirectly incorporated into the safety system,
- the central unit for displaying status and signalization.

All elements, original and new, of the safety system have been connected to a central unit for the purpose of presentation of the state of the system and for the corresponding signalization. This unit, as listed above, is another new element introduced into the structure of the safety system. Block diagram of the modernized safety system of the gamma irradiator is shown in Fig.1.

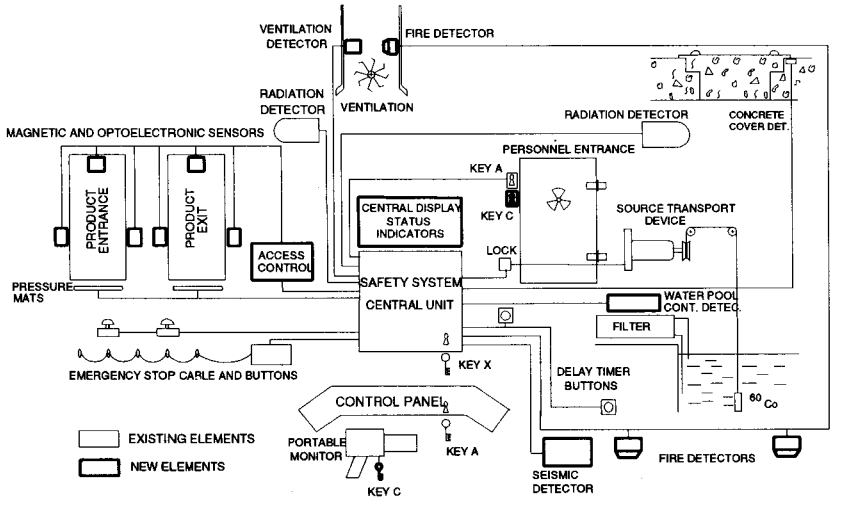

Fig. 1 Modernized safety system of the facility for industrial sterilization in the Institute "Vinca"

3. NEW ELEMENTS OF THE SAFETY SYSTEM

Some of the more interesting new elements of the safety system will be described in more detail. First of all is the central unit for the presentation of the state of the system and signalization. This unit provides the operator with a complete information of the state of the installation and clearly indicates the cause of any trip of the safety system. All data are presented on an LCD alphanumeric display containing 2x20 characters. The messages shown on this display are of the type:

NO ENTRY	AUTHORIZED PERSONS ONLY	SAFETY SYSTEM TRIP
IRRADIATION IN PROGRESS	SPECIAL REQUIREMENTS	MEASURING CHANNEL FAILURE

All possible states of the system have been included in the corresponding set of messages. Each change of the

display is accompanied by a short sound signal in order to draw operator's attention that a change has occurred.

The central unit is provided by an RS-232 interface thus it can be connected to a PC. Fig.2 shows the block diagram (a) and a simplified flow chart (b) of the central unit.

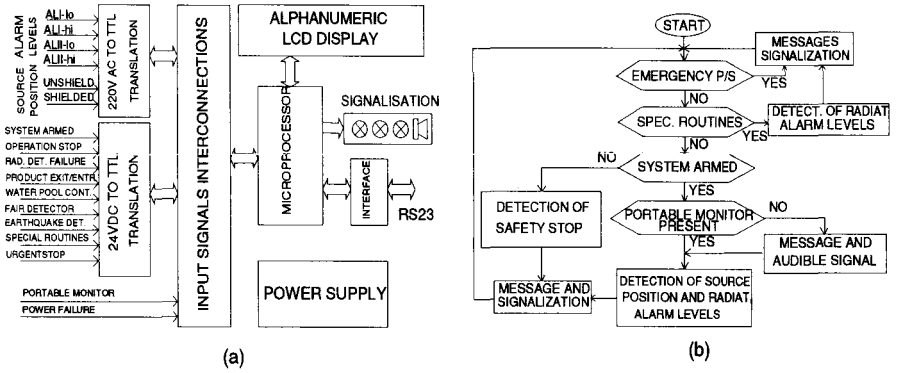

Fig. 2 shows the block diagram (a) and a simplified flow chart (b) of the central unit

The portable radiation monitor, has also been incorporated into the safety system. This has been done indirectly, i.e. the key of the electronic lock C is fixed to the monitor there fore the operator has to carry the monitor when entering or leaving the radiation chamber, thus has to meet the requirement that the chamber should not be entered without this monitor.

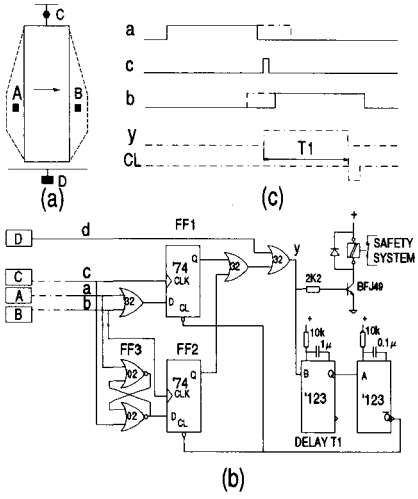

Fig. 3 Detectors position (a), block diagram (b) and signal timing (c) of the electronic control system at the product entry and exit ports

With the original safety system it was possible to enter deliberately the irradiation chamber by stepping over the pressure mats at the product entry and exit ports or by hanging on the conveyor belt. In order to prevent this type of entry into the high radiation area a new, additional, electronic control system has been designed, Fig.3.

The additional electronic protection system "recognizes" the tote boxes and permits only their entry to the maze. This system detects both direct entry or an entry of a person hung on the conveyor belt. The detection is performed by two pairs of photoelectric sensors (A and B) and by one inductive proximity switch (C), shown in Fig. 3. Micro switch D, shown in Fig.3a, registers any activation of the original pressure mat. The electric circuit controlling the product entry and exit ports in shown in Fig. 3b. The timing diagram of the corresponding signals is shown in Fig. 3c.

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ENVIRONMENTAL DOSIMETRY LOW DOSES

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The COGEBADGE® , COGEMA Dosimeter, equipped with a 4 dosimeter thermoluminescent card, is used for environmental dosimetry.

The capacities and limits of the thermoluminescent dosimetry system, allow, in routine operations, the measurement of very low dose levels, including natural radiation background.

For the photon thermoluminescence dosimetry system :

- the critical level (L_C), representing the signal that provides a confidence level 1-α that the result is not due solely to a fluctuation of the background, is equal to 4 μGy.

- the detection limit (L_D) representing the lowest value of dose which can be measured with a given confidence level (95 %) is equal to 10 μGy

- the determination limit (L_Q) representing the lowest value of dose which can be measured with a given precision (10 %) is equal to 120 μGy

The variation of the determination limit L_Q with precision is given in the figure 1.

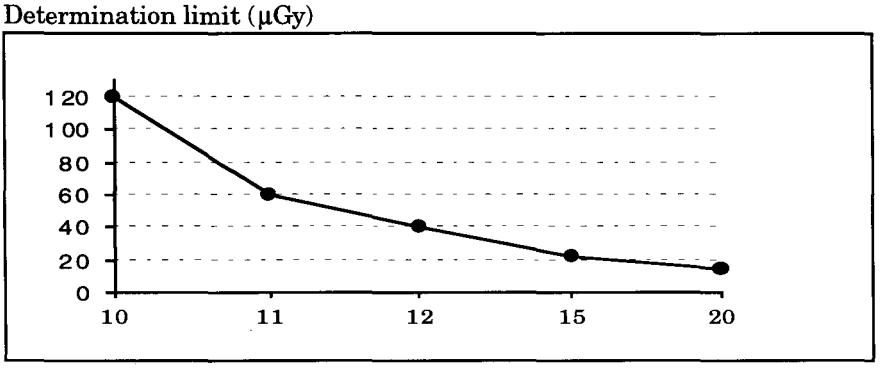

Fig. 1 Variation of the determination limit with precision

THE DEVELOPMENT OF AN EFFECTIVE SAFETY MANAGEMENT SYSTEM

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Abstract

In UK law the responsibility for making arrangements to minimise risk and secure safe operations rests with the employer who in practice delegates this down through line management.

The paper describes a novel approach to assisting managers in meeting their responsibilities and explains how the system operates in a company producing radioactive materials for use in healthcare, life science research and industry.

Introduction

The essence of UK safety law is that businesses must make arrangements to minimise risk and secure safe operations, and that individuals have a responsibility for ensuring that their actions do not endanger colleagues. The responsibility for making the arrangements rests with the Chief Executive and, by inference and common practice, is delegated down through line management.

In the past safe operational practice was achieved within the company through instructions devised by the central safety group and issued to line managers. The safety group had the responsibility to ensure that the company complied with all relevant laws and licenses while the majority of line managers assumed or expected that 'safety' was taken care of centrally and thus had little proactive influence on the safety of their operations. There was also little in the way of formal audit.

Company Safety Arrangements

In 1990, the company safety policy was revised to emphasise the safety responsibilities of line management and included the phrase "managers at all levels are responsible for the safety of operations under their control". The central safety group became responsible for providing a safety management system to enable line managers to demonstrate that they were undertaking the required actions.

The key regulatory requirements and company standards are given to managers through the Company Safety Arrangements. This document tells managers what arrangements they must make to comply with the law fully, without the need for the managers to be familiar with the detail of the legislation. In the UK the regulatory framework is complex with over 30 Acts of Parliament and 80 sets of Regulations applying in the safety field.

The Company Safety Arrangements comprise 25 single pages, each covering a specific topic (for example : safety instructions; safety training; supervision; assessment and control of hazards; audits and corrective action). Each arrangement identifies who is responsible for the action and what the minimum requirements are. Generally the line manager or the site manager (the senior individual responsible for the operations of a complete site) are the focus of the actions. The arrangements are supported by more detailed safety guidance notes which can be referred to by relevant managers when they are determining their actions.

The responsibility for interpreting new and existing legislation for all aspects of safety and for ensuring that it is covered by the requirements laid down in the Company Safety Arrangements lies firmly with the central safety group. Managers can therefore expect that if they comply with the company arrangements that they will be in compliance with the relevant laws.

Manager's Safety Files

The execution of the company safety arrangements requires each manger to issue to staff rules, instructions and procedures appropriate to the nature of the operations under his or her control. The proper recording and review of these provides adequate evidence of compliance with the arrangements (and hence compliance with the laws). Managers were encouraged to collect the documents together or keep references to them in a "safety file" which was to be regarded as a "living system". It represents a simple documented management system. In order to ensure an acceptable and reasonably uniform content in an auditable format the central safety group imposed a quality assurance dimension onto the system. The system must comply with the QA requirements of ISO 9000 where these are relevant to the operations. Essentially, the manager must establish a safety organisation, identify and control all documents, conduct regular reviews and internal audits and have an effective corrective action procedure. Some of the details of ISO 9000 such as the requirement for a manual and, particularly, a management representative, were an inappropriate fit with "arrangements" and with the paramount need for the manger to "own" the safety file. The least prescriptive standard offering safe performance must be the preferred option. However, recent experience has shown that when a management unit has fully embraced ISO 9001, the existing safety file components which are common to both the safety and quality assurance systems also are in full compliance with the quality assurance system.

The basic components of the manager's safety file are shown below :

Manager's Safety File

- Policy
- Organisation
- Appointments
- Instructions
- Training
- Hazard Assessment
- Accident Investigation
- Audit
- Review

Audit

A key area where the current practice differs from the past is in auditing. The present arrangements require three tiers of audit. The first is 'self audit' carried out by the managers to enable them to know where they do or do not comply with laws or company standards and to satisfy themselves that their instructions and systems are being followed by their staff. The second tier is at the site manager level and involves reviewing the manager's self audits and checking the quality of the manager's audits. The third tier is at the corporate level and combines the aim of the site manager's audits with in-depth technical audits carried out in reaction to audit observations, legislative changes etc. In practice the site manager's and corporate audits are either combined or alternated.

Audits have been carried out over the past five years by formally trained auditors to ensure that managers have evidence to demonstrate compliance with each and every detail required by the company safety arrangements, using the arrangements themselves as an audit checklist. The adequacy of implementation has been checked by sampling procedures and questioning staff in the workplace. It is accepted that the likelihood of the auditor uncovering an isolated incidence of bad practice is small, because most operating units will have more than 100 practical procedures each and behaviour of staff is influenced by the presence of the auditor. It is this detail of the operating procedures that must be checked locally (and systematically) by the local audit arrangements.

The Arrangements in Practice

The system was implemented by all technical managers on Amersham's UK sites and effective management was generally achieved within two years. An important benefit was the recognition by managers that the safety file, when completed, offered a useful source of information and enabled them to understand their safety responsibilities and liabilities. There were two main problem areas. The requirement for internal audit was largely ignored and a few managers tried to abdicate their safety responsibilities to a safety specialist. Suitable counselling has now ensured that all direct appointments and instructions are those of the manager, and that a management signature indicates absolute ownership of any consequences arising from the signed document.

The manager's safety file includes as a minimum a statement of policy, organisation chart, key appointments, and references to all required rules, procedures and records. A series of unsupported compliance statements is insufficient unless there is a clear indication of how compliance is achieved. The depth and coverage of the information is tailored to the complexity of the organisation and tasks involved. When a radical change in organisation has occurred, the file has been found to be inadequate and significant changes, for example to supervisory procedures, have been necessary. On the other hand, when management has changed but the organisation remained constant, the documentation has required little alteration. There is, however, a need to train a new manager formally in the whole safety management system.

Conclusions

Experience over five years indicates that the safety management system coupled with audit and review can provide managers with confidence that their operations are adequately controlled and regulators with an auditable trail of evidence to confirm this.

AN INTEGRATED APPROACH TO PLANT SAFETY

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Abstract
An innovative safety approach to the design of plant used in the manufacture of a wide range of radioactive products for healthcare, life science research and industrial applications at Amersham International is described.

The paper also describes how the techniques for reviewing the design of new plant have been adapted to the review of operational facilities and the decontamination and decommissioning of facilities at the end of their production life.

Introduction
Over the last five years safety review arrangements have been developed consistent with those applied in the nuclear industry but adapted in an innovative way to reflect the requirements of a commercial manufacturing environment. These arrangements have been applied to the design of new facilities; reviews of operating facilities; and the decontamination and decommissioning of redundant plant.

Classification and Project Phases
A multistage procedure for safety approval is applied to all new facility projects. At the start, the project is classified by a senior health physicist into one of four categories, depending on its safety significance. Higher significance construction projects are split into 4 stages - Concept, Final Design, Pre-Operational Commissioning and Operational Commissioning.

Hazard Identification/Hazard and Operability Study
A Hazard Identification review is carried out at the Concept stage by a small team using a check list. This early review ensures that the designer is aware of the main safety issues and can take account of them in the facility design.

The developed plant design and process are subjected to a Hazard and Operability Study (HAZOP) to identify all possible deviations from the way the design is expected to work and all the issues associated with these deviations. The group performing the study are selected from the Safety, Manufacturing and Engineering functions. The Manufacturing representative is a supervisor or operator with experience of the type of plant and process being designed.

The technique uses a structured approach for the identification of hazards by the generation of deviations from design intentions using guide words. Many of Amersham's operations are 'hands on' and the guide words may be applied to operator actions as well as plant components.

The HAZOP sessions have become an integral part of the design process and, although when initially introduced were viewed with some suspicion, they are now welcomed as a thorough vetting of the design before potential errors and omissions prove more costly. Having carried out the HAZOP it is possible to quantify the risk of the plant/process to operating staff or the general public by using techniques such as Fault Tree Analysis. The HAZOP is as much a review of quality as it is of safety issues and the technique has been successfully applied to manufacturing operations where the quality of the product is the principal objective.

Peer Review
The HAZOP provides an excellent basis for writing the 'Final Design' Safety Clearance document. This is then subjected to a review by a Peer Group which brings together the design team and an independent group of experienced staff selected from Site and Corporate Safety, Engineering and Manufacturing. The proposals are presented by the design team and subjected to challenge by the Peer Group. All points of view can be balanced at this session and it has proved to be much more effective than using individuals to review a document in isolation. Following this review the 'Final

Design' Safety Clearance is formally approved by the relevant Manufacturing Manager, the Site Engineering Manager and the Site Radiological Safety Officer. Peer Reviews may be held at other stages of the project if deemed necessary by the Peer Review Group Chairman.

The Arrangements in Practice

Amersham is currently investing in new pharmaceutical manufacturing facilities. Two semi-automated production suites situated in a clean-room complex carry out the final stages of radiopharmaceutical manufacturing. They will handle stock dilution, vial filling, sealing and labelling operations and waste management. Nine radionuclides, including Sr-89, Ga-67 and In-111, will pass through the plant which has been constructed and is currently undergoing pre-operational commissioning tests. This was a particularly challenging project to marry the stringent requirements of pharmaceutical regulators where Good Manufacturing Practice guidelines must be observed and radiological protection requirements for protection of operators from radiation exposure. The principal pharmaceutical and radiological design issues and identified solutions are shown in the Table below:

Pharmaceutical design requirement	Radiological design requirement	Solution
Operate pharmaceutical clean rooms at slight overpressure to protect product from particulate	Handle radioactive products in rooms at slight underpressure to prevent escape of product following a spill	Clean room surrounding radioactive handling area at positive pressure. Radioactive material handling room/ enclosures at negative pressure
Deliver air to product to pharmaceutical standards and ensure flow is away from product	Direct air away from operators and towards interior of enclosures containing the radioactive product	Pharmaceutical requirements at point of vial filling and stock bottle emptying. Enclosures at negative pressure to protect operator
Provide minimum disturbance to laminar air flow close to vial filling	Surround the vials with lead shielding as they are filled	Provide some shielding local to the vials but supplement with full shielding on face of enclosure
Label vials after dispensing with product to prevent errors	Label vials before dispensing to reduce operator dose	Label vials after dispensing but automate and provide shielding local to vial location
Provide easy accessibility to dispensing line for full line clearance at end of product run	Provide shielding and remote operation to control operator dose	Bulk stock separated from dispensing line and heavily shielded. Filled vials lightly shielded and shielding designed for easy removal at end of run
Remove radioactive waste after each product run	Store radioactive waste in facility until decayed	Highest activity liquid waste stored. Bulky solid waste removed by remote transfer

Periodic Reviews of Safety

Formal major reviews of safety are carried out for all facilities on a 5 year rolling programme. The plant, process and safety management arrangements are considered in each review which covers normal operating conditions and also reasonably foreseeable fault conditions arising from plant failures, operating failures and external hazards. The HAZOP technique is used as a tool for fault identification.

A report with recommendations for improvements is submitted to a Peer Review in the same way as a new plant safety clearance. The Manager is then required to prepare an Action Plan for improvements which is monitored through to completion.

Examples of areas where improvements have been required are: radiation shielding; fire detection; fire compartmentation; transfers of radioactive material; containment; extract filtration/traps; safety management; barrier monitoring arrangements; activity in air alarm monitoring.

Decontamination and Decommissioning Projects

The main distinction between new plant projects and decontamination and decommissioning projects is the amount of 'on the ground' operational safety issues to be resolved. A round-the-table HAZOP is first carried out on the proposed approach, in order to identify the main protective features required to control contamination and radiation dose and to test the operability of the method. The HAZOP may also identify the need for a 'mock up' and trial operation to be carried out prior to the actual job to confirm its practicability and this enables operating times to be measured where dose control is critical. Detailed Method Statements are produced by the staff who are going to undertake the work. These Method Statements are subjected to a HAZOP approach in the workplace in order to identify potential safety problems before the task is undertaken. A system for ensuring that the significance of deviations from Method Statements are assessed has been developed. The whole process is overseen by the operational Health Physicist.

Conclusion

The integrated approach to plant safety responds effectively to rapidly changing manufacturing requirements and has resulted in real improvements to 'on the ground' operational safety without imposing unacceptable bureaucratic restrictions.

UK IONISING RADIATIONS INCIDENT DATABASE (IRID)

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ABSTRACT

The widespread use of ionising radiations carries with it the potential for incidents and accidents. Their severity can vary from the trivial to the fatal and may involve substantial economic penalties. In order to assist in learning the lessons from events that have occurred, the NRPB, HSE and Her Majesty's Inspectorate of Pollution (HMIP) have established a national Ionising Radiations Incident Database (IRID). This paper provides details of the specification and operation of the database together with the proposed feedback mechanisms.

INTRODUCTION

A fundamental part of any assessment of, and plans for the prevention and mitigation of reasonably foreseeable radiological incidents, is to learn the lessons from accidents that have occurred and to feed this back through training and guidance to users. The efficiency with which this happens depends on the collective experience of individuals. Each may have direct experience of some types of accident, but will almost certainly have to rely on data from publications, conferences and possibly anecdotal information. There is a natural tendency for organisations not to publicise their own accidents, especially if the accident was a 'near miss' that fortuitously did not have serious consequences - this time! As a result our feedback experience from accidents both nationally and internationally can be limited, fragmented and difficult to pass on to the next generation of workers. Thus there is a need for a more coherent means of feedback.

DEVELOPMENT AND OBJECTIVES

UK legislation requires that accidents, incidents and occurrences which satisfy specified criteria are reported to the regulators, HSE and/or HMIP. The regulations also require an employer working with ionising radiations to appoint a Radiation Protection Adviser (RPA) and to involve that person in investigations of incidents and accidents. NRPB acts as RPA to some 800 organisations and provides technical services to many others. As a result, NRPB is involved in the investigation of a wide range of incidents and accidents, including many 'near misses' which would not be legally reportable. An NRPB/HSE feasibility study (1990) on improving feedback concluded that mechanisms were in place in the nuclear and transport sectors, but that there would be significant benefits from a national incident database to cover the non-nuclear sector ie. industry, research, teaching and health care. However it also highlighted some crucial practical problems related to the confidentiality of information; the resolution of which took some time. Impetus to the work was provided by the Commission of the European Communities (CEC), both in supporting work in this area and as a result of their 4th European Seminar on Optimisation of Radiation Protection in April 1993 (1) which concluded that a principal challenge for the future is ".... to improve our feedback from past experiences by developing databases".

As a result of subsequent work the NRPB, HSE and HMIP have entered into a partnership to combine incident and accident data in a confidential form in the Ionising Radiations Incident Database (IRID). The objectives are:

(a) to establish and operate a database that will act as a national focus on radiation incidents primarily in the non-nuclear sector, ie. industry, research, teaching and medicine;

(b) through appropriate publications, to provide feedback and guidance to users of ionising radiations on preventing or limiting the consequences of radiation incidents; and

(c) to provide regulatory bodies and others with advisory responsibilities, analyses of relevant data that help in assessing priorities in resource allocation.

SCOPE

The database is designed to cover radiological incidents including near misses and occurrences involving actual or potential occupational and public exposure. However, some types of incident are excluded from IRID since they are already covered by existing arrangements. Those excluded are:

(a) **Nuclear incidents:** these would include nuclear (as opposed to radiological) incidents that involve criticality or events related to the safety of reactors or nuclear processing plants.

(b) **Transport incidents:** the NRPB runs, under contract to HSE and the Department of Transport (DTp), a database (2) of incidents involving the transport of radioactive material.

(c) **Patient exposure:** The HSE (3) and Medical Devices Agency have systems for reporting and dissemination of relevant data. In addition at a European level the European Federation of Organisations for Medical Physics (EFOMP) are in the process of establishing a database for radiotherapy accidents in Europe (4).

Included are:

(a) **Radiological incidents occurring on nuclear sites:** eg. an incident involving a radiography source being used on a nuclear site.

(b) **Occupational and public exposure in the medical sector:** eg. a radiotherapy source left implanted in a patient discharged from hospital.

(c) **Incidents involving radioactive material in the public domain not arising from discharges authorised by regulatory bodies:** these would include incidents which required the National Arrangement for Incidents involving Radioactivity (NAIR) to be invoked.

(d) **Radiological incidents in the defence sector:** the unattributable nature of the database may now allow such incidents to be reported, without the origins being obvious.

(e) **Unintentional and/or unauthorised discharges of radioactive materials into the environment:** these would include discharges from nuclear and non-nuclear licenced sites.

NRPB are responsible for the management of the database and together with HSE and HMIP are reviewing their files to provide historical data. This will be generally limited to the last decade, although some earlier incidents will be included because of their value in 'learning the lesson'. It is recognised that this data is incomplete and that any retrospective analyses would need to be used with caution.

DATABASE STRUCTURE

The database operates on a Personal Computer (PC) using DBASE V software and consists of 24 fields including one text field. These are defined in a Data Element Dictionary and are summarised below:

1	Case number	13	Occupation of worker(s)
2	Area	14	Type of equipment
3	Incident Date	15	Isotope(s) involved
4	Incident Level	16	Activity
5	Exposure Level	17	Kilovoltage of radiation generator
6	Site Level	18	Cause of incident
7	Nature of accident	19	Availability and effectiveness of contingency plans
8	Number exposed: occupational	20	RPA: appointed, involved in investigation
9	Number exposed: public	21	RPS: appointed, involved in investigation
10	Whole body dose(s)	22	Follow up action (eg improvements)
11	Extremity dose(s)	23	Date of entry in database
12	Internal organ dose(s)	24	Description: text field

Fields 1 to 23 contain either numeric data (eg. dose in mSv) or one or more codes that categorise the incident and have been designed to permit useful analyses of the database; for example searching for incidents over a specified period of time involving industrial radiography and resulting in doses in excess of the dose limit. In defining the fields, a balance has had to be made between the level of detail, its usefulness and the effort involved by those submitting incident data. To maximise the overall benefit it was felt important to achieve a high reporting percentage and that this would only be achieved if the reporting system was not

onerous on the contributors. Therefore the number of data fields has been minimised to that consistent with categorisation and analysis requirements. Field 24 is a text field that contains descriptions of the incidents, the causes, the consequences, follow up actions and lessons to be learned. The format has been designed to allow it to be directly reproduced in publications to provide feedback to the users.

CONFIDENTIALITY AND QUALITY ASSURANCE

The information entered into the database will not contain any names or addresses of organisations or individuals. Only the originator of the incident entry will know the names of the organisations or individuals concerned and all data presented to NRPB will be in a sanitised format. NRPB has undertaken not to divulge any privileged information to a third party. HSE and HMIP are well aware of the natural wariness that potential contributors may have in respect of the involvement of regulatory bodies. Therefore they have given assurances that they will not seek to obtain further information from the other partners (or the contributing organisation, if different) about any incident recorded on the database which was not reported to regulators.

Simple reporting proformas and guidance has been developed to help obtain consistency in the reporting of incidents. To try and minimise the degree of variation in the categorisation of incidents the Database Co-ordinator at NRPB will independently code the data (from the text description) and discuss any discrepancies with the contributor prior to data entry and entry validation. Checks have been devised to identify duplicate reports by automatically cross referencing a number of fields.

FEEDBACK

Undoubtedly the value of IRID will be judged on the quality and accessibility of feedback to users and regulators. Equally this will be a crucial factor in maintaining the flow of contributions from various organisations. Initially the primary feedback route will be through publications:

(a) NRPB/HSE/HMIP will publish periodic review documents that reproduce the text descriptions of incidents, where appropriate enhanced by diagrams or drawings of accident scenes, and analyses of the distribution and trends in incidents.

(b) Incidents that are of particular relevance will be published in NRPB's 'Radiological Protection Bulletin' HSE's 'RPA Newsletter' and HMIP's Bulletin.

A key use of the feedback from IRID should be in the training of all those involved in the use of ionising radiation. Therefore a number of short joint publications (with slides) that can be used as training material targeted at specific work groups will be produced.

FUTURE DEVELOPMENTS

Although initiated by NRPB and HSE and joined by HMIP, it is intended that IRID is made as wide a collaborative effort as possible, involving others in the partnership or as contributors.

There are also valuable lessons to be learned from incidents outside the UK and it is envisaged that one future development will be to add a module covering such incidents. This may be influenced by the development of comparable databases in other countries and international organisations such as IAEA and the European Union (EU). Undoubtedly each country or international organisation will have a need for specific data fields but it should be possible to agree on core data fields and formats - a challenge for the future.

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In a Tritium facility the measurement of Tritium surface contamination is required to insure :

- The radiation protection of workers at different work places,
- the sorting, disposal and storage of wastes and materials.

The most common way to measure surface contamination is to quantify the removable surface contamination by liquid scintillation counting. However this process is unaccurate since the removal efficiency has not been quantified so far. There are very few scientific reports about this issue. Some studies related to stainless steel has been carried out, nevertheless data about organic compounds are still missing.

Our experiments were carried out using the smear technique on four types of specific materials belonging to Valduc Tritium utilities :

- Stainless steel,
- waste containers (steel + painter),
- vinyl,
- floor coating.

Results are gathered on the paper. They highlight we must introduce in the relationship used to evaluate the surface activity a coefficient which depends on the type of material and on the surface state. On another way we can observe a good agreement between our values and those from the theoritical model proposed for stainless steel (1).

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ASSESSMENT OF THE EFFICIENCY OF PERSONAL PROTECTIVE EQUIPMENT FOR USE IN RADIOACTIVE CONTAMINATED ENVIRONMENT - RECENT IPSN CONTRIBUTION

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Abstract

In all industrialized countries important studies have been accomplished with the aim of improving the personal protective equipment (PPE) performances, i. e. of minimizing the exposure to hazardous environment and other risks for workers attending to operate in a contamined area and, at the same time, to increase their physiological comfort.

In the same field, IPSN, in collaboration with other European laboratories, has achieved in the last four years, an important research program, in order :

- to determine the efficiency of protection of the different categories of PPE,
- to improve the design and to develop new material of construction of special type of PPE (e.g. ventilated-pressurized protective clothing, ventilated hoods,...),
- to optimize the time of use during routine operation or emergency situations, by considering both technical performances and physiological aspects such as heat constraints and health effects.

After a brief review of the main characteristics required for the different categories of PPE, the present communication will present the directions of research which were preferred in order to achieve the objectives specified here above. Then it will conclude with a review of the instructions of use, maintenance and wearer training, which are the essential factors to be considered by every responsible of operational health of the employers in a nuclear facility, in relationship with the PPE management program.

1 - PERFORMANCE LEVELS OF THE DIFFERENT CATEGORIES OF PPE

In order to evaluate the level of protection offered by a particular PPE, the following definitions are used :

- **Nominal Protection Factor (NPF)**

The ratio of the average concentrations of contaminant measured in the ambient atmosphere and inside the PPE, at the point where the wearer draws breath. The concentrations taken into account are the average concentrations recorded during a standardized test.

$$NPF = \frac{C\ ambient}{C\ inhaled}$$

- **Permeance or Total Inward Leakage (TIL)**

Quantity corresponding to the reciprocal value of the NPF. It is expressed in %.

$$TIL = \frac{1}{NPF}$$

According to these definitions and as usually admitted in the literature [1], [2], the efficiency of the different categories of PPE can be classified in accordance with table 1.

Categories of PPE / Definitions	Range of Nominal Protection Factor	Range of average leakage into the PPE (in %) during standardized test (Permeance)
Ventilated-pressurized protective clothing (Air feed impermeable suit)	10 000 - 100 000	0.001 - 0.01
Ventilated hood (Air feed impermeable hood)	5 000 - 50 000	0.002 - 0.02
Filter type respirator (Full face mask with filter)	2 000 - 10 000	0.005 - 0.01
Non ventilated protective clothing (Blouse or impermeable suit)	2 - 20	5 - 50

Table 1 : Efficiency of different categories of PPE

2 - MAIN RESEARCH DIRECTIONS

2.1 - Ventilated-pressurized protective clothing or hoods

- Improvement of the protection factor

For air feed protective clothing the protection factor level mainly depends on :

- the **static leaktightness** which in term depends on the intrinsic tightness of the clothing during normal working conditions; the studies have been focused on the improvement of the welding or seams and fasteners quality.
- the **dynamic leaktightness**, which is essentially dependant on the efficiency of the exhaust device(s) and on the aeraulic performances of the protective equipment.

Concerning this last field, both investigations have permit to increase namely the NPFs by a factor of 4 (i.e. from 20 000 to 80 000) by developing special high efficiency exhaust devices, and by increasing the internal air flow rate inside the equipment (see figures 1 and 2).

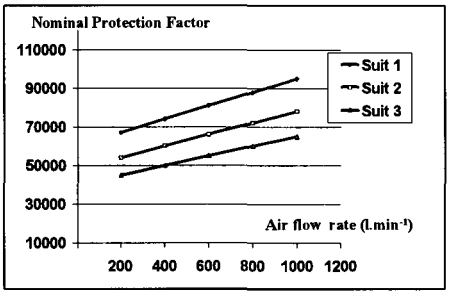

Figure 1 : Variation of NPFs by increasing air flow rate in ventilated-pressurized clothing

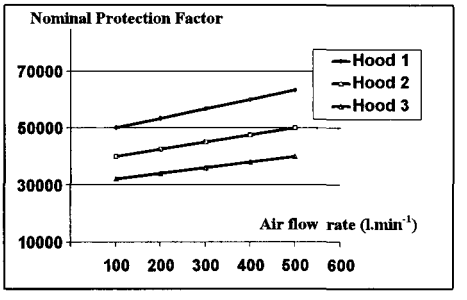

Figure 2 : Variation of NPFs by increasing air flow rate in ventilated-pressurized hoods.

- **Improvement of physiological comfort (new concept of ventilation of suits)**

The studies were carried out in cooperation with occupational physicians to the definition of a system of "direct skin or body ventilation" which allows a better thermal regulation of the wearers, particularly in case of intervention in hot environments.

The results have proved that for the same ventilation rate and the same physical work rate, the acceptability duration may be improved by 50 % by means of the refreshing due to the direct skin ventilation, in comparison with the traditional clothing in which the ventilation is fed into the suit, i.e. only over the underwear.

- **Studies concerning new material of construction of PPE**

In addition with efficiency against particle penetration, design of PPE must comply with other typical requirements, such as mechanical resistance, flexibility, flammability behaviour, suitability to disposal (by incineration), gas permeation, etc. Table 2 here after presents the compared performance of principal materials, replacing PVC, which has been so far the most common one.

PVC (standard or non flammable)	Low cost Easy welding Non flammable	Non incinerable High Cl content (36 %)
POLYURETHANE (standard or non flammable)	Incinerable Good mechanical & chemical resistance	High cost Difficult to weld
POLYETHYLENE (standard or non flammable)	Low cost No Cl content Incinerable	Flammable Mechanical resistance lower than PVC
PVC coated with polyester	Low permeability to gas (e.g. tritium), Excellent mechanical & chemical resistance	High cost Heavy material Non flammable
Polyethylene - Vinyl Acetate (EVA)	No Cl content Incinerable Acceptable mechanical & chemical resistance	Flammable, Difficult to weld and assembly, Higher cost than PVC
Non woven tissue (TYVEK) (polyethylene standard or non flammable)	Low weight, Incinerable Flammability depends on (Cl, Br, ...) content	Higher cost than PVC Mechanical & chemical resistance lower than for the previous material

Table 2 : Comparison between different material of construction of PPE

Airborne concentration	Recommended personal protective equipment	
	Continuous use	Short duration use in special circumstances
Less than 0.3 DAC	No requirements. Half face respiratory may be appropriate	No requirements. Half face respiratory may be appropriate
Greater than 0.3 DAC Less than 30 DAC	Full face respirator with particulate filter	Full face respirator with particulate filter
Greater than 30 DAC Less than 150 DAC	Air fed respirator, hood, blouse or impermeable suit	Full face respirator with filter, or air fed respirator, or impermeable suit
Greater than 150 DAC Less than 300 DAC	Air fed impermeable suit	Air fed respirator or impermeable suit
Greater than 300 DAC	Air fed impermeable suit	Air fed impermeable suit

Table 3 : Guidance for the selection of PPE (particulate hazard) for normal operations or emergency situations (IAEA Recommendations)

2.2 - Non ventilated-non pressurized protective clothing

Due to the fact that this kind of protective clothing does not provide the entire protection of the respiratory tract of the wearer, it was interesting to assess the different kinds of equipment in use (blouses, coveralls, 2 pieces suits,...) in order to determine the protection factor given by the protective suit without any additional respiratory protective device.

Two kinds of particle permeation test, using Na Cl agent test, with different particle sizes (respectively 0,6 µm and 1,4 µm of aerodynamic mass median diameter), have been successively performed :

- at first, the efficiency of the material of construction has been implemented. Figure 3 here after gives the results obtained for several types of non wowed materials.
- at second, the efficiency of the complete suit has been implemented (this test is called "whole suit test"). Figure 4 here after shows the results obtained on different types of protective clothing.

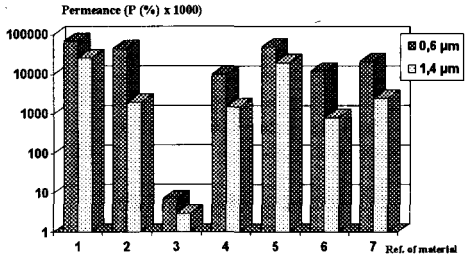

Figure 3 : Particle penetration test for material of construction of protective suits

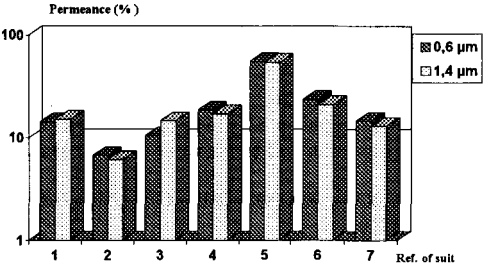

Figure 4 : Particle penetration test for protective clothing ("Whole suit test")

The conclusion of these tests are :

- Figure 1 : results depend on both particle sizes and permeability of material,
- Figure 2 : results depend only on the design of the suit (position of openings, straps on legs, neck, arms,...) and very little on material of construction. Total Inward Leakage can vary from 5 to 50 %.

These results have been confirmed by several European laboratories using the same test procedures, during intercomparison exercises [3], [4]. This means that the use of non ventilated protective clothing must be limited for very low contamined environment or combined with appropriate respiratory protective equipment in case of high or medium contamined environment for routine or emergency operations.

3 - CONCLUSION

Due to the different levels of efficiency, PPE should be chosen in accordance with the risk analysis. Table 3 here above could be a good help for the selection of the appropriate PPE. In addition other parameters should be taken into account, such as :

- examination of the means for removal or reduction of the sources of internal/external other than individual contamination or exposure (constructive provisions, confinement, ventilation or preferential extractions,...);
- evaluation of environmental parameters (toxic gases presence, fire risk, temperature, etc.) and of human factors (training and experience of operators);
- preparation of working protected areas (mobile or fixed tents with appropriate ventilation joined to equipment for radiological surveillance);
- choice of the most appropriate PPE for the intervention : reusable or disposable (disposal problems procedures);
- economics factors : equipment cost + decontamination cost/disposal cost (related to wastes volume);
- personal training, control and maintenance of equipment, etc.

4 - REFERENCES

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[4] To be published jointly by IPSN/BIA.

Proffered Paper Poster Session No.: PS- 5
Poster Report No.: P53-8

ACCIDENTAL INTERNAL EXPOSURE OF ALL GROUPS OF CHERNOBYL NUCLEAR POWER PLANT EMPLOYEES

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INTRODUCTION

Accidental internal exposure of Chernobyl NPP employees has started from April, 1986 and it was found to be decreased to pre-accident level at the end of 1987. Significant number of people from all groups of staff and temporary employees were measured using whole body counters situated in Clinical Department of the Institute of Biophysics, which has represented the main body for medical assistance and expertise in these people including those, who suffered from acute radiation syndrome as well as the people engaged in all kinds of works at Chernobyl NPP site. Technical characteristics of the equipment and techniques used to assess the internal exposure are given in [1].

INTERNAL EXPOSURE OF PEOPLE DIED FROM COMBINED RADIATION INJURIES

Internal exposure doses did not exceed 12 Gy (thyroid gland), 1 Gy (lungs) and 0.25 Gy (whole body) for all mortal cases excluding two patients (Reg. Nos.: 24, 25). Internal exposure doses for these two cases are given in Table 1.

Table 1 Internal exposure of staff members died due to the Chernobyl accident (maximum levels)

Reg. Number	Internal exposure dose, Gy		
	whole body	lungs	thyroid gland
24	1.8	2.2	23
25	0.8	1.1	21

INTERNAL EXPOSURE OF STAFF WORKED AT THE SITE OF CHERNOBYL NPP DURING THE EXPLOSION AND FIRE AT UNIT 4

Table 2 demonstrates thyroid dose distribution in 204 persons including 143 acute radiation syndrome victims. Maximum thyroid doses are about 20 Gy, average thyroid dose is (1.5±0.2) Gy, 95% percentile is 1.8 Gy.

Table 2 Thyroid dose of staff worked at the site of Chernobyl NPP during the explosion and fire at Unit 4

Dose range, Gy	Number of people
0 - 1.23	173
1.23 - 3.67	18
3.67 - 6.11	4
6.11 - 8.55	4
8.55 - 10.99	2
10.99 - 13.43	2
13.43 - 15.87	0
15.87 - 18.31	1

According to whole body counter measurements the lung doses for these people did not exceed 1.2 Gy, where the average lung dose was found to be (0.15±0.07) Gy and 95% percentile is 0.7 Gy. Whole body internal doses did not exceed 0.1 Gy.
Radionuclides provided internal exposures are: Zr-95, Nb-95, Ce-141,-144, Cs-134,-137, I-131.

INTERNAL EXPOSURE OF PEOPLE EMPLOYED FOR ACCIDENT CONSEQUENCES ELIMINATION IN 1986

The described contingent includes persons employed from May to December, 1986 within 11 consequent periods. Table 3 gives the characteristics of these groups.

Table 3 Employee groups observed in 1986 ã.

Group Number→	1	2	3	4
Period of work→	26.04-01.05	01.05-10.05	10.05-20.05	20.05-1.06
Number of people→	178	31	20	25

Group Number→	5	6	7	8	9	10	11
Period of work→ (months of 1986)	VI	VII	VIII	IX	X	XI	XII
Number of people→	56	52	50	48	41	23	18

Total number of observed people is 542.

Table 4 Average values of radionuclide intake (nCi) in Chernobyl employees (1986)

Radionuclide	Number of group										
	1	2	3	4	5	6	7	8	9	10	11
Cs-134	195	52	29	36	35	36	30	20	21	22	14
Cs-137	410	96	58	73	67	78	67	44	48	51	37
Ru-103	218	N/M	102*	N/M	N/M	N/M	N/M	N/M	N/M	N/M	N/M
Zr-95 + Nb-95	784	N/M	22*	N/M	N/M	N/M	N/M	N/M	N/M	N/M	N/M

*Average for groups 1,2,3 (i.e. in May, 1986). N/M - not measured (below detection limit).
Table 4 shows that inhalation of radionuclides was practically finished in the second half of May, 1986, which is certified by the absence of "inhalation radionuclides" (Zr-95, Nb-95, Ru-103). Total incorporated activity was stabilized on the level of 50 nCi, which is typical for local population burdens found after the Chernobyl accident in urban population of Kiev and Gomel and corresponded to the foodstuff radioactive intake.

Table 5 gives the comparison between NPP staff and temporary employed workforce ("liquidators").

Table 5 Comparison of the internal contamination between NPP staff and "liquidators" (April, 1986), nCi

Radionuclide	Staff	"Liquidators"	Total Average
Cs-134	229	96	195
Cs-137	505	229	410
Ru-103	279	198	218

We did not calculate individual doses for 542 persons, which majority had internal doses below 0.0015 Gy. Maximum observed internal doses were equal to 1.16 Gy (lungs) and 0.03 Gy (whole body). Average individual whole body internal exposure dose of Chernobyl NPP employees did not exceed 0.003 Gy in 1986. Persons with lung doses above 0.05 Gy represents 2.4% of all measured employees and lung dose groups correspondingly are of the following percentage: 1.7% (>0.15 Gy), 0.9% (>0.75 Gy), 0.2% (> 1 Gy). All these lung doses are completely accumulated within 1986.

INTERNAL EXPOSURE OF PERSONS EMPLOYED IN 1987

Total number of Chernobyl NPP employees measured by our whole body counter in 1987 is 1192. 10 occupational groups and one group of people whose occupation was not exactly identified were selected. 56% of measured people represents "pure" nuclear power plant staff and about 44% are temporary workforce.

Cs-134,-137 were found in all measured persons. The low contents of Ru-106 (3-5% of total radiocaesium activity) and Co-60 (less than 3-5% of MPBB) were found in 43 persons. No other nuclides were measured in 1987.

Average radiocaesium activity is (122±6) nCi, 97% percentile is 500 nCi and 90% percentile is 200 nCi. No correlations between occupations and incorporated activity burdens were revealed (0.008, p=0.2%). Weak correlation was found between age and incorporated activity (correlation coefficient=0.15, regression coefficient=(31±6)).

Thus, the internal exposure of Chernobyl NPP employees does not corresponds to the occupational activities and their average internal exposure whole body doses were calculated to be equal to 0.6 mGy.

CONCLUSIONS

1. Internal exposure of thyroid, lungs and whole body of persons died from combined radiation exposure could not make effect in clinical outcomes excluding 2 patients (7% of whole lethality).
2. Internal exposure whole body doses in two aforementioned mortal cases represented less than 15-30% of total whole body dose at the moment of death.
3. Lung doses of died patients did not exceed 0.5 Gy.
4. Thyroid doses of died patients did not exceed 25 Gy.
5. Whole body doses of the internal exposure of died patients were less than 3-4% of external exposure doses given in [2].
6. Accidental exposure doses of thyroid in Chernobyl NPP employees did not exceed 20 Gy, where average dose was (1.5±0.2) Gy and 90% percentile is equal to 1.8 Gy.
7. Accidental internal exposure of Chernobyl NPP employees was finished at the end of May, 1986 (1 month after the accident).

LITERATURE

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PAPER TITLE Radiological consequences of accidents on the nuclear-powered submarines.

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

The experience of medical provision of the personnel, suffered during nuclear installations accidents on the nuclear-powered submarines shows, that during such accidents personnel is exposed to Gamma-radiation in doses, which cause acute radiation syndrom (ARS), up to the extremly grave degree of heaviness. Besides, Betta-radiation of radioactive gases, penetrating into nuclear-powered submarine accomodations during severe accidents of nuclear installations, gives rise to acute radiation injuries of skin (ARI) of the personnel.

Gamma-radiation collective doses, intercepted by nuclear-powered submarines personnel during nuclear accidents ranged from 17 up to 74 people/Gy, and average individual doses - from 0.16-0.38 Gy up to 0.57-0.22 Gy. Collective dose of Betta-radiation on the skin was equal to 218.9-816.6 people/Gy, and average individual dose on skin was from 2.3-4.8 to 5.7-4.0 Gy. Individual dose on thyorid at the cost of summary iodine radionuclides inhalation averages between 1.9-2.8 and 2.4-10.3 Gy.

Death as a result of acute radiation syndrom determines the average risk for nuclear-powered submarines personnel equal to 0.5-0.8E(-4) per year. Risk of acute exposure stochastic effects comprise 9-14% of the risk of acute radiation syndrom fatal results.

Harm index expressed in years of expected life expentancy reduction for 1000 people accounts for 2.3-3.9. Probability of exposure fatal consequences comprise 10-15% of the total risk magnitude of the death for various reasons (fires, explosions, sections flood etc.) Thus, total magnitude of the death risk accounts to amount of the order of E(-3).

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PAPER TITLE Dose characteristics of the IHEP reference fields

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5 (see page 7)

ABSTRACT (See instructions overleaf)

One of most important problem for radiation monitoring procedure at IHEP is to create and investigate the reference radiation fields, which adequately simulate the accelerator radiation conditions, for the goals of correct dosemeters calibration It is assumed that the IHEP reference fields set has to include the following groups of reference fields: 1) based on neutron radionuclide sources, 2) behind the biological accelerator shield (including high-energy fields), 3) creating by charged particles such as muons

The results of dose characteristics investigation of the IHEP reference fields based on radionuclide sources ^{239}Pu-Be and ^{252}Cf are presented in the paper The neutron ambient dose equivalents measured in these fields are compared with those calculated from the neutron spectra measured by multysphere spectrometer and monocrystal stylbene spectrometer

The instruments being used in the dose characteristics measurement as well as the calibration procedure of these instruments on the primary National Standard neutron and proton fields at VNIIFTRI (State Standard of Russia) are discribed

Neutron response functions of the instruments are calculated in energy range up to 1 GeV in order to analyze the systematical errors arising when the instruments could be applied in the high-energy reference fields Configuration of a set of dosimetric and spectrometric instruments requiring for high-energy reference fields investigation is discussed Preliminary measured dose characteristics of the IHEP high-energy reference radiation fields are given

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PAPER TITLE : Precision Calculation of the Neutron and Proton Absorbed Dose Distributions in the Tissue-Equivalent Phantom at Energies up to 100 GeV

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MAJOR SCIENTIFIC TOPIC NUMBER: 5.5

ABSTRACT

The calculational investigation of the high-energy neutron and proton absorbed dose distributions in the tissue-equivalent plate phantom was carried out. The calculations were performed with the ROZ6H/SADCO-2 program package.The ROZ6H discrete ordinates code is based on the solution of the Boltzmann equation for neutral and charged particles in the 1-D geometry by numerical schemes of second-to-fourth order of accuracy. The SADCO-2 nuclear data library contains cross-sections for neutrons (with energy 0.02 eV–10 TeV), photons (0.1–20 MeV), protons, pions, kaons (20 MeV–10 TeV) as well as a stropping power for all charged particles. The ROZ6H/SADCO-2 package allows one to calculate particle spectra with error about 5–10%. The absorbed dose (D) was calculated as follows : $D = \int F_i(E) k_i'(E) dE$, where $F_i(E)$ is the spectrum of particle of the i^{th} type; $k_i'(E)$ is the total kerma factor for neutrons and photons at energies below 20 MeV and it is the partial one for neutrons, protons, pions and kaons at energies E>20 MeV. (The partial kerma factor is the sum of energy transferred to charged particles excluding fast (E>20 MeV) charged hadrons.) The calculated absorbed dose distributions on the phantom depth for normal and isotropic incident neutrons and protons with energies 20 MeV–100 GeV are presented in the report. The presented data are assigned both for specialists in the radiation therapy and for working up recommendation of acceptable levels of irradiations.

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PAPER TITLE : Calculation of High-Energy Radiation Attenuation Parameters in Iron and Concrete Shielding

AUTHORS NAMES: D.V. Gorbatkov and V.P. Kryuchkov

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MAJOR SCIENTIFIC TOPIC NUMBER: 5.5

ABSTRACT

The calculational investigation of radiation fields formed by high-energy neutrons and protons behind iron and concrete shielding was carried out with the ROZ6H/SADCO package. The ROZ6 discrete ordinates code with SADCO-produced multigroup data sets provides a high accuracy of the high-energy hadron transport calculations. The dependencies of the following components of radiation on the shielding thickness and source energy were determined: thermal neutrons; fast neutrons; neutrons with E>20 MeV; photons with E<20 MeV; charged particles. The formula for approximation of dose equivalent (H) attenuation in the transition range of shielding thickness (t) was proposed : $H_i(t) = exp(a_i^0 + a_i^1 t + a_i^2 t^2)$.
For the range of equilibrium attenuation the calculated data were described by the 'length of relaxation' method: $H_i(t) = H_{i0}(E) exp^{-t/\lambda}$. The absence of equilibrium between high-energy (E>20 MeV) and low-energy (E<20 MeV) radiation components in the iron shielding was taken in account : $H(t) = H_o^{(E>20)} exp^{-t/\lambda_h} + H_o^{(E<20)} exp^{-t/\lambda_l}$. The following designations in the above relations are used: a_i^0, a_i^1, a_i^2 are the parameters; λ_i is the attenuation length of radiation component of the i^th type; H_{i0} is the dose build-up factor of the i^{th} type radiation. Calculated approximation parameters for all mentioned components are presented in the paper. The uncertainty of presented data is about 5–10%. The comprehensive comparison of calculated attenuation lengths with numerous experimental data shows a high reliability of the proposed approximation. Presented data are assigned for the practical radiation problems solution on high-energy proton accelerators at energies 20 MeV–300 GeV.

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FORM FOR SUBMISSION OF ABSTRACTS

PAPER TITLE : Partial Kerma Factors for High-Energy Hadron Absorbed Dose Calculations

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MAJOR SCIENTIFIC TOPIC NUMBER: 5.5

ABSTRACT

A much used for absorbed dose calculations at neutron energies E<20 MeV kerma approximation $D_i = \int F_i(E) k_i(E) dE$, can not be extended to a high-energy range, because there is not the charged particle equilibrium. It was proposed to calculate the hadron (n, p, π, K) absorbed dose at energies 20 MeV–10 GeV by presented equation using partial kerma factors (k'_i(E)) instead the total one (k_i(E)). The partial kerma factor may be calculated as $k'_i(E) = \sigma_i \gamma \sum_j E_j + (dE_i/dx)/\rho$ for p, π, K; $k'_i(E) = \sigma_i \gamma \sum_j E_j$ for n. In the above relations F_i(E) is the spectrum of hadron of the i^{th} type; σ_i is the microscopic total cross section of hA-interaction; γ is the number of target atoms per unit mass; dE_i/dx is the stopping power of primary charged particle; ρ is the mass density; E_j is the energy of produced in hA-interaction heavy charged particle (A>1) or the energy of proton with E<20 MeV. Thus, the partial hadron kerma factor is defined as the sum of energy transferred in a reaction to charged particles with high stopping power only: d, t, ^{3}He, α, residual nuclei and low-energy protons. The equilibrium conditions for those particles exist always in a practice. The total and partial kerma factors calculated with hybrid (at E<100 MeV) and cascade-evaporation (at 100 MeV<E<10 GeV) models are presented. Good agreement of calculated total kerma factors with the experimental one at energies E<70 MeV and with calculational literature data at E<1 GeV was shown. Calculated partial kerma factors for the energy range 20 MeV–10 GeV are assigned to solve the practical problems of hadron absorbed dose calculation.

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PAPER TITLE **Recombination Index of Radiation Quality of High Energy Neutron Beams.**

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MAJOR SCIENTIFIC TOPIC NUMBER 3 ..... (see page 7)

ABSTRACT (See instructions overleaf)

Neutrons have been considered created in beryllium, copper and lead targets bombarded by 660 MeV protons from phasotron of Joint Institute for Nuclear Research, Dubna. Mean neutron energies in the beams formed at zero degree respectively to the direction of proton beams equal approximately to 350 MeV, 280 MeV and 200 MeV, respectively. Recombination index (RIQ) has been taken as a parameter describing the radiation quality.

Measurements performed using a 3,8 cm^3 recombination chamber filled with TE gas up to 500 kPa. Two voltage values : U_R =30v and U_S =800v applied to the chamber electrodes sequentially. The recombination index of radiation quality (Q_R) derived from measured currents of the recombination chamber using the following relation :

$Q_R = (1/R) [1-i(U_R)/i(U_S)]$ where i - are the ionisation currents of irradiated recombination chamber related to the monitor reading; U_S - is the voltage applied to the chamber ensuring nearly saturation conditions; U_R - is the recombination voltage chosen during the calibration in a reference gamma field, such that : $i(U_R)/ i(U_S) = 1 - R = 0.96$

All result agree within experimental and calculation uncertainties (10%) The RIQ values practically do not differ also with values of quality factor defined in ICRP -21

It is shown only minor difference of RIQ for three beams considered.

RIQ decreases with depth in the phantom water 5 at 1cm depth to approximately 3.2 at the depth of 12 cm, where a broad maximum of depth dose distribution taken place. The radial distribution of RIQ are similar for three beams too.

It can be concluded that there is no essential differences radiation quality of the beam. The beam with energy of 350 MeV is preferable, because of higher beam intensity.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Multisphere Neutron Spectrometry Measurements In A High Energy Neutron Beam

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5 ..... (see page 7)

ABSTRACT (See instructions overleaf)

The multisphere neutron spectrometer detectors (commonly referred to as Bonner Spheres), augmented with threshold neutron detectors, were exposed to well characterized beams of neutrons generated by striking either a thick tungsten target with a beam of 800 MeV protons or a thin lithium target with 800 MeV protons at the Los Alamos Weapons Neutron Research facility. The experiments were performed in the 15° flight path from the tungsten target and in the 0° flight path from the lithium target. The purpose of these experiments was to compare the neutron energy spectra determined from unfolding the sphere and threshold detector data with the energy spectra determined by time of flight measurements for the neutron beam. The Bonner sphere set, including 5.08cm, 7.62cm, 12.70cm, 20.32cm, 25.40cm, 30.48cm, and 45.72cm diameter moderators, plus a Bare and a Cd covered detector was augmented with a $^{12}C(n,2n)^{11}$ threshold detector and a Bismuth fission counter to provide data for the energy regions above 20 and 50 MeV respectively. Paired sets of ^{6}LiF and ^{7}LiF TLD chips were used as the neutron sensitive elements at the center of the moderator spheres. Several different average energy neutron beams were obtained from the tungsten target by inserting various thickness of polyethylene filters in the beam upstream of the detector location. The integrated beam intensity during each exposure was monitored by a small plastic scintillation counter inserted in the beam, which provided a count proportional to the dose equivalent, calculated for each filter thickness. The average energies for the beam spectra from the tungsten target ranged up to approximately 400 MeV, and peaked near 800 MeV from the lithium target. The data from the detectors larger than the beam diameter were corrected using Monte Carlo calculations. The corrected data were analyzed using four different spectrum unfolding computer codes to determine which technique would give the best results, ie. the best match with the time of flight measurements. The results indicated that all of the unfolding codes were reasonably good in determining the gross neutron field parameters of dose, dose equivalent, and average energy. The conclusion can be made that the multi-detector neutron spectrometer will give sufficiently accurate spectral information for neutron energies up to 800 MeV for dosimetry and shielding evaluation purposes.

IRPA9
1996 International Congress on Radiation Protection
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Vienna, Austria

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Abstract No 90809
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FORM FOR SUBMISSION OF ABSTRACTS
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PAPER TITLE

IHEP's Set of Neutron Reference Fields

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5

ABSTRACT

Measured spectra for 8 neutron fields and superposition method for dosimetric apparatus calibration with the help of these neutron fields are presented. 7 of these fields are based on ^{252}Cf and Pu-α-Be sources and the 8th one is the field of high energy leakage neutrons behind the biological shielding of proton synchrotron on 70 GeV. Mean energy of the neutron spectra of these fields are from 0.11 MeV to 3.0 MeV and 79 MeV, correspondently. The neutron spectra of the first 7 fields have been measured with the help of Bonner spectrometer and scintillation spectrometer, based on single crystal of stilbene. The 8th spectrum have been measured with the help of Bonner spectrometer supplemented by modified detector with 5 cm copper converter and by scintillation detector which detected an inelastic reactions ^{12}C(n,α) ^{9}Be and ^{12}C(n,n)3α. It has been shown that the superposition method allows one to calibrate dosimetric detectors more correctly than commonly used techniques.

TRANSMISSION FACTORS FOR THE NEUTRONS FROM SOME RADIOISOTOPE PRODUCTION REACTIONS FOR PET

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INTRODUCTION

The purpose of this work is to present the dose transmission factors of ordinary concrete, for neutrons originated during the production of some radioisotopes used in positron emission tomography (PET). PET is a technique used in nuclear medicine which use in image diagnostics is in permanent increasing, because of two unique characteristics : the spatial resolution is independent of depth, and, it is possibly to make absolute corrections to tissue attenuation. These characteristics make it possible to determine the real concentration levels of the radioactive tracer in tissues, as well as the parameters related to the functioning and metabolism of tissues and organs.

The availability of adequate positron emissors is one of the restrictions for the use of this technique in nuclear medicine Because of the short half lives of these isotopes (from 1 minute to 2 hours approximately), it is a common practice to install a particle accelerator capable to produce them in the surroundings of the PET. The positron producer isotopes commonly used are: ^{11}C, ^{13}N, ^{15}O, and ^{18}F, which are produced by means of the reactions $^{16}N(p,\alpha)^{11}C$, $^{16}O(p,\alpha)^{13}N$, $^{13}C(p,n)^{13}N$, $^{15}N(p,n)^{15}O$, $^{14}N(d,n)^{15}O$ and $^{18}O(p,n)^{18}F$.

This work is only related to the shielding necessities of cyclotrons with accelerating protons up to energies from 10 to 13 MeV, and for the reactions $^{18}O(p,n)^{18}F$ and. $^{13}C(p,n)^{13}N$ This is so, because some recent studies indicate that the optimum energy for these types of cyclotrons is around 11 MeV. As well, the reactions $^{18}O(p,n)^{18}F$ and. $^{13}C(p,n)^{13}N$ generate the most restrictive conditions for the shields [1].

The main requirement for shielding arise from rapid neutrons, coming from the nuclear reactions that are produced in the target, which have a high penetration and a high quality factor. The gamma radiation produced in the target normally is not relevant for the dimensioning of the shields, because the last ones must be capable to attenuate photon radiation originated in them, in particular by capture of thermal neutrons in hydrogen.

DETERMINATION OF THE SPECTRUM OF INCIDENT RADIATION

The spectrum of incident radiation was estimated supposing that the projectiles are fully stopped in the thick target, and that it may be neglected the lost of particles in nuclear reactions. The calculation of neutron spectrum was made using the cross sections for thin target , $\sigma(E)$ [2-7], and the number of incident particles interacting between the beam energy and the reaction threshold energy.

The number of neutrons produced by protons with energies between E and dE is $\sigma(E)$ N dE/(dE/dx), where N is the atomic density of the isotope considered in the target, and dE/dx is the lost of energy per unit length travelled by the particle [8].

The energy scale of the neutrons emitted is determined by analysing the energy distribution of the reaction, taking into account the masses of the intervening particles, the energy of the incident particle, and the angle between the velocity vectors of incident and emitted particles.

For neutrons emitted at 0°, when exist only one open reaction channel that produce a residual nucleus in ground state, the energy of the incident proton (E_p) and the emitted neutron (E_n) are related by $E_n=E_p+Q$, where Q represents the energy balance of the reaction. In the case that the residual nucleus remains in an excited state with energy W, the energy of the emitted particle will be $E_n=E_p+Q-W$. In both cases, the recoil energy of the residual nucleus was neglected, which is essentially valid if the mass of residual nucleus is much more than the mass of the emitted particle. Using this calculating method, it was possible to estimate the spectrum of neutrons emitted at 0° in the reactions $^{18}O(p,n)^{18}F$ and. $^{13}C(p,n)^{13}N$, in thick target condition, and for protons with energies between 10 and 13 MeV.

DETERMINATION OF THE TRANSMISSION FACTORS

It was used the discrete coordinates code ANISN[9] for obtaining the neutron spectra in the output of infinite slabs of ordinary concrete This code applies the method of discrete coordinates, or S_n, which is a numerical technique of finite differences, for solving the Boltzmann transport equation for steady state.

Because of the great variation of concrete compositions, we selected the TSF 5.5 concrete [10], with a density of 2.3 g/cm^3 and 15.5% by weight of water, as representative of ordinary concrete.

The source used was a parallel beam of neutrons, of unitary fluence, incident in almost right angle ($\cos\theta$= 0.9894) over the shield. The energy distribution of the beam was the estimated, in the form already described, for the reactions $^{18}O(p,n)^{18}F$ and. $^{13}C(p,n)^{13}N$. The calculations were made in xy (slab) geometry, with quadrature order 16 and Legendre polynomial expansion of order 3 (S_{16} P_3).

We used the cross sections corresponding to the coupled library VITAMIN C [11], with 171 neutron groups and 36 photon groups. That the library is coupled means that the secondary photons generated by the neutron interactions appear as sources for the gamma groups, which are subsequently transported by the ANISN code.

The neutron and photon spectra obtained outside the shield were converted to effective dose by means of the factors conversion presented in a publication based in the publication 60 of ICRP [12], for anterior-posterior (AP) irradiation geometry.

RESULTS

In figures 1 and 2 we present the transmission factors for effective equivalent dose (Sv cm^2), obtained for neutrons generated in the reactions $^{18}O(p,n)^{18}F$ and. $^{13}C(p,n)^{13}N$, with protons of 10, 11, 12 and 13 MeV, inciding almost normally on ordinary concrete. These factors take into account the primary radiation and the secondary gamma radiation generated in the shield.

CONCLUSIONS

The data presented are a suitable tool for making, in simple form, the approximated dimensioning of primary shielding, required for cyclotrons commonly used in the production of radioisotopes for positron emission tomography. It must be pointed that, the transmission factors obtained are slightly conservative in relation to real shielding, because they were calculated using a parallel beam of neutrons, inciding in almost right angle on the shield.

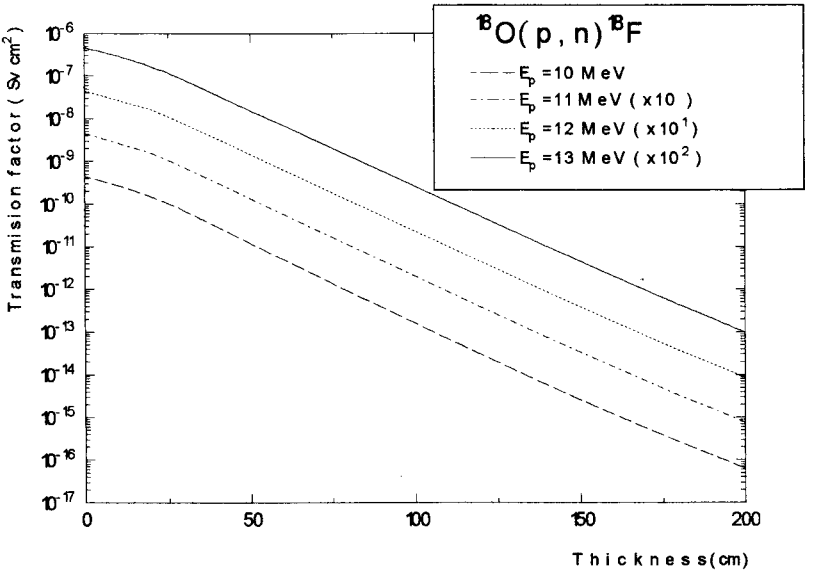

Figure 1: Transmission factors for the reaction $^{18}O(p,n)^{18}F$.

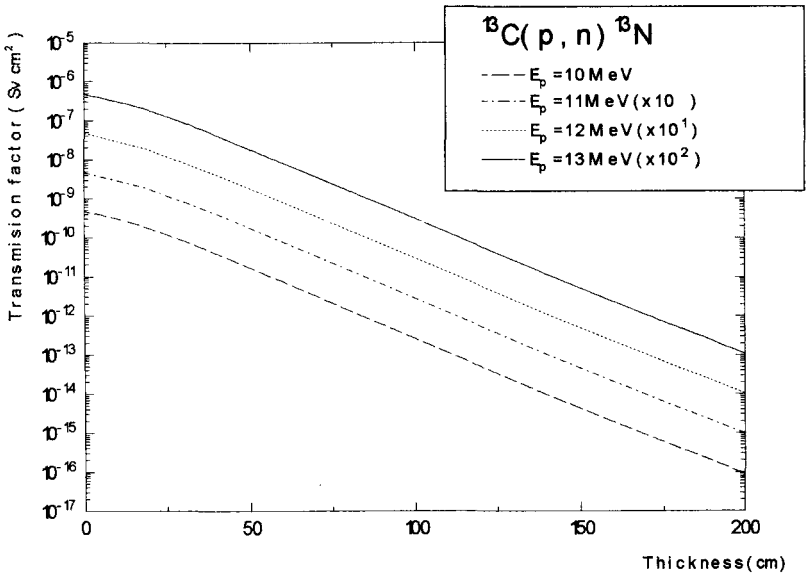

Figure 2: Transmission factors for the reaction $^{13}C(p,n)^{13}N$.

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REDUCTION OF EXPOSURES TO CYCLOTRON PERSONNEL INVOLVED IN THE MANUFACTURE OF RADIOPHARMACEUTICAL PRODUCTS

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Abstract

The management of the radiation exposure of cyclotron operators is particularly challenging in a commercial environment where the production cycle, from the onset of bombardment to the shipment of finished radiopharmaceutical products, can be as little as six hours.

This paper addresses the components of the radiation protection program - from management commitment and support, through policies and practical procedures to engineering provisions - which have successfully delivered significant reductions in personnel doses over the past five years.

Introduction

Amersham Healthcare/Medi-Physics currently operates five variable energy cyclotrons in the United States for the production of radioactive materials for the formulation of radiopharmaceuticals for use in diagnostic nuclear medicine. Cyclotrons accelerate positive or negative ions to bombard specific enriched targets either inside the machine or externally in a vault. The energy and current of the ion beam is adjusted to maximize production yields from specific nuclear reactions for specific product nuclides. Nuclides produced by bombardment are typically short half-life photon emitters such as I-123, Ga-67, Tl-201, In-111 and Rb-81.

Radiation Protection Issues

There is a need to provide reliable delivery of product to health care professionals for scheduled administration to patients. Due to the short half-life of these products, the time from the onset of bombardment to administration of the product to the patient may be less than 24 hours. This tight scheduling allows little time for unplanned delays or machine failures. In this context, it is essential to maintain effective radiation protection practices while meeting the needs of customers.

The main sources of radiation exposure associated with cyclotrons are activated targets, internal machine components associated with targets and beam production, and external components activated by recoil neutrons. Positive ion machines produce more activation. Machine operators are exposed to these sources of radiation during day-to-day operation of the machines including installation and retrieval of targets and routine maintenance. They are also exposed during repairs and major maintenance which includes rebuilding of machine elements.

Management Arrangements for Dose Reduction

For an operator dose reduction programme to be effective it must be supported directly by senior management through a clear statement of policy, through the setting of firm and aggressive limits on permissible exposure, and through provision of necessary resources. A culture must be established where it is unacceptable to receive dose for the sake of the product without proper regard for radiation protection.

Careful day to day management of procedures for cyclotron operation and routine maintenance is a major contributing factor in the reduction of dose. This effort is enhanced through daily accountability for operator exposures by cyclotron management; close cooperation with radiation protection staff; mandatory survey of radiation levels prior to any vault entry; and strict compliance with decay time requirements prior to vault entry. An effective preventive maintenance programme needs to be planned and implemented, and preparation made for contingencies for backup production in the event of a major machine breakdown. Regularly scheduled routine maintenance has resulted in improved production yields and reduced breakdowns, both significant in reduction of operator dose. Exposure of highly skilled technical staff should be limited during routine operations to ensure their availability during critical repairs in potential high dose rate situations.

The management of dose in the course of repair and major maintenance or rebuilds needs to be a cooperative effort between cyclotron management and the radiation protection staff. This effort is initiated by a meeting to review the type and scope of work to be done in advance. A plan is developed which specifies what work is to be done at what time and by which operator(s). Each operator is allocated an individual dose limit for the course of the project with an additional limit on collective dose. The commitment to observe all limits, policies and procedures is confirmed prior to the initiation of work. An agreement is reached to ensure that significant changes in the type or scope of work will be conducted only after additional review and discussion. One individual assumes a supervisory role in cyclotron operations and one in the radiation protection group.

The commencement of major repair and rebuilding operations is typically scheduled at least 48 hours following the end of the most recent bombardment. The optimum decay period has been determined by comprehensive radiation level profiles conducted during previous post-bombardment time intervals. Once the work is initiated, it is critical that radiation protection staff maintain vigilance in the area with the single objective of ensuring all practices are conducted in the interests of keeping doses as low as reasonably achievable without becoming involved in actual repair operations. It is necessary to maintain this independent role in the overall operation to repair the machine and restore operation as quickly as possible.

Engineering Provisions for Dose Reduction

Routine exposures received during normal cyclotron operations are usually limited by provisions which limit operations in relatively high radiation areas. These provisions include local shielding of components in proximity of targets; construction of external machine components from materials with minimal neutron activation or activation to short half-life or low activity nuclides; installation of external components outside primary neutron flux; carefully designed target

transport systems; automated post-bombardment target delivery; and effective instrumentation for measurement of radiation levels in appropriate locations. There has been a recent trend towards the use of negative ion machines which produce far less secondary machine component activation.

Additional provisions are available to enhance dose reduction in repair and major maintenance situations. These include the use of power tools to reduce the time spent in high radiation areas; the use of remote direct reading dosimeter systems to monitor operator dose in real time without interruption; availability of inactive replacement parts for rehearsal of repair operations; careful planning and use of localized shielding without increasing job time; proper handling and storage of radioactive machine parts with a means of easy identification of the location of smaller parts; and the avoidance of performing work on external components with internal components exposed.

Results

Amersham Healthcare/Medi-Physics implemented the measures described in this paper beginning in 1990 with the objective of reducing doses in the cyclotron facilities to less than the ICRP 60 recommended level of 20 mSv per year by 1996. The results of this initiative are provided in the following table of annual doses for the years 1989 - 1994. All doses are effective dose equivalents in mSv.

Year	Maximum Individual Dose	Collective Dose	Average
1989	63.1	2211	15.8
1990	49.5	2315	11.9
1991	33.0	1592	9.3
1992	30.1	1450	8.0
1993	31.8	1550	8.6
1994	21.3	1242	7.2

In five years following 1989, the maximum dose received by an individual was reduced by 66%, while the collective and average doses decreased by 44% and 55%, respectively. Results for the year 1995 are not available at the time of writing this paper, yet projections indicate that this trend has continued and additional reductions are expected. It is noteworthy that the significant reduction in the maximum dose received by an individual was not achieved by distributing the dose among more individuals.

Conclusions

The approach described has been very effective in reducing radiation doses to staff in a commercial environment where increased regular production has been required from predominantly positive ion cyclotrons.

Comparison between calculated and measured equivalent doses caused by heavy ions

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Ambient equivalent doses behind thick shields are calculated using the data derived by Weise (1). These data include neutron yields, estimated by using measured differential neutron and proton production cross sections of heavy ion reactions, and effective neutron dose transmission factors for ordinary concrete.

The ambient neutron doses are measured by normal REM-counters during the course of high energy physics experiments. Accelerated ions of carbon, nickel and gold are stopped in an iron beam dump. The specific energies (energy per nucleon) are between 1 and 2 GeV/n. The ambient doses at the location of the REM-counter are calculated using data for the reaction Ne $\rightarrow$ Fe at 2 GeV/n. The results are

accelerated ion	specific energy (Gev/n)	accelerated ions per second	calculated dose (µSv/h)	measured dose (µSv/h)
C	2.0	2.7 E+07	4.7 E-01	3.7 E-01
Ni	1.9	9.2 E+05	1.6 E-02	8.5 E-02
Au	1.0	1.8 E+06	1.9 E-01	3.5 E-01

For the carbon and gold ions the measured doses match well to the calculated doses. In the case of the nickel ions there is a discrepancy, the measured results are a factor of about 5 higher than the calculated results. These differences may be caused by beam losses outside the beam stop.

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PAPER TITLE **Application of low-pressure tissue-equivalent proportional counter in IHEP radiation protection**

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5. (see page 7)

ABSTRACT (See instructions overleaf)

The mixed radiation fields behind shielding of the 70-Gev proton synchrotron is composed of thermal and fast neutrons; high energy particles, pnotons and charged particles. One of the major techniques for direct dose equivalent measurements in such mixed radiation fields is based on low-pressure tissue-equivalent proportional counters (TEPC). The TEPC has been designed in collaboration with Institute of Biophysics (Moscow). The main technical data of TEPC are presented The instrument allows us to measure kerma and dose equivalent for photons and neutrons simultaneously in a single measurement. The separation technique in treatment of the pulse height spectrum with threshold in terms of linear energy is used to obtain the neutron dose equivalent. Calibration of the TEPC in terms of neutron and photon kerma are presented. The problem conserned with TEPC calibration in terms of ambient dose equivalent have been solved by the calibration in the National primary standard neutron fields of VNIIFTRI (State Standard Committee of Russia, Mendeleevo). TEPC are used for metrological ensurance of radiation protection at IHEP. The results of a simple routine neutron dosimeters calibration by TEPC in a mixed accelerator radiation fields are presented. Experimental results of dose characteristics measurements are presented for the reference fields of IHEP and JINR (Dubna, Russia). The results are compared with the other dosimetric systems ones.

VARIATION OF RADIATION DOSES ESTIMATED ON TIME RESOLVED PHOTON ENERGY SPECTRA OF PULSED RADIATION FIELDS

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INTRODUCTION

Since the dose and dose rate effectiveness for low LET radiations became evident by experimental results in animals as well as theoretical considerations, determination of radiation dose due to time resolved energy spectra of low dose and high dose rate radiation fields such as pulsed radiation fields has become important for radiation protection practice(1). From this point of view, scintillation spectrometer that can be used for analysis of photon energy spectra in the energy region from 0.1 MeV to 100 MeV has been developed for estimation of radiation doses due to X- or gamma-rays induced by operation of nuclear facilities. The main purposes of this paper are: to determine the time resolved photon energy spectra of pulsed radiation fields and to describe the differences between the mean dose rates averaged over total measuring times and the dose rates averaged over pulse duration periods during which radiations are emitted by the accelerator.

EXPERIMENTAL

1.Scintillation detector and its response function

Scintillation detector was constructed with a 7.6 cmϕ x 7.6cm$l$ BaF_2 crystal and a HTV R594 photomultiplier tube with breeder resistances. A 30 x 30 response matrix of the detector was constructed by modifying energy deposition spectra calculated by the EGS4 Monte Carlo code(2) into response functions whose energy resolution was experimentally determined by using radioactive sources. Fundamental properties of the spectrometer and an accuracy of unfolded spectra were investigated by using mono-energetic gamma-rays from radioactive sources and neutron capture gamma-rays from the E-3 beam facility of the Kyoto University Research Reactor. Photon energy spectra could be estimated within an error of about 8 % of the incident photons(3).

2.Measurements

Measurements of the time resolved photon energy spectra were carried out in the workplace around the electron linac facility of the Kyoto University Research Reactor Institute. A measuring point in the experimental room was about 1.1 m away from the wall of 2.7 m in thickness which was situated between the target room and the experimental room. A Ta target of the linac was placed in the target room at 2.6 m away from the wall and, as a result, the total distance between the target and the detector was about 6.4 m. The measurements was also performed outside the experimental room near the entrance for sending in experimental equipments. The second point was situated at about 8 m away from the first measuring point.

The linac was operated with the beam energy of about 30 MeV and the beam width of about 10 ns. The repetition frequency of a bunch of electrons(an electron pulse) was 300 pulses s^{-1}. Photon energy spectra were measured in the two time regions of time interval between

successive electron bunches, from 0 to 10 μs and 10 μs to 3.3 ms, by coincidence counting method using a ORTEC Digital Delay & Gate.

RESULTS AND DISCUSSION

1.Time resolved photon energy spectra

A time spectrum of photons (time series variation of photon counting rate) observed by the scintillation detector was shown in Fig.1. The abscissa of the figure shows the elapsed time (62.5ns /channel) from the incidence of a bunch of electrons into the Ta target. As is evident from the figure, most of photons were observed within a few microsecond and the maximum counting rate was approximately 10 counts s^{-1} and the total counting rate during 0 to 10 μs was almost 30 counts s^{-1}. This means that approximately one photon/ 10 electron bunches could be detected in this time period and that the distortion of energy spectrum due to chance coincidence of incident photons would be scarcely occurred here.

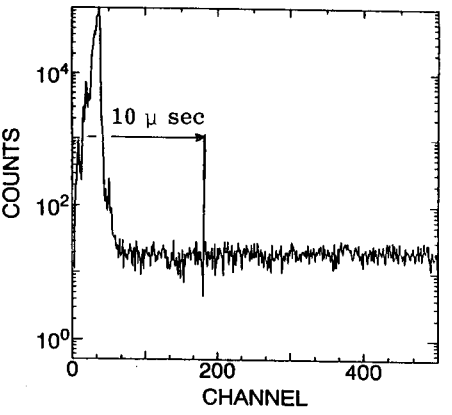

Figure 1. Time spectrum of photons observed by BaF_2 spectrometer.

Examples of photon energy spectra observed in the experimental room are shown in Fig.2. The solid and the broken histograms in Fig.2 represent the spectra measured during the former and the latter periods of the time interval between electron bunches from the linac, respectively. Since high energy photons and neutrons are considered to be generated in a few tens of nanosecond after incidence of electron bunches into the target, the photon energy spectrum shown by solid histogram seems to be mainly originated in high energy photons and their scattered ones and partly due to photons generated by neutron inelastic scattering in the wall. On the other hand, the result by broken histogram in the figure shows the energy distribution up to the energy of about 8 MeV. The higher energy part of this spectrum seems

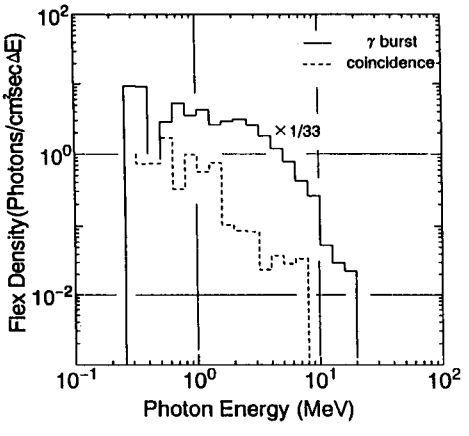

Figure 2. Time resolved photon energy spectra in the experimental room.

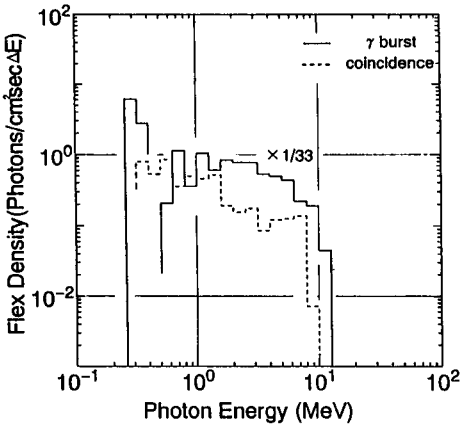

Figure 3. Time resolved photon energy spectra near the entrance of ex. room.

to be caused by neutron capture gamma-rays emitted from the surroundings of the spectrometer.

Photon energy spectra observed near the entrance to the experimental room are shown in Fig.3. It is evident from the solid histogram that the intensity of photons below 5 MeV becomes weak compared with that shown in Fig.2 and the photons more than 15 MeV do not have enough intensity to be detected by the present method. On the contrary, the broken histogram reveals us that the energy spectrum around 7 MeV which would be caused by neutron capture by the iron shutter of the entrance becomes more intensive than that in the experimental room.

2. Variation of dose rates within the interval of electron pulses

Air absorbed dose rates calculated on the energy spectra illustrated in Figs.2 and 3 are shown in table 1. The second and the third columns of the table show the absorbed dose rates given by the first intensive high energy photons and fast neutrons and those by gamma-rays from naturally occurring radionuclides and photons by slow neutron capture reactions, respectively. The mean dose rates during measurements are shown in the last column of the table. As is evident from the table, the dose rates in the former period (0 to 10μs of the pulse spacing time) are more than hundreds of times than those obtained in the later period (10μs to 3.3ms of the pulse spacing time). And, furthermore, the dose rates in the former period will show values by several times as much if the realistic high intensity pulse duration time shown in Fig.1 is taken into consideration.

Table 1. Variation of dose rates within the interval of electron pulses.

(unit:nGy/h)

time interval	0 to 10 μs	10 μs to 3.3 ms	mean
inside exp room	2.7×10^4	7.8×10^1	1.7×10^2
near entrance	8.3×10^3	8.5×10^1	1.1×10^2

In radiation protection practice, total doses are usually determined using personal dosimeters and dose rates are often interpreted as the mean doses illustrated in the last column of the table. But it is to be noted that the total doses are originated in repeated irradiation of extremely high intensity photons momentary emitted. So it may be concluded that dose rate effectiveness for irradiation by pulsed radiations with a duration less than a few nanoseconds must be investigated to get definitive information on the DDREF, though the ICRP has already recommend that the value 2 be used for the factor.

CONCLUSIONS

From the investigation stated above, conclusions are as follows:

(1) Energy spectra of pulsed photons determined by using the BaF_2 spectrometer changed remarkably with time. High intensity and high energy photons were observed within a few nanosecond after incidence of electron bunches into the target.

(2) The maximum dose rate caused by pulsed radiations became more than a thousand of times the mean dose rate averaged over the total measuring period.

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SHIELDING DESIGN CALCULATION FOR SPring-8 INSERTION DEVICE BEAMLINE

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1. INTRODUCTION

SPring-8, an 8 GeV class synchrotron radiation facility of Japan, is now under construction and the beamlines are in the design stage. The first beam will be emmited by October in 1997. The SPring-8 is composed of an electron/positron linear accelerator (linac), a booster synchrotron injector, and a storage ring (Fig.1). The linac is about 140 m long and accelerates electrons (meaning electrons and positrons) to 1 GeV. The booster synchrotron of which circumference is about 400 m, accelerates the electrons from the linac up to 8 GeV. The electrons are then injected into the storage ring, which is capable of storing circulating currents up to 100 mA at 8 GeV. The circumference of the storage ring is about 1500 m with 38 straight sections including 34 standard ones of 19 m in length and the 4 long ones of 40 m in length. The electrons emit synchrotron radiation while they are being deflected in the fields of the ring bending magnets or specially arranged magnets called insertion devices which are placed in the straight sections. The synchrotron radiation is introduced to an experimental floor by a beamline through ratchet-shaped bulk shielding walls of the storage ring as illustrated in Fig.1. Synchrotron radiation extremely high brilliance and high power density is available at the SPring-8 and

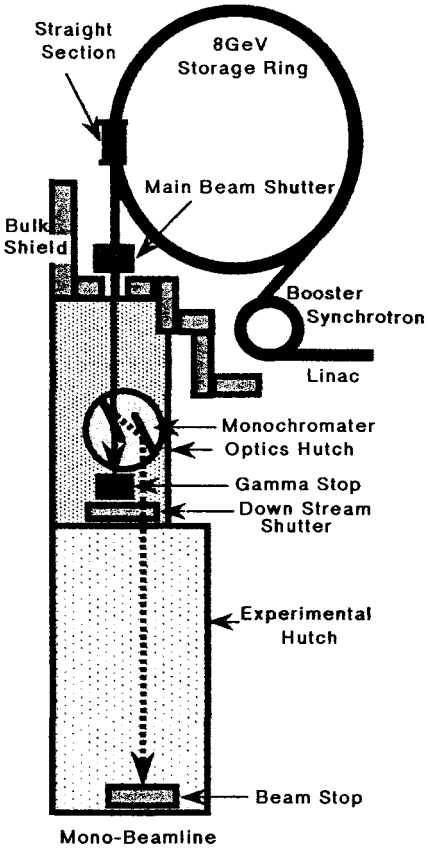

Fig.1 Illustration of the Spring-8 insertion device beamline for monochromatic beams.

therefore the shield of the beamlines for radiation safety must be designed under severe conditions.

The design criteria of 6.67μSv/h and 2.0 μSv/h are employed in occupied areas and at the boundary of controlled area.

Main shielding components for the beamline safety are a main beam shutter, down stream shutter, gamma stop, beam stop, and optics and experimental hutches. In the present paper, we will focus on the calculation for the beam shutter, the gas bremsstrahlung gamma stop and the hutch wall. In the calculation, the following 4 kinds of radiation sources were considered; (a) neutrons, photons and muons due to electron beam loss, (b)synchrotron radiation, (c)bremsstrahlung from the residual-gas molecules in the straight section of the storage ring and (d) photoneutrons generated at the gamma stop struck by gas bremsstrahlung. In this paper, we consider the monochromatic beamline as illustrated in Fig.1.

2. SHUTTER

Optimum thickness of the main beam shutter was calculated by using a modified Jenkin's formula(1) for neutrons and photons and a Swanson's formula(2) for muons. Gas bremsstrahlung calculation was also performed with tabulated data(3). As a results, we found that the tungsten of 18 cm thick or more are necessary on the standard sections. For the down stream shutter, the thickness was calculated by using STAC8, a shielding design code for synchrotron radiation(4). The beam stops are the same as the down stream shutter except for the mechanical movement.

3. HUTCH

One of the functions of the hutch is to shield the scattered photons of synchrotron radiation. Therefore, the structure of hutch depends strongly on the source. The thickness was calculated by using STAC8 as functions of a scattering angle and polarization.

In case of a high-energy inelastic scattering wiggler beamline(5), the key parameters are as fallows; the total power of 18.0 kW, the period length of 12 cm, the peak magnetic fields of 1 T, the period of 37 and the critical energy of 42.7 keV. The shield wall composed of a lead plate of 30 mm thick and an iron plate of 10 mm thick were found to be required for the side wall of the optics hutch including a local shield at 2 m distance from the beam.

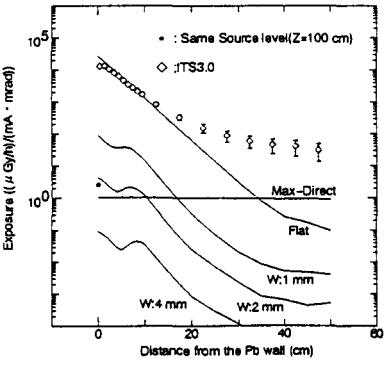

Fig.2 Dose distribution due to ground shine on the floor outside the hutch. (Photon source is scattered photons in 90 ° direction by Cu at 2m from scattered point. W denotes width)

The ground shine which is caused by the scattering photons from the concrete ground were also considered. For the purpose, another type of hutch wall composed of lead plate of 10mm thick sandwiched by iron plates of 5 mm thick was employed. The calculation was performed with ITS3.0(6) and G33-GP(7), assuming synchrotron radiation

perpendicularly scattered by Cu scatterer. As shown in Fig.2, the calculational result shows that the lead band of 30 cm or more in width and 2 mm thick is required along the foot of the hutch for attaining the dose which is equal to that at 1 m high from the floor.

4. GAMMA STOP

Gas bremsstrahlung is generated by the interaction of the stored electrons or positrons with residual gas molecules or ions in the storage ring vacuum chamber, which becomes important especially in the straight section because of an invasion into the beamline. Gamma stop made of lead is a standard beamline component of the SPring-8 to prevent bremsstrahlung from expanding to the down stream of the beamline.

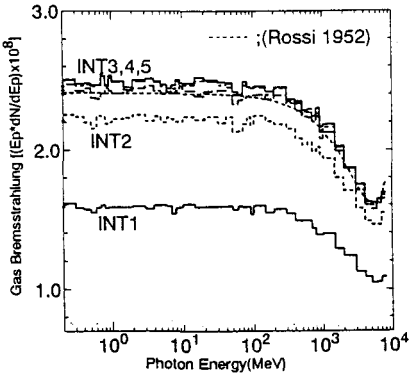

Fig.3 Gas bremsstrahlung spectra depending on the number of the interactions generated by EGS4 resulting from 0.1A 8GeV electron and interacting with 0.1205g/cm^2 air. (INT1;single interactions, INT2;double interactions, INT3.4.5; triple or more interactions, ---;theoretical data[8][9])

The residual gas pressure is designed to be below 0.133 μPa(10^{-9} torr). The EGS4(10) calculated gas bremsstrahlung spectra are shown in Fig.3. As shown in the figure, the gas bremsstrahlung generated within single interactions between the electrons and 0.1205 g/cm^2 air molecules is about 60% of that generated within multi-interactions. The bremsstrahlung is nearly saturated within triple or more interactions. In case of the interactions with 0.01205 g/cm^2 air molecules, the gas bremsstrahlung generated within single interactions is almost saturated. Most of the bremsstrahlung is emitted within 0.1 mradian.

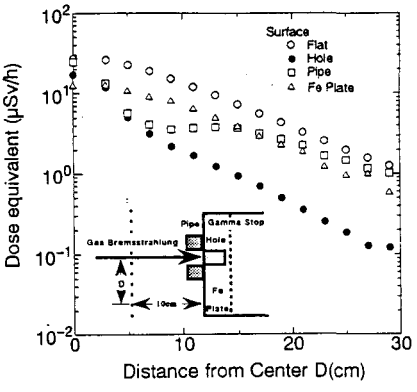

Fig. 4 Dose distribution due to back scattering by the gamma stops of various shapes. (Doses are at the point D(cm) from the center of the beam on the plane set at 10 cm in distance. Hole size;1cm in radius and 3cm in depth, Pipe size; 4cm in outer radius,3cm in thickness and 3cm in length, Iron plate; 5cm in thickness).

On the basis of the results of the EGS4 calculations, together with tabulated data(3), the gamma stop of 30 cm in thickness and 26 cm in diameter is recommended. Dose distribution due to the back scattering that depends on the shape of the gamma stop was calculated. As shown in Fig.4, the gamma stop with a hole is found to be beneficial to reduce the dose due to the back scattering.

The photoneutrons are produced through the interactions of gas brems-

ABOUT THE RADIATION ENVIRONMENT AND SHIELDING OF THE HEAVY ION ACCELERATORS

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INTRODUCTION

The aim of this work is to develop the semiempirical method of the radiation environment and shielding prediction in wide range of energy and mass numbers of accelerated ions (nuclei) on base of works having been published early /1-5/ in the light of contemporary information. The algorithm is based on some approximations, one of which permits to use accumulated information about parameters of radiation fields formed in proton-nuclei reactions. The major components of radiation field of proton and ion accelerators are neutrons and photons, as it is known. Therefore, only these components are considered. Energy range of the proposed algorithm is between Coulomb barrier and 10 GeV/nucleon.

RADIATION ENVIRONMENT ESTIMATION

The main quantities of radiation environment are neutron yield Y and radionuclides activity, Q, which depend on mass number of nucleus-projectile A_c, its atomic number and energy per mass unit E_c and also on flux of nuclei-projectiles (ions) F. With the aim of radiation environment prediction, it is suggested to use the following expression as a basis of neutron yield, Y_{ns}, estimation from stopping-length (thick) target :

$$Y_{ns} = \frac{L}{A_t} \sum_j \int_{R_{jth}}^{R_j} F_j \sigma_j dR \tag{1}$$

where R_j are the ranges of incident and secondary ions, R_{jth} is the range of ion j with the energy of the threshold of neutron production, A_t is the mass number of nucleus-target, L is Avogadro number, σ_j is the neutron production cross section in nucleus (j) with nucleus - target (t) interaction. At the ions energy more than 60 MeV/nucleon Y_{ns} was calculated by semiempirical formula

$$Y_{ns} = N\left(\sigma_{ns}\eta_{ns} / \sigma_p\eta_p\right) \tag{2}$$

where N is the number of protons equivalents /1/, σ_{ns} and σ_p are the cross sections nucleus - nucleus and proton - nucleus inelastic interactions, respectively, η_{ns} and η_p are the numbers of inelastic interactions of nuclei and protons incident on target, respectively. The function f takes into account fragmentation of nucleus-projectiles.

$$f = 1 + 0.19\,\eta_{ns}\sqrt{A_c} \tag{3}$$

Figure 1 shows the neutron yield from stopping-length target; the results have been obtained by (1),(2) and experiments which have been compiled in /5/. Radiation environment can be estimated by Y_{ns} in the first approximation /5/. It is worth while to consider residual activity at ion accelerators consisted of two components: induced by ions (or nuclei) Q_{ns} and neutrons Q_{nns}. The activity Q_{ns} can be represented as

$$Q_{ns} = N\frac{\eta_{ns}}{\eta_p} f Q_p \tag{4}$$

where Q_p is the activity induced by protons with energy $E_p = E_c\ A_c$ /6/. The activity Q_{nns} can be represented as

$$Q_{nns} = Y_{ns}Q_{np} / Y_p = N\sigma_{ns}\eta_{ns} f Q_{np} / \sigma_p\eta_p \tag{5}$$

where Q_{np} is the activity induced by neutrons at proton accelerator with the energy $E_p = E_c\ A_c$. On the data /6/ base it may expect that

$$Q_{ns} / Q_p = D_{ns} / D_p \tag{6}$$

$$Q_{ns} / Q_{np} = D_{nns} / D_{np} \tag{7}$$

where D_{ns} and D_p are the γ -radiation dose rates conditioned by Q_{ns} and Q_p respectively, D_{nns} and D_{np} are the γ - radiation dose rate conditioned by Q_{nns} and Q_{np} respectively. The results of measurements and calculations are compared in the table.

Table. The ratio of γ - dose rates measured /7/ and calculated with formulas (4)-(7) for Cu target irradiated by protons, ^{4}He and ^{12}C with energy 3.65 GeV/nucleon at other equal conditions.

Quantity	Measured	Calculated
$D_{^4He} / D_p$	3.4 ± 0.9	$3.2^{a)}$ $(4.8)^{b)}$
$D_{^{12}C} / D_p$	8.8 ± 2.4	$6.1^{a)}$ $(14)^{b)}$

a) Calculated with expressions (4) and (6); b) Calculated with expressions (5) and (7).
The formulas (4)-(7) gives possibility to estimate dose rates near of the ions accelerator.

DIFFERENTIAL DISTRIBUTION

Phenomenological model with using simple formulas for calculation of double differential cross sections of neutron production $d^2\sigma / dEd\Omega$ in order to predict of the radiation environment and shielding from neutrons has been developed in /2-5/. The basic regularities of the neutrons production in nucleus-nucleus interactions have been obtained by analysis of data /6,10,27-35/ from ref. /5/ and were used for development of the model. Analysis has been showed that there is the similarity in differential cross sections of neutron production for different collided nucleus at different energy. The neutron production cross section is presented in the form of the sum of four components:

$$d^2\sigma / dEd\Omega = \sum_{i=1}^{4} (d^2\sigma / dEd\Omega)_i \qquad (8)$$

The first component of the sum describes high energy part of neutrons spectrum. The second component describes neutrons generated in single nucleon collisions of the nucleus-projectile with nucleus-target. The third component is presented as the product of nucleus-nucleus interaction inelastic cross section σ_{ns}, number N and the double differential distribution of cascade neutrons in proton-nucleus interactions. The fourth component describes the distribution of the evaporation neutrons. The description of these components is given in /5/.

The information about $d^2\sigma / dEd\Omega$ permits to determine double differential neutrons yield from thick target, that lets to estimate the radiation environment both before and outside the shield. The dose of neutrons with the energy more then E_t emitted from the thick target at the distance r, outside the shield with the thickness x can be calculated as:

$$Hr^2 = \int_{E_t}^{\infty} (d^2Y / dEd\Omega) Bh \exp(-x/\lambda) dE \qquad (9)$$

The neutrons flux emitted from thick target with energy more then E_t outside the shielding for various angles can be expressed as:

$$\frac{dY}{d\Omega}(x) = \int_{E_t}^{\infty} \frac{d^2Y}{dEd\Omega} B \exp(-x/\lambda) dE \qquad (10)$$

where B and λ is buildup factor and attenuation length of neutrons fluence (dose), respectively, at the ordinary concrete, h is a conversion factor from the fluence of neutrons to the dose.

Neutrons flux distribution from thick Cu target bombarded by 86 MeV/nucleon ^{12}C ions with the intensity of 10^{11} s^{-1} in the heavy concrete measured /8/ and calculated by the algorithm is presented in Fig.2. The difference of data in Fig. 2 at the concrete thickness x=0 can be explained due to detection by carbon detector of the charges-fragment nucleus-projectiles emitted from the target. Comparison of the experimental data /9/ and calculated data obtained by the expression (8) is plotted in Fig.3. This comparison was carried out for energy spectrum of neutrons emitted from the thin gold target at $\theta = 0^0$ at the interaction the gold nucleus at 800 MeV/nucleon energy with it.

CONCLUSIONS

The phenomenological model for the radiation environment and shielding prediction has been examined by the experimental data. Comparison of calculated and measured data shows the reasonable agreement. Thus, the algorithm allows to estimate quickly the radiation environment and shielding of heavy ion accelerators in the energy range from tenth MeV/nucleon to tenth GeV/nucleon with atomic mass of nucleus projectiles and nucleus-target from Helium to Uranium.

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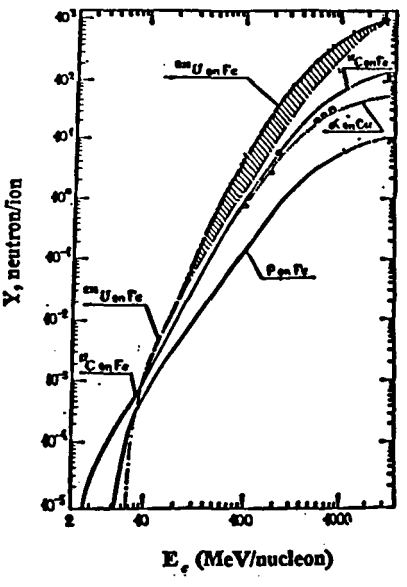

Fig.1. Total neutron yield from the thick iron or copper target per one ion with energy of E_0.

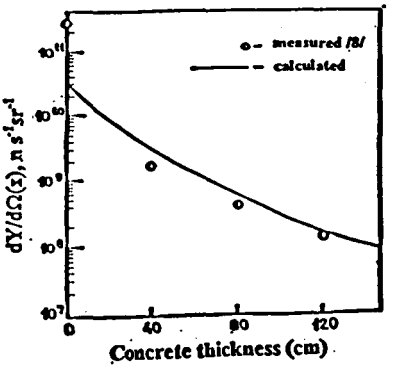

Fig.2. Neutron flux (E>20 MeV) attenuation in heavy concrete.

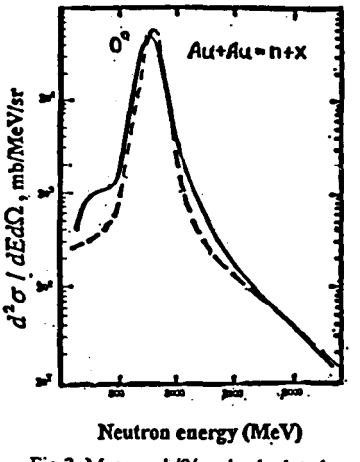

Fig.3. Measured /9/ and calculated neutron spectra produced by 800 MeV/nucleon Au on Au.

GAMMA RAY AND NEUTRON ATTENUATION OF PYRITE-POLYMER CONCRETE

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INTORDUCTION

Concrete is one of the widely used shielding materials, but has only moderate shielding properties for neutrons and gamma rays. Its hydrogen content, an element most effective for fast neutron shielding, is very low (~0.56 % by weight). Enriching concrete with hydrogen-containing materials would enhance its neutron shielding properties, while its gamma ray shielding properties as well as its mechanical strength would be improved by using heavy aggregates. Polyethelene $[(CH_2)_n]$ and polyvinyl chloride (PVC) $[(CH2\text{-}CH\text{-}Cl)_n]$ were introduced as part of concrete constituents to improve its neutron attenuation properties. Both of these compounds are rich in hydrogen atoms. Although PVC has less hydrogen density than polyethelene, the chlorine atoms in the polymer have a higher atomic number and a higher neutron absorption cross-section than either hydrogen or carbon atoms.

In order to improve the gamma ray attenuation properties, pyrite (FeS_2) heavy aggregate ore was selected, which has higher density and higher effective atomic number over ordinary concrete. The attenuation properties of Barite heavy aggregates were presented elsewhere(1).

In a radiation field of neutrons and gamma rays, the appropriate concentration of polymer and heavy aggregate can be selected to give the optimum total dose attenuation.

MATERIALS AND METHODS

The polyethelene used was in the form of small balls 3.3 mm in diameter and their specific gravity was 0.95. The PVC was in the form of cylindrical pellets, 3.3 mm in diameter, 3 mm in height, with a specific gravity of up to 1.4. The heavy aggregate ore consisted of up to 90% Pyrite and Pyrrhotite, up to 5% Chalcopyrite, some sphalerite and traces of other minerals. The fine aggregate which was basically feldspar and quartz has apparent, saturated surface dry and bulk specific gravities of 2.706, 2.661 and 2.642, respectively.

Type-1 Portland cement was used. The weight ratios of cement, of fine aggregates:course aggregate:polymer were, 1:1:1:1 (denoted as Py-PVC-1 and Py-Poly-1 for PVC and polyethelene concrete respectively), 1:1:1:1.4 (denoted as Py-PVC-1.4 and Py-Poly-1.4 for the two types respectively) and pyrite polymer free concrete (pyrite conc.). The water to cement ratio was 0.5.

A neutron dosimeter that approximates human body dose-equivalent over a wide neutron energy was utilized (type NM2-NE Technology). The gamma ray exposure was measured by a portable ionization chamber (model 660, Victoreen), with accuracy better than 5% at ^{137}Cs gamma ray energy. The neutron source was ^{241}Am-Be with a neutron emission rate of $6.6x10^6$ n/s and a tolerance of 10%. Its neutron dose-equivalent rate at 1 m was 66 μSv/h. The gamma ray source used was ^{60}Co, of $1.4X10^8$ Bq activity.
Broad-beam geometry was used, where, in most practical situation, the narrow beam conditions for neutrons do not prevail. The distance between the point source and the detector

was kept constant at 1m. The Lateral dimension of the specimen was selected to give maximum scattering towards the detector. A 60 cm. Lateral dimension was enough to scatter most neutrons and gamma rays.

RESULTS AND DISCUSSION

The natural log of relative neutron dose versus shield thickness is shown in Fig.1. The slope of the lines would be the neutron linear removal coefficient. Clearly the concrete type with higher polymer concentration shows higher attenuation and that polyethylene type exhibits higher attenuation over PVC type. The value of the neutron removal coefficient for Py-Poly-1.4 was approximately twice that of the Pyrite Polymer-free type. The natural log of relative gamma ray exposure rates are shown in fig.2. Unlike neutron attenuation, in this case the pyrite concrete of no polymer shows the highest attenuation followed by ordinary concrete and PVC concrete. The least attenuator was the polyethylene type. This is attributed, mainly, to the high density and the higher effective atomic number of pyrite ore over the rest types. The attenuation coefficients data presented here for pyrite-polymer concrete is close to that of barite-polymer type (1).

It may be concluded that in a mixed field of neutron and gamma rays, the appropriate concentration of polymer or heavy aggregate can be selected to give the highest attenuation of the total dose.

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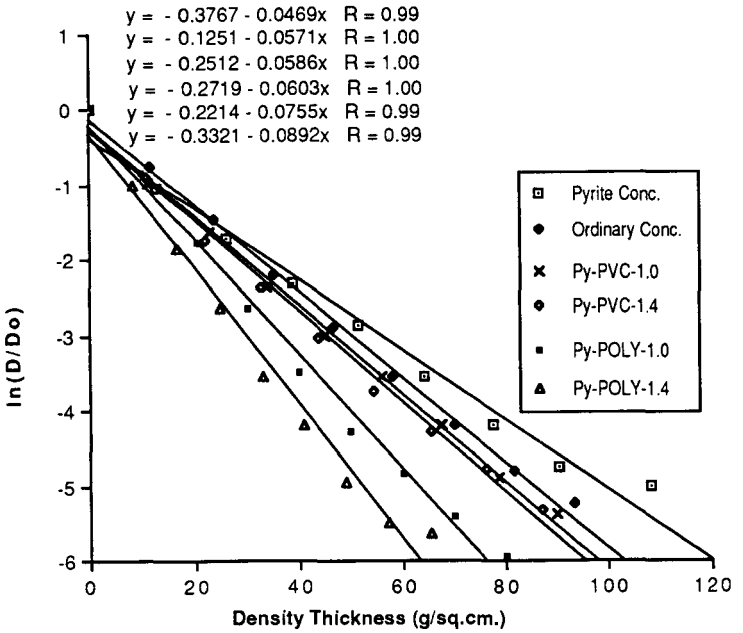

Fig.1. Natural log of relative neutron dose rate in broad beam geometry of pyrite concrete loaded with either polyethylene or PVC, shown with the sraight line fitting equations.
The source-to-dosimeter distance was 1 meter.

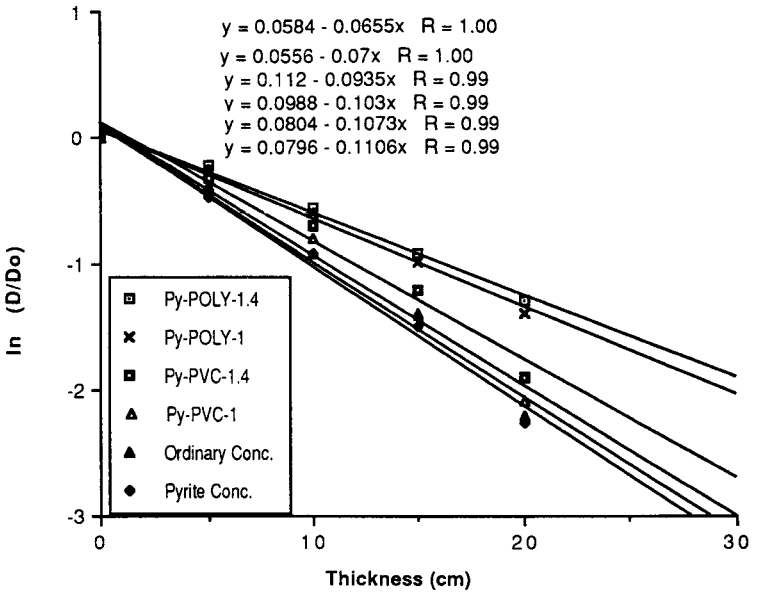

Fig.2. Natural log of relative gamma ray exposure dose rate versus thickness of pyrite concrete loaded with either polyethylene or PVC, shown with the straight line fitting equations. The source-to-detactor distance was 1 meter.

SHIELDING CALCULATION ON THE "TESLA" ACCELERATOR INSTALLATION IN THE INSTITUTE OF NUCLEAR SCIENCES "VINČA"

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Introduction

The TESLA Accelerator Installation is planned to be an ion accelerator facility consisting of an isochronous cyclotron - the VINCY Cyclotron, a heavy ion source - the mVINIS, a D^- and H^- ion source - the pVINIS, three low energy experimental channels, and five high energy experimental channels. The principal schema of the TESLA Accelerator installation is shown in Figure 1.

In creating its concept the main aim was to obtain a multipurpose machine, i.e. to enable the acceleration of ions in a wide range of specific charges. The first requirement was to obtain heavy ions with energies well above the Coulomb barrier, i.e. well above 5 MeV per nucleon, and make the machine a good instrument for research in nuclear physics. The second requirement was to obtain sufficiently hight intensities of protons and deuterons with energies above $\approx$ 60 MeV, and make the machine a sufficiently good instrument for applications in medicine.

The low energy experimental channels of the TESLA Accelerator Installation are the channels for experiments with the ions from the mVINIS Ion Source. These channels are: the channel for physics of multiply charged ions - L_1, the channel for surface physics - L_2, and the channel for materials science with medium energy heavy ions - L_3.

The high energy experimental channels of the TESLA Accelerator Installation are the channels for experiments with the ions from the VINCY Cyclotron. These channels are: the channel for nuclear spectroscopy and hyperfine interactions - H_1, the channel for heavy ion nuclear reactions - H_2, the channel for physics of very thin crystals and material science with high energy heavy ions - H_3, the channel for radionuclides production and research with high energy neutrons - H_4, and the channel for proton beam therapy - H_5.

High-energy heavy-ions are planed also in use for in-situ analysis of the elementary process in the materials exposed, including research on microdosimetry and track structure and energy deposition of accelerated particles.

The VINCY cyclotron will be able to deliver $\approx$ 1 μA of 36 MeV per nucleon O^{8+} ions, $\approx$ 100 nA of 23 MeV per nucleon Ar^{16+} ions and $\approx$ 700 nA of 7 MeV per nucleon Xe^{28+} ions. It will be able to deliver also heavy ions of lower energies - above $\approx$ 3 MeV per nucleon. This machine will be able to give $\approx$ 20 μA of 73 MeV deuterons, and $\approx$ 2 μA of 66 MeV protons. It will be able to give also deuterons of lower energies - above $\approx$ 43 MeV, and protons in the energy region of 22 - 36 MeV.

Shielding criteria for medium energy particle accelerators

Full understanding of the prompt radiation field generated requires knowledge of the primary interaction in the target material and the subsequent progression of the interaction products through the structure and surrounding experimental material and shielding.

All existing medium and high energy accelerators require some shielding for two reasons:

- to limit the biological effects on occupationally exposed persons and population exposure to acceptably values, according to basic radiation protection principle (ALARA - As Low As Reasonable Achievable (ICR 91, BSS 94)),
- to reduce experimental background conditions to tolerant values.

The design of a practical shielding arrangement for medium and high energy accelerator is in general a rather complex task for which no simple hand-book formulae are adequate. Nevertheless, it is possible by using a series of methods to make reasonably precise design in many instances and to allow for an adequate

range of design in many other cases (KOM 86).

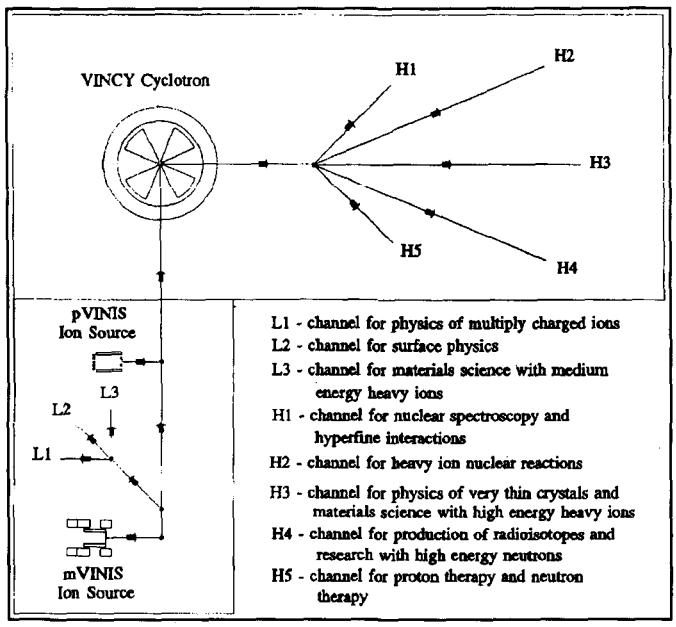

Figure 1. The principal scheme of the TESLA Accelerator Installation (TAI 93)

At its best the calculation for the shielding of a medium, as well as high energy accelerators involves, many simplifying assumption to make the problem tractable. Nevertheless the magnitude of the cost at stake and the engineering problems of support and handling forbid the casual use of large margins of safety, and require an attempt at precision compatible with the knowledge of the basic factors of production and propagation of radiation (PAT 73, TRS 88).

It is a fact familiar to all who have considered the shielding of a heavy particle accelerators that the radiation component that dominate in the prompt radiation field and the shielding problem, on the accelerator with performances like the TESLA Accelerator Installation have, is neutron radiation.

The beam losses interacting with components of accelerator structure, as well as beam of accelerated particles interacting with experimental targets, irradiated specimen or patient will give rise to high neutron fluxes, for which shielding must be provided, and also lead to component damage and induced radioactivity, with the attendant risk of increased radiation exposure of maintenance personnel.

Health physics problems at heavy-ion accelerators, with exception of accidental exposure close to or directly in the accelerator beam, are largely due to their neutron production. Thus an understanding of neutron production by charged particles is an essential weapon for the armoury of the health physicist, who must design shielding for such accelerators, estimate their production of radioactivity, and measure the radiation field they produce.

Experimental data on neutron yields and spectra from stopping protons are available at proton energies above 100 MeV; however, there is much less data for protons and for deuterons in the energy range between 50 and 100 MeV, where inelastic cross sections change rapidly with energy (BRO 83, FAS 76).

Shielding calculation

The first goal of all efficient accelerator shielding design is to attenuate the high radiation intensities produced by the accelerator and its associated equipment to levels that are acceptable outside the shielding, at minimum cost and without compromising the utility of the particle accelerator for its designed purposes. The

first and the most important and the most complicated step is determination of radiation sources, i.e. the space, time, particle and energy distribution of radiation fields (NCR 77). It is, therefore the most complicated step in solving radiation protection and shielding problems also in TESLA Accelerator Installation, especially due to the substantial lack of information about interaction cross section for the most important accelerated particles - protons and deuterons, in the energy range of interest (BRO 83).

In order to make critical analysis of existing project for cyclotron room we were used a simple exponential function to calculate ambient dose equivalent rate on the outer side of cyclotron shielding room, at working places for occupationally exposed workers and in the environment and working places occupied with other workers and member of the public.

According to planned characteristic of accelerated ion beams, their energies intensities, mass and charge of ions, we decided that the biggest problem for cyclotron shielding will be with deuteron acceleration. As a source of secondary neutrons we assumed will be sides of vacuum chamber, due to interaction of deuterons with average energy about 30 MeV/nucleon lost from acceleration. We were assume that the number of deuterons lost from acceleration, J, will be about 10^{13}, and that emitted neutron spectrum will be with average energy in the range from 5 to 20 MeV. On the basis of these assumptions we were calculate ambient dose equivalent rate, $\dot{H}$, out of the shield walls from equation (ALE 95):

$$\dot{H} = J \cdot f \frac{Y \cdot B \cdot h}{4 \cdot \pi \cdot R^2} e^{-x/\lambda} \qquad (1)$$

where Y - neutron yields per one deuteron lost from acceleration, B - buildup factor, λ - ambient dose equivalent attenuation length, h - neutron dose conversion factor far neutrons penetrated through shielding wall, x - shielding wall thickness, and R - distance from neutron source to the point of interest.

Conclusion

On the basis of this strongly approximated calculations we concluded that thickness of 150 cm of the side walls of accelerator room is not enough to perform safe radiation environment outside cyclotron room. We also concluded that in order to be able to use large basement area it is necessary to shield the space just under accelerator shielding room.

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INDUCED RADIOACTIVITY OF THICK COPPER TARGET IRRADIATED BY 3.65 GEV PROTONS

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INTRODUCTION

In recent years it is discussed the applications of the high energy proton accelerators in industry and medicine (for electrical energy production, radioactive waste transmutation, proton therapy etc.) (1,2). For these purposes it are necessary the machines with high intensity beams that may lead to complex radiation problems. Major such problem is γ-radiation induced radioactivity in accelerator equipment, which brings the main contribution to the personal total dose as rule more than 60%.

The present study deals with the induced radioactivity of the thick copper target irradiated by 3.65 GeV protons.

EXPERIMENTAL

The copper target of $\varnothing 100\times 130$ mm^2 dimensions is irradiated with beam of protons of 3.65 GeV energy and $7\cdot 10^{10}$ protons/cycle during 4 hours approximately. The experimental and monitoring conditions are the same as those described in (3). The irradiation control is performed with an ionization chamber. The total number of incident protons G is determined using the reactions of activation $^{27}Al(p,x)^{18}F$ and $^{27}Al(p,x)^{24}Na$. The values of the reaction cross sections are taken from paper (3). The number G determined with the activation detectors agrees with that one obtained with the ionization chamber within the measurement error (the chamber miscounts have been corrected). The following value is determined: $G=(5.0\pm 0.4)\,10^{13}$ protons.

The dose rate was measured with the use of the scintillation detector with NaJ(Tl) crystal of $\varnothing 63\times 63$ mm^2 dimensions and the multi-channel amplitude analyzer. Figure 1 illustrates the experiment configuration.

SIMULATION

The code package is developed for the calculation of the radioactivity induced by high energy protons. It allows :

1) To get the information about the stable elements and it nature abundance and the gamma radioactive nuclides with the half-life more than 5 minutes on the flight.

2) To calculate the spallation cross sections by using semiempirical Silberberg-Tsao's (4) and Rudstam's (5) formulae for protons and also to estimate the spallation cross sections for the nuclei. The peripheral reactions are also accounted in the this code.

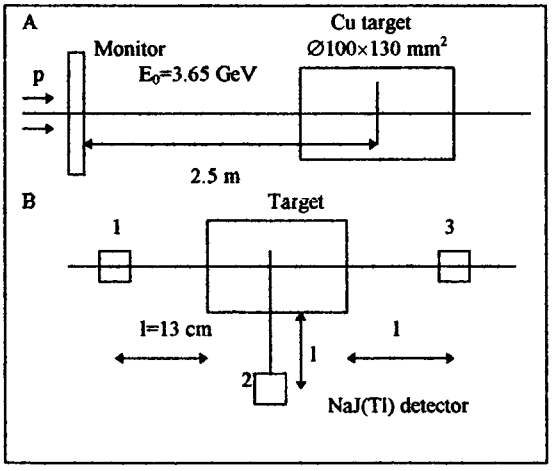

Figure 1. Geometry: (A) - target irradiation; (B) - dose rate measurement from the: 1 - beam inlet, 2 - side, 3 - beam outlet.

3) To calculate the induced gamma radioactivity with Silberberg-Tsao's formulae for the protons and nuclei. The data are equivalent to Barbier's ones (6) and are the dose rate existing at 1 cm of a supposed point source of 1 gram of target material, the dose rate existing in front of an infinite radioactive wall of great thickness, the specific activity and the total activation cross section.

4) To simulate the dose rate distributions around the thick targets irradiated by the high energy protons. The hadron inelastic interaction distributions over the targets are calculated by Monte-Carlo method using SHIELD code (7).

Below, the several comparisons of the numerical and experimental results are presented to demonstrate the capabilities of

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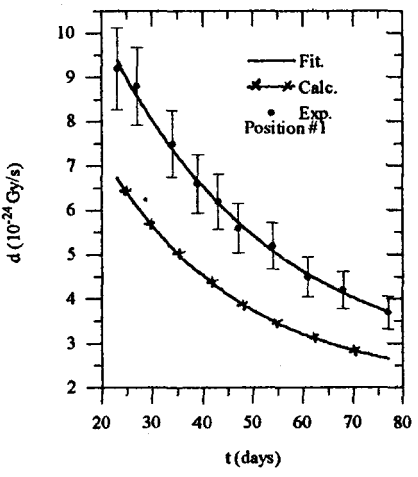
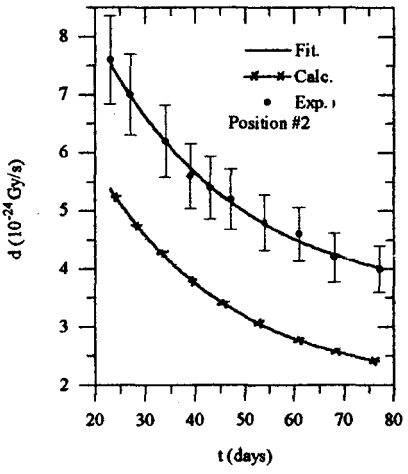
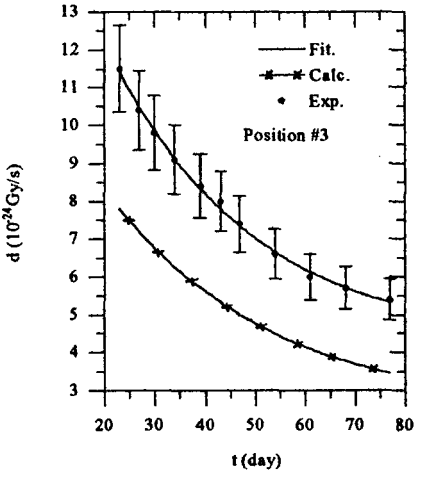
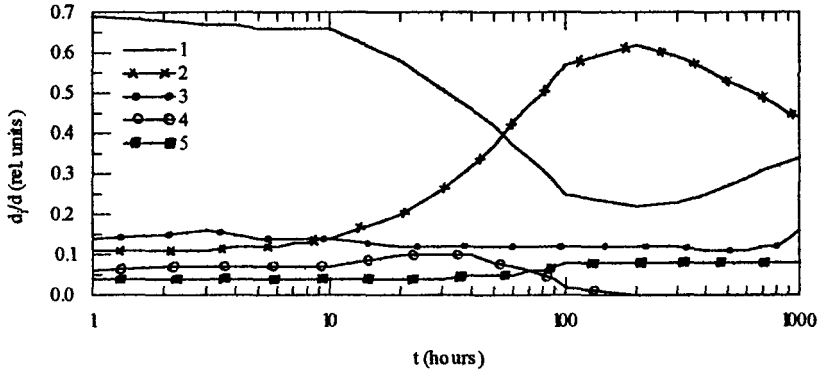






←↑Figure 2. Dose rates per one incident proton d at various detector positions, t is cooling time.

↓Figure 3. Relative contributions to dose rate d_i/d by various type hadrons at position #1: 1 - neutrons with E_n>10 MeV, 2 - primary protons, 3 - secondary protons with E_p>10 MeV, 4 - neutrons with E_n<10 MeV, 5 - $\pi^{\pm}$ with E_π>10 MeV; t is cooling time.



this code package. There are presented the simulation results of the dose rates in dependence on the time for the cylindrical Ø100×130 mm^2 thick copper target irradiated by 3.65 GeV protons. Relative contributions in the dose rate from radioactivity for the primary beam and the secondary particles (of neutrons with energy lower than 10 MeV and different types of the hadrons with energy more than 10 MeV) are obtained for the proton irradiation.

RESULTS AND DISCUSSION

Figure 2 shows the experimental data and simulation results of the dose rates per one incident proton d in dependence on the time at l=13 cm and various points for our target irradiated by 3.65 GeV protons. The indicated error is the monitoring inaccuracy that does not exceed 10%. The additive error of the γ-radiation dose rate has the systematic nature and is less than 15%. The exponential law $d=d_0+A\exp(-(t-t_0)/B)$ was used for the fitting by χ^2 criterion (8). The velocity of decay of this target induced radioactivity decreases with t increasing and is in good qualitative agreement with formulae by A.Sullivan (9). The calculated values d are obtained with Monte-Carlo method. The difference between calculation and experimental values of γ-radiation dose rate does not exceed 30% that it is allows to use our package for analysis of the induced radioactivity of thick targets.

Simulation data presented at figure 3 are the time-dependent relative contributions in the dose rates d_i/d from induced activity by the primary beam and secondary particles - neutrons n with energy $E_n<10$ MeV and different types of hadrons (i=p, n, $\pi^{\pm}$) with energy $E_i>10$ MeV. The induced activity is mainly produced by the primary protons and secondary neutrons with $E_n>10$ MeV. At the cooling beginning the secondary neutron contribution nature is the radioactive copper isotope production by the peripheral reactions. For this reason it is very important to account the single isotope production reaction channels. The approximations (4) may lead to the large inaccuracy for energies less than ~100 MeV (for the threshold region) and light nuclides, produced in spallation reactions (10). Therefor it is necessary to use experimental isotope production data libraries for the low energy hadrons, for example (11).

Also the package allows us to get of spatial distribution of radionuclides in thick target and other additional information to predict correctly the radiation danger levels for high energy proton accelerators.

ACKNOWLEDGMENTS

The authors are grateful to the Drs. A.V.Dementyev and N.M.Sobolevsky for the useful comments.

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ESTIMATION AND MEASUREMENT OF Fe-55 AND Ni-63 IN HARDWARE ACTIVATED AT HIGH ENERGY ACCELERATOR FACILITIES

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1,Introduction

At high energy accelerator facilities, accelerator components are exposed to primary and secondary particles with energy spectra from thermal to high energy,and as a result,they become radioactive through various types of nuclear reactions including spallation reactions[1, 2]. From a viewpoint of accelerator radiation protection, evaluation and measurement of radioactivities in these hardware is one of the most important things. Most radioisotopes produced are γ-emitters, and their radioactivities can be readily determined with a conventional radiation detector. While ^{55}Fe(EC decay,half-life:2.7y) and ^{63}Ni(β^- decay,half-life:100.1y) ,which are formed in accelerator components composed of iron and steel, emit no γ-ray, and from their nuclear properties, chemical separation of these radioisotopes from the irradiated samples is unavoidable prior to the measurement. These long-lived radioactivities accumulate in proportion to the machine operation time and become dominant in the residual activities several years after beam-off. These isotopes are conceivable to be produced abundantly because of the large cross-sections leading to their formation. Therefore,measurement and evaluation of these radioactivities are very important for safe handling of these accelerator components, and no report however is available on the measurement of ^{55}Fe and ^{63}Ni formed in the bulk hardware.

This paper is concerned with the development of measurement methods and the evaluation of ^{55}Fe and ^{63}Ni activities formed in iron and steel commonly used in accelerator hardware. The chemical separation of these radioisotopes from the irradiated materials was studied extensively. The ^{55}Fe and ^{63}Ni radioactivities were also estimated on the basis of the neutron fluxes measured by using a set of Au foils and Cu threshold detectors and compared with those experimentally obtained by a liquid scintillation counting method.

2,Experimental

In order to measure the integrated hadron fluxes(mostly neutrons),an activation method was adopted: a set of Au foils(about 1g) and Au foils wrapped with 1mm-thick Cd and copper discs(about 6g and/or 3.4g) were placed at various positions in the beam-line tunnel(EP2) of the National Laboratory for High Energy Physics(KEK) 12-GeV proton synchrotron. Iron and steel discs were placed at the same positions. Table 1 shows the cross-sections and effective threshold energies used for the calculation of integrated hadron fluxes in a copper threshold detector. The beam line tunnel is about 40m long, 6m wide and 4m high,being enclosed by heavy concrete blocks. The average intensity of primary protons was $(0.5\text{-}1.3)\times10^{12}$ protons per second, which was continuously monitored with SEC(secondary emission chamber). After the irradiation of a certain term,the radioactivities formed in the activation detectors and samples were measured with a high resolution Ge-detector connected to 4k PHA. To avoid the intereferences of short-lived radioisotopes,their radioactivities were measured about two weeks after the end of irradiation.

The ^{55}Fe and ^{63}Ni were measured with a liquid scintillation counter(LSC)(Packard Tri-Card 200CA/LL). Fig.1 shows the chemical separation procedures of iron and nickel from the irradiated discs and the preparation for the counting samples for LSC. 16mL of EX-H emulsfier was added to the iron or nickel solution thus separated in a glass counting vial of 20 mL and shaken vigorously. The milky gel-samples for LSC thus

prepared were allowed to stand for about one day and then measured several times for 30 min. A window of 0-70 keV was set to cover the entire energy region of β^--rays from ^{63}Ni(E_{max}=65.9keV), whereas for the measurement of ^{55}Fe radioactivity, three window widths of 0-7 keV,7-1200 keV and 0-1200 keV were set to measure the X-rays of 5.8 keV(K_α) and 6.4 keV(K_β) and Auger electrons of 5.2-5.8 keV. The extent of quenching was determined by the external standard method using a ^{137}Cs source.

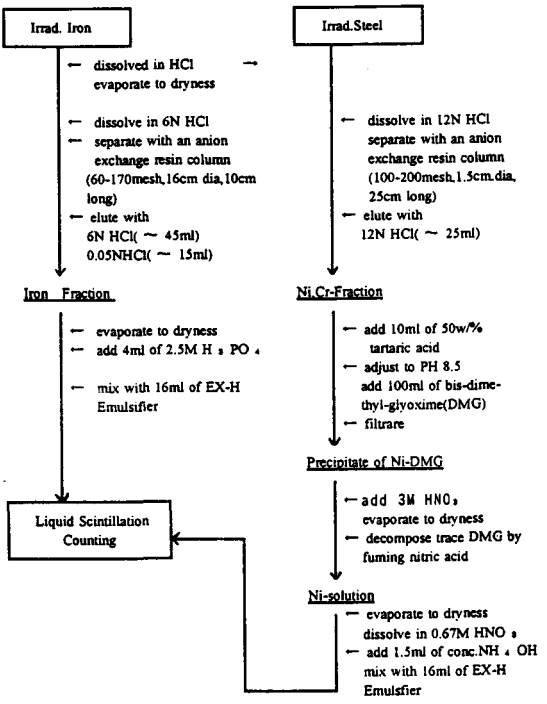

Fig.1 Procedures for chemical separation of ^{55}Fe and ^{63}Ni from irradiated iron and steel samples and preparation of counting samples for LSC

3, Results and Discussion

The chemical yields for the whole procedures shown in Fig.1 were about 91% for ^{55}Fe and 94% for ^{63}Ni. Both γ-ray and liquid scintillation spectrometries showed no contamination in the samples thus prepared. The counting sample preparations for LSC were simple and applicable to the ^{55}Fe and ^{63}Ni measurement with high sensitivity. For iron and steel samples. the detection limits were 0.82 Bq/g(Fe) and 0.035 Bq/g(SUS,Ni content of 15%) by assuming the counting time of 100 min. and 95% confident level(2 σ).

Thermal neutron fluxes were $(2.0\text{-}5.2)\times10^5 cm^{-2}s^{-1}$, and almost the same at any places in the EP2 tunnel. ^{63}Ni is primarily produced in steel discs through thermal neutron capture reactions of ^{62}Ni with natural abundance of 3.66%; its cross-section is 14.6 b. The steel samples used were all SUS discs with nickel content of 15%. As Table 2 shows, the ^{63}Ni radioactivities calculated from the thermal neutron fluxes were relatively good agreement with those obtained experimentally. On the other hand,the ^{55}Fe mainly formed through the thermal neutron capture of ^{54}Fe and (n,2n) reaction of ^{56}Fe, which have large cross-sections of 2.25 b and about 600 mb for the neutrons of thermal and 18 MeV, respectively. In order to calculate the ^{55}Fe activity, the cross-sections for the (n,2n) reaction were taken from ENDF/HE-VI library. Fig.2 shows the relation between the radioactivities calculated and exprimentally obtained at six different positions. The calculated values for the two positions downstream from beam-loss points and close to the beam line were considerably overestimated, indicating that the (p,pn) reaction of ^{56}Fe is

Table 1. Cross sections and effective threshold energies for calculating the integrated hadron fluxes using a Cu threshold detector

Nuclide	Threshold * energy(MeV)	Cross-section ** (mb)	Half-life
^{60}Co	13	7.8	5.27y
^{58}Co	23	18	70.9d
^{56}Co	35	3.7	77.1d
^{54}Mn	50	13	312d
^{48}V	72	5.5	16.0d
^{46}Sc	85	4.7	83.8d

*Routti(1974) [3] , **Asano et al.(1983) [4] ,

considered to contribute somewhat the ^{55}Fe formation at the positions close to the beam line. This disagreement can be explained in terms of coexisting high energy protons in the measured fluxes and the difference of the excitation functions of the (n,2n) and (p,pn) reactions. ^{55}Fe and ^{63}Ni survive for long time after beam-off and emit no γ-ray. Therefore no radiation level does not necessarily mean that all radioactivity had decayed out. From a viewpoint of radiation safety, it is very important to evaluate their radioactivities prior to an actual handling of radioactive accelerator components. The present study indicated that a rough estimation of ^{55}Fe and ^{63}Ni activities is possible by measuring the neutron fluxes by a set of Au foils and Cu threshold detectors.

Table 2. ^{63}Ni radioactivity in SUS samples

	n_{th} ($cm^{-2} s^{-1}$)	^{63}Ni(Cal.) Bq/g(SUS)	^{63}Ni(Exp.) Bq/g(SUS)
Sample-1	2.2×10^{6}	2.0 ± 0.6	3.7
Sample-2	2.0×10^{6}	2.2 ± 0.7	3.7
Sample-3	3.0×10^{6}	2.7 ± 0.7	2.1
Sample-4	5.2×10^{5}	3.2 ± 1.1	1.4
Sample-5	4.0×10^{5}	2.0 ± 0.6	2.7

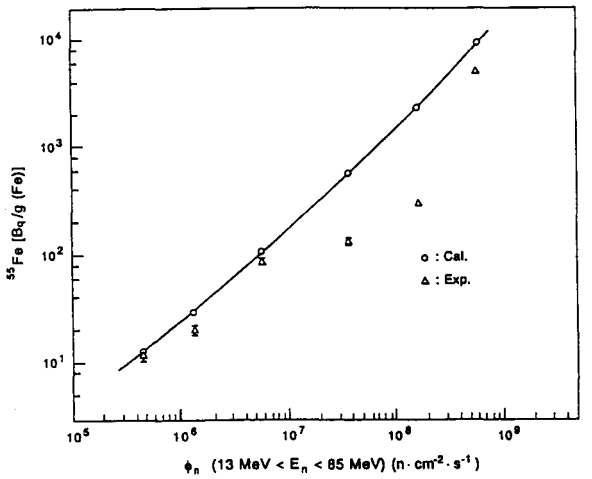

Fig.2 Relation between ^{55} Fe activity [Bq/g(Fe)] and neutron fluxes (Φ n;13MeV < En < 85MeV,n/cm ^{2} /s)

4,Acknowledgement

The authors wish to express their thanks to Prof.M.Takasaki,Dr.K.H.Tanaka and Mrs Y.Kato and H.Ishii for the experimental arrangements and operation of the proton beam-line. We would also like to thank Mr Takahara for the ^{55}Fe and ^{63}Ni measurement and maintenance of the liquid scintillation counter system.

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INTERCOMPARISON OF PERSONAL DOSIMETERS USED IN U.S. DEPARTMENT OF ENERGY ACCELERATOR FACILITIES

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An intercomparison of the dose equivalent response of personal dosimeters in use at U.S. Department of Energy (DOE) accelerator facilities took place at the European Laboratory for Particle Physics (CERN). This is the third such intercomparison sponsored by the DOE. The two previous intercomparisons were performed in a U.S. laboratory using a source of high energy neutrons. In this intercomparison dosimeters were exposed at CERN in two shielded positions relative to a production target that was hit by a positive hadron beam of 205 GeV/c. The neutron energy spectra present at these two locations had been calculated earlier using Monte Carlo methods and agreed well with those that had been determined experimentally using Bonner sphere spectrometer systems. The dose equivalents at these two positions were measured by CERN personnel using a tissue equivalent proportional counter system. For exposure, the dosimeters were mounted onto spherical phantoms made of polyethylene at several angular positions. The DOE dosimeters were mailed to CERN and following exposure returned for readout. The results of this intercomparison are relatively consistent with the two others previously performed in the U.S. The relative dose equivalent responses of neutron dosimeter types such as: albedo, nuclear emulsion and track etch plastics were found to have variations relative to the mean value responses of up to a factor of three. The possible causes for such large variations will be discussed.

INTRODUCTION

The U.S. Department of Energy (DOE) currently administers 12 laboratories that perform high-energy accelerator-based research. The accelerators at these laboratories produce radiations having energies ranging from a few MeV to the TeV range. The high-energy neutrons produced by these accelerators present a difficult radiation protection problem, because they can penetrate large thicknesses of shielding (1,2).

The DOE operates a program to accredit suppliers of personnel dosimetry for DOE facilities. This program is entitled the Department of Energy Laboratory Accreditation Program (DOELAP)(3). The two neutron sources used for DOELAP proficiency testing are unmoderated and D_2O-moderated ^{252}Cf. The unmoderated ^{252}Cf source has a mean energy of about 2.3 MeV with a high-energy tail extending to approximately 10 MeV. However, this source is not appropriate for testing personal dosimeters that are expected to measure dose equivalents produced by neutrons with energies of several hundred MeV.

The intercomparison of dose equivalents determined with personal dosimeters used at DOE high-energy accelerator facilities was intended to show the degree of consistency in high-energy neutron dosimetry at the participating laboratories. The high-energy neutron fields generated at CERN provided a more realistic approximation to the actual fields encountered at DOE high-energy accelerator facilities than could be provided by the two neutron sources used in the DOELAP program.

MATERIALS AND METHODS

The neutron fields used for this intercomparison were generated at CERN by bombarding a copper target with a beam of positive hadrons (protons and pions) with a momentum of 205 GeV/c. The secondary radiation produced by this interaction at 90° passes through shields of either concrete or iron (4). The radiation fields outside these shields contains a large portion of high-energy neutrons that can be used to irradiate personal dosimeters or area survey meters. Determinations of the neutron energy spectra for the two experimental conditions were carried out using Monte Carlo methods and Bonner sphere spectrometer systems (4). Good agreement was found between these two methods.

Personal dosimeters from participating DOE facilities were mailed to CERN from the Pacific Northwest National Laboratory. Appropriate numbers of control dosimeters were not irradiated, so that the background and transit doses could be evaluated. At CERN, the dosimeters were irradiated on two types of plastic phantoms.

Calibration exposures were performed using a rectangular solid phantom measuring 40 x 40 x 15-cm and an irradiation distance of 2 m from a ^{238}Pu-Be neutron source. For exposures at the accelerator experimental positions, dosimeters were mounted on spherical polyethylene phantoms at several angular positions. Following exposures at CERN, the dosimeters were mailed back to the participants for readout.

RESULTS AND DISCUSSION

Dose equivalents determined from the readings of the personal dosimeters used in this study were compared to the mean of all values determined by all of the participants. The dose equivalents determined by the participants were also compared to the values determined by CERN. The ratios of average readings to the mean for the participants are shown in Figure 1, for the concrete shielding configuration and 2, for the iron shielding configuration.

It can be seen from this plot that the extremes of average readings for dosimeters used by different participants vary from the mean by up to a factor of 3. The variations are in both directions and do not appear to correlate to dosimeter type or participant. The results for the concrete shielding configuration seem to be somewhat more consistent.

The dosimeter types used in this study had different dose equivalent responses as a function of neutron energy. In addition, the dosimeters were calibrated using different reference neutron sources. Therefore, it is not surprising that their responses to this high-energy neutron field varied by large amounts. In order to determine whether the inherent dose equivalent response of the dosimeters as a function of energy, the reference calibration source or other uncertainties were responsible for these differences, it would be necessary to design a specific experiment to test the effect of each influence quantity. It is expected that future intercomparison exercises will include such tests.

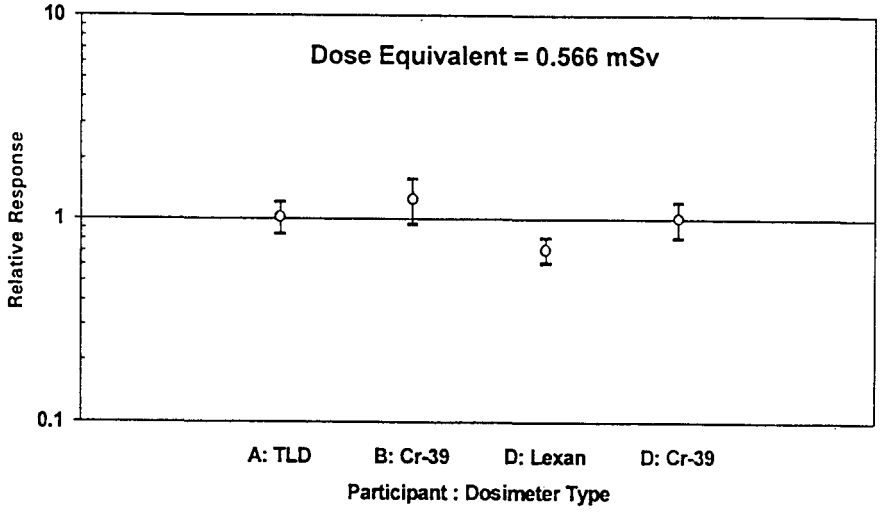

Figure 1. Dose equivalent responses relative to the mean for participants A-D in concrete shielding configuration.

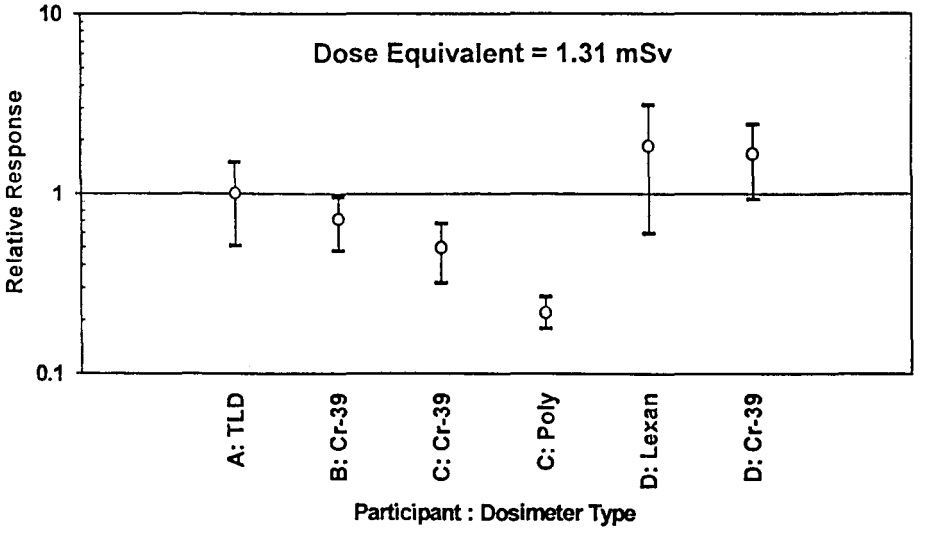

Figure 2. Dose equivalent responses relative to the mean for participants A, B, D in iron shielding configuration.

ACKNOWLEDGMENTS

We wish to thank our colleagues at CERN for providing beam time for this study and for their expertise in carrying out the exposures. This work was supported in part by the Health Physics Program Division of the U.S. Department of Energy. The Pacific Northwest National Laboratory is operated for the US Department of Energy by Battelle Memorial Institute under Contract DE-AC06-76RLO 1830.

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RADIATION MONITORING SYSTEM FOR ELECTRON ACCELERATORS AT ATOMINSTITUTE VIENNA

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INTRODUCTION

The Atominstitute Vienna has two Van de Graaff accelerators installed, with a maximum voltage of 2 MV and 400 kV respectively. Both accelerators are used only in electron mode, therefore the affect of X-rays produced by the electron beam on adjacent rooms and the operators area has to be monitored and limited. This is of great importance all the more than both accelerators are installed in rather normal laboratory rooms in the basement of the building. There is no special shelter as it is usual for such radiation facilities. To determine the maximum permissible beam current and/or to prescribe restricions in adjecent rooms hitherto it was necessery to make tedious measurements of the dose rate in areas affected by radiation from the accelerator. Whereas the dose rate at the workplace of the operator and the experimental area was monitored continuos, there was no possibility to do this for the neighbouring rooms, especially above the accelerator. It was in the responsibility of the operator to operate the accelerator within prescribed limits. For the 2 MV accelerator, when operated with external beam, the dose rate due to scattered electrons had to bee considered too.

As the accelerator area must not been accessed during operation, an interlock system with redundancy 2 and video monitoring of the area is provided. This is not described in detail here.

By official order it was neccessary to update the complete monitoring system for both accelerators. This system is nearly identical for both accelerators (no electron doserate to consider for 400 KV because of distance) and is described in brief for the 2 MV accelerator. In a second step of the project it is planned to use the acquired parameters of the accelerator system for automatic operation of the accelerator.

REQUIREMENTS and DRAFT

The absolute dose rate limit by official order was fixed to 25 μSv/h for all locations in the vicinity of the accelerator area. The first step was to find out the critical locations to decide how many detectors are required and where they have to be positioned.

In the operator area the maximum dose rate occurs at the barrier between accelerator area and operator area. This position requires 3 detectors. One is a filtered GM-detector for the X-ray doserate, the others are identical proportional ß-detectors with thin entrance windows, but one of them with a ß-absorber of perspex and therefor sensitive only for X-rays. The difference of the proportional detectors is a good measure for the electron dose rate. By comparison of the three detectors malfunctions can be detected. The reading of the proportional detector should not be lower then that of the unfiltered one. The reading of the filtered one should be equal to that of the GM-detector. A function test by natural radiation is self evident.

The maximum dose rate in adjacent rooms is found above the accelerator, at a position depending on the experimental arrangement. Extensive investigations for different cases showed that the position on the ceiling of the accelerator area exactly above the standard position of the X-ray target (gold) is a sufficient worst case approximation for all arrangements used and all electron energies from 0.5 to 2.2 MeV. As there is no practical possibility to locate detectors on the floor of the room above (this is the experimental area of the research reactor Triga Mark II) it was decided to mount a GM-detector at this position on the ceiling. To get a measure for the dose rate on the floor above the GM-detector is mounted inside a filter of lead and aluminium which simulates the absorption of the ceiling. This method would work perfect for a concrete filter but lack of space prevented this. Therefor a correction with an experimental determined function depending on the voltage of the accelerator is necessary.

Beside the accelerator voltage it is neccessary at least for plausibility tests to acquire the beam current. For external beam this may be difficult or impossible in a direct way. Acquiring all partial currents in the accelerator allows by balance-sheet of this currents to get the beam current without direct measurement. Considering the planned second step of the project it is useful to acquire all data which are normally displayed on the control panel and all logical states of switches and relays. This allows the simulation of the control panel on the computer screen.

The data acquisition applays therefore to several analog and digital signals as inputs. The only output at present is a failsafe logical output to shut down the accelerator. When acquiring data one has to keep in mind the possibility of transient signals everywhere in the accelerator wiring. They occure during voltage break downs in the accelerator tube or tank. Isolating amplifiers and isolating logical modules are therefor a must for each signal line. Data acquisition and evaluation is done on an intelligent board independent of the host computer which is free for other jobs such as operating the accelerator as planned for the future. Beside the main task of the program that is monitoring the radioation levels, giving warnings or shutting down the accelerator there are also a lot of routines watching the technical status of the accelerator and giving hints to the technical service personal.

OPERATION

Switching on the main power switch of the accelerator powers the computer. When ready the program ACMOS (**AC**celerator **MO**nitoring **S**ystem)is loaded from the disk and starts with a selftest. After a detector test with background radiation a login screen is displayed and a floppy disk is requested for the logfile and a log file on the disk is created and/or opened if it exists already for this day. Each day has its logfile which is stored on the disk. When starting the system another day the logfile is transfered to floppy disk. The login requires a password which is given by the master user which decides also the different permissions connected to this password (external beam, warning light in the accelerator area off, part of the interlock system off for service purposes, etc.). Only the master user can alter the warning and shut off levels, the permissions and several calibration factors. When a key switch on the operating panel is activated the power up procedure of the accelerator is continued via the computer and the actual status is stored in the log file and also printed. The computer screen shows all the acquired data after some averaging. The actual dose rate for display is measured in time preset mode with sliding averaging, whereas for limit evaluation and for the logfile this is done in preset count mode to speed up the response. Each time any of the acquired data differs significant from that of the last entry in the log file a new actual data set of all operating parameters and acquired data is added to the log file. If necessary this entry contains also messages for the technical service personal or the reason for shut down if this happened. This messages are always displayed on the screen and printed. Depending on the relevance some messages on the screen are accompanied by an acoustic signal and some messages have to be quit by the operator.

Shut down can be initiated by the operator, by a limit condition or if a fatal error in the system is detected. It causes another entry in the log file and closes it. When the shut down is caused by main power off the computer power is switched off with time delay to allow correct saving of the log file. The operation of the computer is monitored by a watchdog system which also causes a shutdown in the case of a hangup of the program.

The system is in use with only minor modifications since January 1996.

SOME ASPECTS OF IMPLICATION OF NEW ICRP RECOMMENDATIONS INTO NEUTRON RADIATION MONITORING AT IHEP

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INTRODUCTION

New Radiation Safety Standards developing by National Commission on Radiation Protection of Russian Federation (NCRP) will base on the ICRP Recommendations given in the ICRP Publication 60 [1]. It changes the approach to exposure reglamentation realized in actual Russian radiation safety standards and requires to estimate consequences of this reconsideration for radiation monitoring practice of the Institute for High Energy Physics (IHEP).

PROBLEMS OF TRANSITION TO ICRP-60 RECOMMENDATIONS

Present Radiation Safety Standards and the ICRP Recommendations (ICRP-60) differ in:
— the recommended quality factor dependence on linear energy transfer (LET);
— the basic and operational quantities for radiation risk evaluation for external exposure;
— occupational and public dose limitation.
The effective dose concept adopted by Russian NCRP instead of the critical organ concept requires from radiation safety services to use in their activity new ICRP operational quantities as well as new dose limits. When the ICRP-60 Recommendation will be introduced into neutron radiation monitoring practice, correspondent changes can be in:
— procedure of metrological maintenance;
— instruments used in the neutron radiation monitoring;
— organizational and limitational base of neutron radiation monitoring.

METROLOGICAL BASE OF THE IHEP RADIATION MONITORING

The metrological base of the IHEP radiation monitoring have been revised recently when the neutron reference fields [2] used for metrological maintenance of routine dosemeters have been certified [3] in terms of the neutron ambient dose equivalent rate. These neutron reference fields are:
— standard commercial calibration system UKPN-1M with ^{252}Cf and $^{239}Pu-Be$ radionuclide sources and a collimator made of polyethylene with 3% of ^{10}B;
— ^{252}Cf and $^{239}Pu-Be$ radionuclide sources in free air;
— ^{252}Cf in 30-cm diameter iron or polyethylene moderator.
The $Q(L)$ dependence recommended in ICRP Publication 21 [4] has been put on the basis of the certification. So, for IHEP radiation monitoring practice where new operational quantities are already used, implication of the ICRP-60 recommendations will be reduced to re-certification of the reference fields in terms of $H^*(10)$ based on new $Q(L)$ dependence.

ACCURACY OF AREA MONITORING

Possible changes in routine dose measurements accuracy concerned with implication of the ICRP-60 recommendations have been estimated for routine instruments most widely used in IHEP area monitoring such as rhodium monitors (RM) and passive monitors (PMS) usually calibrated in $^{239}Pu-Be$ field. RM is a counter with rhodium foil converter in polyethylene moderator of 10-inch diameter and is usually used in controlled areas. PMS is a $^{6}LiF - {}^{7}LiF$ pair in polyethylene moderator of the same diameter and is used in supervised areas.

The systematic errors of ambient dose equivalent measurements have been calculated for RM and PMS in the IHEP radiation fields where neutron spectra have been measured [2,5,6]. The energy dependences of ambient dose equivalent factor $h^*(10)$ of [7] based on the $Q(L)$ dependence of ICRP-21 and of [8], where $h^*(10)$ has been recalculated on base of the $Q(L)$ dependence of ICRP-60, are used for calculating H_0 — neutron ambient dose equivalent in a radiation field. The energy dependences of detector response to neutrons, $R(E)$, for RM and PMS, under the condition of their calibration in the $^{239}Pu-Be$ reference field are used for calculating H_{RM}, H_{PMS} — responses of the dosemeters in terms of ambient dose equivalent for both $Q(L)$ dependences.

Table 1. Mean ratios of H_{RM}/H_0, H_{PMS}/H_0 , and H_0^{60}/H_0^{21} calculated for three groups involving most available neutron spectra of IHEP radiation fields.

Ratio	$Q(L)$ dependence	Pu-Be	Behind upper and side shielding at exp. hall [5]	Behind side shielding and far from it , [6]	Reference fields on base of ^{252}Cf [2]
H_{RM}/H_0	ICRP-21	1	0.75	1.26	1.21
	ICRP-60	1	0.79	1.17	1.13
H_{PMS}/H_0	ICRP-21	1	0.72	1.11	1.15
	ICRP-60	1	0.75	1.04	1.04
H_0^{60}/H_0^{21}		1.26	1.18±0.05	1.34±0.04	1.38±0.06

The RM and PMS measurement systematic error is given by deviation of ratio H_{RM}/H_0 or H_{PMS}/H_0 from 1. Mean values of this ratio for three groups of neutron spectra are presented in Table 1. One can see that systematic error will reduce with transition to ICRP-60 recommendations in case of considered dosemeters and neutron spectra.

The difference in absolute values of ambient dose equivalents of ICRP-21 and ICRP-60 in IHEP workplace fields can be also found in Table 1 where the ratios H^{60}/H^{21} of these values calculated for the three neutron spectra groups are given. Transition to ICRP-60 will lead to dose increase up to 1.4 times while the neutron fluence remains the same.

APPLICABILITY OF EXISTING RADIATION MONITORING INSTRUMENTS

ICRP-60 recommends the occupational dose limit to be equal 100 mSv per 5-year period. The annual occupational dose limit of 20 mSv will be in this case in 2.5 times lower than the former one. It can lead to correspondent reduction of working range of dosemeters used in radiation monitoring and can result in impossibility of further application of some dosemeters. When transit to ICRP-60 recommendations, there are two factors working in the same direction: 1) dose limit changes and 2) the increase of ambient dose equivalent up to 40 percents mentioned above. The administrative control dose levels established in IHEP are recalculated in Table 2 for new dose limits in simplest way, i.e. divided by 2.5 (except supervised area where ICRP-60 recommended annual dose level is directly applied).

In Table 2 we have estimated the possibility of application of available monitoring instruments in case of adopted ICRP-60 recommendations by comparison of minimal doses measured by the instruments with new levels, so Table 2 contains ratios of minimal measurable doses to new levels. The values of minimal measurable dose have been recalculated by multiplying by a factor of 1.4, reflecting with some conservatism the fact of the ambient dose equivalent factor increase after transition to ICRP-60 recommendations. According to ICRP-35 Recommendations [9], an instrument could be still used in radiation monitoring if its minimal measurable dose less than new value of control dose level in 10 times and over.

As it is seen from Table 2, all the instruments will preserve their applicability in case of simple decrease of dose levels as recommended by ICRP-60.

PERSONNEL MONITORING

Brief analysis of personal exposure on base of personnel monitoring data for 1990–1993 years has shown that annual personnel doses greater than 20 mSv had occured in approximately 1% cases (10–12 cases per year instead of about one case for 50-mSv limit), while the mean annual personnel dose was about 3 mSv. Since new ICRP-60 Recommendations allows to exceed 20-mSv annual limit in case when mean dose over a 5-year period does not exceed 20 mSv, such low mean annual doses and overexposure frequency allow us to meet to new requirements without considerable changes in personnel radiation monitoring procedure. Reconsideration of administrative classification of workplaces seems to be enough to optimize personal exposure.

Table 2. Ratio of minimal measurable dose to established control dose level recalculated according to ICRP-60 requirements on dose limits, for basic IHEP dosimetric instruments

Established for:	Area 1	Area 2	Supervised	Individual
Control level, H_{est}	91 μSv/h	17 μSv/h	1 mSv/year	2.8 mSv/run
Dosemeter	H_{min}/H_{est}			
Active				
RM	0.0046	0.025	–	–
DPN *	0.0018	0.0094	–	–
^3He-ion.chamb.*	0.014	0.076	–	–
Passive				
PMS	–	–	0.014	–
MK–20 nucl.film	–	–	–	0.07–0.014
TLD IKS ▽	–	–	–	0.05
TLD6011,7011 ◇	–	–	–	0.022–0.07

* ^{10}B - covered counter in polyethylene moderator of 10" diameter, used in radiation monitoring of injection channel

* ^3He - filled ionization chamber in polyethylene moderator of 10" diameter, used in investigational measurements

▽ alumophosphate TL-dosemeter for individual photon dose monitoring

◇ LiF TL-pairs for individual neutron dose monitoring (production of NPO "Praktika", Moscow State University, Russia)

CONCLUSIONS

The transition to ICRP-60 recommendations will not involve any changes in instruments used in the IHEP routine neutron radiation monitoring. Modification of the IHEP metrological base will be not nesessive too, but we shall need in new certification of the instruments used for metrological maintenance.

The decrease of the annual dose limits will require optimization in the monitoring instruments application at some supervised areas and new classification of workplaces in controlled areas.

It has been shown, for most widely used in IHEP neutron radiation monitoring dosemeters such as RM and PMS, that transition to ICRP-60 recommendations concerned with operational quantities will not increase systematic error of the IHEP neutron dose measurements if the calibration procedure remains the same. The mean absolute values of measured doses will increase on 18% in radiation fields behind the upper and the side accelerator shielding, and on 30–40% — in other available IHEP radiation fields.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19, 1996
Vienna, Austria

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Receipt
Author 20682
Acceptance
Mini-Presentation

FORM FOR SUBMISSION OF ABSTRACTS
(Instruction for preparation on reverse)

PAPER TITLE

RADIATION MONITORS FOR A REAL-TIME RADIATION MONITORING ON UNK.

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MAJOR SCIENTIFIC TOPIC NUMBER 5.5

ABSTRACT

A brief description and characteristics of some radiation monitors designed in IHEP for a real-time radiation monitoring system for UNK project are presented. The radiation monitors output the information about radiations on coaxial cables using 16-bits digital signals. On these cables the power supply of radiation monitors is provided. A transmission of signals on distance up to 1.5 km is possible. Most attention has been concentrated on informative reliability. Monitors make the autodiagnostics of some their own parameters and the results of the autodiagnostics are passed in signals.Data and experience received in the tests of radiation monitors in radiation fields at experimental area of 70 GeV IHEP's accelerator and at start-up of a UNK injection channel are analyzed.

IRPA9
1996 International Congress on Radiation Protection
April 14-19, 1996
Vienna, Austria

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Abstract No 90603
Receipt
Author 20347
Acceptance
03. Aug. 1995 Mini-Presentation

FORM FOR SUBMISSION OF ABSTRACTS
(Instruction for preparation on reverse)

PAPER TITLE

IHEP reference fields in quality assurance system of radiation control at charged particle accelerators

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ABSTRACT

The set of neutron reference fields with the mean energies from thermal up to 4 Mev were created at IHEP on the base of the radionuclides sources (Cf-252, PuBe) and special systems for neutron spectra degradation. These reference fields are a system of common usage and research directions in framework of Metrological Council on Charged Particle Accelerators (MCCPA). MCCPA is consist of authorities from IHEP (Protvino), JINR (Dubna), INP(Gatchina), Mendeleev's Metrological Institute (S.Petersburg).

IHEP reference fields:

- are supplied by instruments and methods of sertification, that include Russian State Standarts;
- serve for calibration of measurement instruments for radiation control systems at the charged particles accelerators;
- are used for intercomparisons of various measurement instruments (dosemeters, ratemeters, spectrometers).

These works are carring out during 10 years in framework of MCCPA and provide the quality assurace of radiation control at charged particle accelerators in Russia.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

FORM FOR SUBMISSION OF ABSTRACTS
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Abstract No.	90182
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Author	
Acceptance	
Mini-Presentation	

PAPER TITLE Problems of Accelerator Dziven System radiation protection

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7) 5.2, 5.5.

ABSTRACT (See instructions overleaf)

Recently in many scientific centers conceptual study of Accelerator Driven System (ADS) various versions, i.e.subcritical reactors with a external source of neutrons as a proton accelerator and neutron-producing target is carried out. Conducted in SSC RF ITEP study shows about the serious problems, connected with provision of ADS radiation safety, in some aspects distinguished from similar problems for existing nuclear reactors. Following problems on maintenance of ADS radiation protection are formulated.

1.Calculation of a system shield of target-blanket in view of interaction with 4 type of radiation: a protons with energy 1-1,5 Gev, a neutrons of different energy from target and blanket, mesons and γ-radiation.

2. Study of changes of the radiating characteristics of a target under action of protons and neutron flux, formed in target and blanket, and processes of formation volatile of products of spallation-reactions in a target and their influence to a radiation conditions in the system of the proton transportation and on process of regeneration of a target.

3. Calculation of activation level and biological shield of high-power proton accelerator design (with a proton current up to 300mA), magnets of a system of a beam formation and transportation to a target in connection with possible losses of protons during acceleration at normal operation of a accelerator.

4. Study of emergency beam deposition consequences in designs and systems, specified in items 3, in particular, in a system of a beam reception at failure of magnets of a system of proton transportation and other.

Results of the experts ITEP investigations on separate aspects of ADS radiation safety and protection are given.

RADIATION PROTECTION AND THE NATURALLY OCCURRING RADIOACTIVE MATERIALS (NORM)

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There are many industries dealing with naturally occurring radioactive materials (NORM), some of them without knowing that their industrial processes and/or their regular wastes involve radioactivity. However, an increasing number of industries that produce NORM wastes are being sued, wherever there is a legal framework to do so. In particular, NORM wastes produced for a long time by the oil industry became foci of legal battles in the United States and elsewhere. The ripple effect of these judicial battles will influence the decision making processes of NORM wastes producing industries, mostly because of the costs incurred by remedial and preventive actions concerning NORM contamination. The regulation of NORM will occur sooner or later, and such actions may become mandatory. A foreseeable consequence of such regulation is a change in attitude concerning the sources and materials associated with NORM. Among those industries likely to be affected one can mention: niobium; rare earth processing; oil production; phosphate; uranium mining and milling; zircon; water treatment; and waste water treatment. The paper will briefly review data on exempt concentration activities, as suggested by the Basic Safety Standards based on realistic environmental and dosimetric models. These activity concentrations are compared with those found in a number of extractive industries, and may be used to establish derived limits from a pre-established dose limit.

INTRODUCTION

When one reads sections **1.4** (14) and (15) of the 1990 Recommendations of the International Commission on Radiological Protection, better known as ICRP Publication 60 (1), one can reasonably interpret that, according with the ICRP, the main goal of radiation protection is to protect human beings from unnecessary exposure by establishing standard of protection which takes into proper account the beneficial aspects of the practices leading to radiation exposure. In addition, there is concern with the environment to the extent that transfer of radionuclides through environmental compartments may affect the radiological protection of humans. In so being, the practices leading to releases of naturally occurring radioactive materias (NORM) into the environment can be easily justifiable, by and large, from the view point of taking into proper account the beneficial aspects of most such practices. However, because one needs to be concerned with the environment in which NORM releases may affect the radiological protection of humans, one cannot just ignore such releases.

A number of well respected international organizations — the Food and Agriculture Organization (FAO), the International Atomic Energy Agency (IAEA), the International Labour Organization (ILO), the Nuclear Energy Agency of the Organization for Economic Co-operation and Development (OECD/NEA), the Pan American Health Organization (PAHO) and the World Health Organization (WHO) — recently sponsored the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) (2). The BSS was approved by the IAEA's Board of Governors in 1994. The NORM issue was not specifically contemplated in the BSS, however, exempt activity concentrations and exempt activities of radionuclides were established based on the following general principles (2,3):

"(a) the *radiation risks* to individuals caused by the *exempted practice* or *source* be sufficiently low as to be of no regulatory concern;

(b) the collective radiological impact of the *exempted practice* or *source* be sufficiently low as not to warrant regulatory control under the prevailing circumstances; and

(c) the *exempted practices* and *sources* be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in (a) and (b)."

Table 1 lists the exempt activitiy concentrations established in the BSS for ^{222}Rn (plus ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po), ^{224}Ra (plus ^{220}Rn, ^{216}Po, ^{212}Pb, ^{208}Tl -36%-, ^{212}Po -64%), ^{226}Ra (plus ^{222}Rn, ^{218}Po, ^{214}Pb, ^{214}Bi, ^{214}Po, ^{210}Pb, ^{210}Po, ^{210}Pb), ^{228}Ra (plus ^{228}Ac), ^{228}Th (plus ^{224}Ra, ^{220}Rn, ^{216}Po, ^{212}Pb, ^{208}Tl - 36%-, ^{212}Po -64%), Th-nat (including ^{232}Th and progeny), ^{230}U (plus ^{226}Ra, ^{222}Rn, ^{218}Po, ^{214}Po), and U-nat (including the ^{238}U progeny), which are the most relevant radionuclides in the case of NORM releases, and activity concentrations found in some selected extractive industries..

Table 1. Exempt activity concentrations for the most relevant radionuclides (plus their progeny) in the case of NORM releases, as established in the Basic Safety Standards and activity concentrations found in selected extractive industries.

RADIONUCLIDE	ACTIVITY CONCENTRATION (kBq/kg)	
	Exempt (3)*	Extractive industries
Rn-222	10	
Ra-224	10	
Ra-226	10	2.7 (zircon sand - 4) — 658 (radioactive scale - 5)
Ra-228	10	368 (radioactive scale - 6)
Th-228	1	1.1 (zircon sand - 4) — 200 (radioactive scale - 6)
Th-natural	1	0.7 — 111(zircon sand - 4,7)
U-230	10	
U-natural	1	3 — 30 (zircon sand - 4,7)

*Reference numbers are within parenthesis.

EXTRACTIVE INDUSTRIES

As an anticipated consequence of the 1991 ICRP recommendations (1) and the adoption by national authorities of any one version of the rules suggested in the BSS (2) , many extractive industries will have to adjust their practices to more stringent limits to take care of NORM releases and the exposure of individuals in the workplace. The heavy fraction of rutile, ilmenite, zircon or monazite rich mineral sands usually have high ^{232}Th and ^{238}U, resulting not only in exposure of individuals in the workplace, but also in NORM releases.

The radioactive scales in oilfield tubulars can be considered unwanted byproducts of some operations in the petroleum industry. The concentration levels found in some radioactive scales can be much higher than the exempt activity concentrations listed in Table 1. As an example, ^{226}Ra and ^{228}Ra activity concentrations as high as 1 x 10^3 kBq/kg have been reported in the literature (5,6,8,9). Inhalation of ^{222}Rn and ^{220}Rn plus exposure to other alpha emitting natural radionuclides occur in

workplaces of the petroleum industries, as well as discharge into the environment of huge amounts of NORM wastes.

The phosphate fertilizer industry is another example of industry that produce NORM with enriched levels of ^{238}U and ^{226}Ra (10,11). Exposure of individuals in the workplace and environmental effects due to NORM occur also in the phosphate fertilizer industry.

There are many other industries that produce NORM wastes and expose individuals in the workplace. In some cases the exposed individuals and their managers are unaware of NORM . Radiation protection needs to be urgently implemented in such places. Because litigation tends sometimes to precede regulation strict liability claims concerning NORM have been on trial in the United States

CONCLUSIONS

1. The BSS does not contemplate the NORM issue directly, but can be used as a guidance for establishing rules to be adopted at national and international levels.
2. Individual exposure in the workplace and environmental effects resulting from NORM should be taken into proper account by the international radiation protection community.
3. The license of an extractive industry containing NORM should contain radiation protection and radioactive waste management plans to be approved based on well established rules adopted by the competent national authority.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

03. Aug. 1995

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PAPER TITLE RADIATION EXPOSURE AT TIG WELDING WITH THORIATED TUNGSTEN ELECTRODES

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MAJOR SCIENTIFIC TOPIC NUMBER 5.6. (see page 7)

ABSTRACT (See instructions overleaf)

In 1990 we got the order from the Bavarian Government to carry out a research project with the title "Man-Modified Materials Containing Natural Radionuclides". First of all, there had to be found exposure pathways, which are relevant in Bavaria, regarding to the radiation exposure of population and working persons. By means of a literature study and own calculations, we found relevant subjects and proposed to investigate them more intensively [1]. One of these topics was the handling with thoriated tungsten welding rods.

The properties of the tungsten rods are improved through the addition of radioactive thorium. In 1991 - 1993, as well as in 1995, we investigated the radiation exposure by handling with thoriated tungsten welding rods. We investigated the different exposure pathways and determined the spezific activity in dependence to the different types of welding rods. By carrying out surveys with the users, we determined the exposure pathways for the individual exposed persons: TIG-"hand-welders", TIG "machine-welders", labourers, other persons. We measured the activity concentration of the breathing air while welding, at grinding the electrodes and by staying in the rooms where usually it's welded. The size distribution of the aerosol-attached activity was determined with cascade impactors. The main emphasis was the comparison of the different sampling systems at the measuring of the activity concentration of the breathing air.

For the various exposed persons, at the extern irradiation with gamma- and beta-radiation and the inhalation of radioactive aerosoles, the radiation exposure was calculated by means of experimentelly determined data. And so, a maximum dose about 20 mSv per year was estimated.

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THORIUM EXPOSURE DURING WELDING AND GRINDING WITH THORIATED TUNGSTEN ELECTRODES

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ABSTRACT

Applying Tungsten Inert Gas (TIG) shielded welding process, often thoriated tungsten electrodes are used. Thorium is added to facilitate arc starting and to increase the arc's stability. If the electrode is touched down on the workpiece - not standing for a proper use but not always to avoid - a part of it will be vaporised. In the course of this, Thorium will be emitted. Only a sharp electrode will guaranty the wanted quality. Therefore the welder had to grind the electrode from time to time, also emitting Thorium in breathable form.
The presented contribution gives an overview about the results of numerous personal measurements of airborne activity concentration during grinding and welding.
Under proper use of thoriated electrodes at welding airborne activity (Th-232) of up to 2.6 mBq/m³ were found. This leads to an maximum annual body burden of 1.5 Bq. This is far below the limits of German regulations of non radiation workers. During grinding activity concentrations up to 0.5 Bq/m³ were measured. Even taking realistic annual working time into account limit may be exceeded. Radiation protection measures are to be enhanced, suggestions are made.

MATERIALS AND METHODS

Breath samplings had been taken in 19 plants under realistic conditions. Samplings during grinding were taken near breath openings of the patient (fig. 1/2). For the welding measurements a modified protective shield had been used, so that samplings could be taken beyond the shield in welders breath area (fig. 3). With the known volume rate and the sampling time the sampled volume could be calculated.

Fig. 1: Dust sampling during grinding

Fig. 2: Personal dust sampler

Fig. 3: Dust sampling during welding

ANALYSING FILTER SAMPLINGS

Expecting enough activity on that samplings, taken during grinding, these filter samplings were analysed by direct alpha spectrometry. The detection limit of this method (10 mBq Th-232 per Filter) was low enough therefore. These samplings were analysed by BIA.[1)]

The total activity on the welding filter samplings were far below the detection limit of alpha spectrometry. Therefore these samplings were analysed by gamma spectrometry after neutron activation. With this method a detection limit of about 0.1 mBq Th-232 per filter could be reached.

RESULTS

In table 1 the results of welding measurements are presented. The mean value (without measurement no. 1) of activity concentration is about 0.5 mBq/m³. The result of measurement no. 1 does not fit the other values. In this case, the welder continual touched down the electrode on the workpiece not in proper use.

The average of yearly inhaled activity by welding - based on realistic working time in the involved plants - is nearly 500 mBq/y.

measure no.	working time per year [h]	volume (m³)	Th-232 activity per filter (mBq)	activity concentration mBq/m³	yearly inhaled activity (Bq/a)
1	1760	0,945	22,0	23,3	49,2
2	2000	0,63	< 0,28	< 0,44	< 1,1
3	300	1,26	0,24	0,2	0,07
4	960	1,26	0,24	0,2	0,2
5	880	1,26	0,12	0,1	0,1
6	2000	1,26	< 0,13	< 0,1	< 0,24
7	1440	0,882	0,24	0,3	0,5
8	1320	0,945	0,33	0,35	0,6
9	840	1,103	0,49	0,44	0,4
10	1020	0,42	0,49	1,2	1,5
11	480	1,103	< 0,19	< 0,17	< 0,1
12	1200	1,26	< 0,21	< 0,17	< 0,3
13	480	1,26	< 0,21	< 0,17	< 0,1
14	480	1,05	2,72	2,6	1,5
15	1760	1,05	< 0,23	< 0,22	< 0,5
16	30	1,26	0,23	0,18	0,01
17	960	1,26	1,18	0,94	1,1
18	1740	1,26	< 0,21	< 0,17	< 0,4
19	1920	1,211	< 0,24	< 0,2	< 0,5

Tab. 1: Results of measurements during welding with thoriated tungsten electrodes

Table 2 shows the preliminary results - for the influence of some parameters the discussion is still going on - of the measurements during grinding the electrodes. The alpha spectrometry analyses shows, that the Th-230 activity of the used electrodes was clear below 20 % of the Th-232 activity.

In most cases no technical ventilation is used on the wheel stand. The mean value of inhaled activity per single grinding is 4.5 mBq (measurement no. 3 and no. 19 omitted).

[1)] H. Siekmann, D. Schwaß; Berufsgenossenschaftliches Institut für Arbeitssicherheit, St. Augustin, Germany

measure no.	number of yearly grindings	volume (m^3)	inhaled Th-232-activity per grinding (mBq)	activity concentration (mBq/m^3)	yearly inhaled activity (Bq/a)
1	7820	0.21	3.86	177	30.18
3	900	0.21	0.05	2.4	0.05
5	320	0.21	0.87	40	0.28
12	1680	0.21	2.34	109	3.93
15	700	0.21	15.04	514	10.53
16	50	0.21	2.62	153	0.13
17	320	0.21	6.00	275	1.92
18	580	0.21	0.55	69	0.32
19	45000	0.11	0.06	1.8	2.70

Tab. 2: Selected results of measurements during grinding of thoriated tungsten electrodes, no. 3 and no. 19 were sampled with technical ventilation in use, all others without ventilation

DISCUSSION

Welding: While recently a possible hazard through inhaled Thorium during welding is reported [1], this mesurements seem to confirm with other literature [2,3] which show Thorium exposure during welding well below of the anual limit of intake. The estimated maximum value of yearly inhaled activity (1.5 Bq/y) is below the German limit for non radiation workers (6 Bq/y). Of course a longer yearly welding time could lead to an exposure reaching the limit. Improper use of the electrodes (measurement no. 1) could exceed the limit. So the ALARA principle should taken into account and the possible incorporation should be reduced through technical ventilation during welding.

Grinding: Two of the grinding results are above the limit. With the determined average value of inhaled activity per grinding (4.5 mBq) more than 1350 grindings per year - corresponding to 6 grindings per day - could lead to an intake above the limit. Especially if one person grinds the electrodes for a number of welders (case no. 19 e. g.) this could be a problem if no technical ventilation is used.

For grinding thoriated tungsten electrodes the use of technical ventilation on the wheel stand is therefore always recommended. When cleaning the waste box of the wheel stand, breath protective equipment should be used.

ACKNOWLEDGEMENTS

The authors would like to thank the involved industrial injuries insurance institutes, the Norddeutsche Metall BG and the Maschinenbau und Metall BG for additional samplings. They also are grateful to the BIA for alpha spectrometry measurements of the grinding samplings.

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A TWO YEAR STUDY OF NORM LEVELS IN THE FACILITIES OF A MAJOR MALAYSIAN OIL AND GAS EXPLORATION AND PRODUCTION COMPANY

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ABSTRACT

Four comprehensive surveys of offshore and onshore facilities of a major Malaysian Oil and Gas E and P Company have been completed in the two year period since March 1993. Data include measurements of external gamma doserate, surface contamination, Rn and Tn progeny levels, particulate radioactivity in ambient air and radium in liquid effluent and sludge. Monitored quantities have yielded values which are for the most part in accord with those of undisturbed environments although limited occurrence of elevation of doserate, and of radium concentration in sludge have been observed. In the latter instance measured concentrations of ^{226}Ra and ^{228}Ac have been within values of less than 1Bq g^{-1}, being in general very much less than this. Comparison with reported oil and gas facility NORM levels in other parts of the world indicate present levels to be relatively low.

INTRODUCTION

The Malaysian Atomic Energy Licencing Board (AELB) have developed radiological monitoring guidelines in response to an established potential for inadvertent technological enhancement of Naturally Occurring Radioactive Material (NORM) in oil and gas facilities (1-12). AELB document LEM/TEK/30 (1) calls for establishment of oil and gas baseline data for a wide range of radiological parameters, also listing conservative levels above which AELB notification is required. These parameters and notification levels, comprise of external gamma doserate (0.5μSv h^{-1} inclusive of background), surface contamination (0.04 Bq cm^{-2} for alpha emissions, 0.04 Bq cm^{-2} for other emissions), Rn and Tn progeny levels (0.01WL and 0.02WL respectively), airborne ^{238}U, ^{232}Th, ^{226}Ra, ^{228}Ac and gross alpha and beta contaminants in dusty working environments (0.01 Bq m^{-3} for ^{238}U, 0.002 Bq m^{-3} for ^{232}Th, notification levels not specified for other parameters) and in addition the monitoring of total specific activity (TSA) of scale and sludge (0.1 Bq g^{-1} above prevailing background). As with regulatory authorities elsewhere (5) the AELB requires TSA to be quoted assuming secular equilibrium, viz TSA = (6 x specific activity of ^{226}Ra) + (8 x specific activity of ^{228}Ac), ^{226}Ra and ^{228}Ac characterising the presence of ^{226}Ra and ^{228}Ra progeny in the ^{238}U and ^{232}Th chains respectively. The associated gamma emissions of ^{226}Ra and ^{228}Ac are generally of sufficient prominence to facilitate their detection. Note that account is only taken of those progeny contributing significantly to the total activity following the elapse of several half-lives. Present results are from the accumulated data of an on-going comprehensive study of the facilities of a major Malaysian Oil and Gas E and P company. The results have been obtained over a period of two years (March 1993 to April 1995) and are associated with the routine operations of oil production.The study comprised of four separate surveys, each of three months duration, with two monitoring exercises per year, these being conducted in the second and fourth quarters. The even spacing of surveys provided for the elucidation of possible trends in recorded values of the parameters.

METHODS

The currently reported study encompassed 9 installations sited off the east coast of Peninsular Malaysia and 57 installations sited off the coasts of Sabah and Sarawak. Also included in the study were two Peninsular Malaysia onshore installations (one gas terminal and one supply base) and two East Malaysian onshore installations (one crude oil terminal and one supply base). None of the offshore platforms currently surveyed use injected seawater for artificial lift. This is relevant since incompatibility between the chemistry of injected seawater and formation water is known to promote deposition of NORM throughout the process line. The measuring equipment used in the conduct of these surveys comprised of: (i) field equipment which allowed *in situ* assessments and; (ii) laboratory-based facilities for gamma spectrometry and gross alpha and beta analyses. Table 1 lists the main items of equipment and the lower limits of detection typically achievable using standard practices for the levels which prevail.

RESULTS AND DISCUSSION

For many of the parameters a significant fraction of monitorings have been found to be within background. As an example, from a total of 3509 surface contamination measurements performed in areas where production fluids potentially come into contact with exterior surfaces (eg in vicinity of metering packages, launchers and receivers) less than 10 measurements were statistically above background (eight at the limit of detection, one at a value of between 0.04 - 0.05 Bq cm^{-2} and one at the level of 0.40 - 0.41 Bq cm^{-2}). Similar findings have been obtained for Rn and Tn progeny. For parameters with sufficient numbers of data recording above minimum detection levels the associated distributions display a strong positive skew best characterised by geometric means and geometric standard deviations. Particular examples are *in situ* measurements and TLD evaluations of external gross gamma doserate (Figures 1 and

2 respectively). The broader distribution of values observed for TLD evaluations was due to the positioning of TLD chips (3 chips per packet) at points which *in situ* measurements indicated to provide elevations of prevailing background (eg at the base of first stage separators). Elevation of above twice background were of highly limited occurrence. Figure 3 provides results for gross β levels measured in air, displaying the characteristic positive skew and indicating the majority of measurements to be below minimum detectable activities (11). Figure 4 shows the distribution of TSA in sludge, computed from measured values of ^{226}Ra and ^{228}Ac (each being less than 1Bq g^{-1}) as discussed above. A significant fraction of the samples of sludge indicated TSA's of below 1.0 Bq g^{-1} while more than 95% of the sludge samples showed levels in excess of the conservative AELB notification level of 0.1 Bq g^{-1}. Present results indicate considerable consistency with lower or typical values for routine operations in other parts of the world, including the North Sea, USA, Italy, Tunisia and the Congo (2,3,6), being much lower than the upper range values quoted for doserates, and for radium activities in scale, sludge and other precipitates. The majority of presently reported data are either consistent with prevailing levels of the undisturbed local environment or represent subtle elevations from this. No current evidence exists for the presence of trends in any of the monitored parameters. Current data provide a firm basis from which future changes might be identified. The results of this study add to a more extensive survey currently being conducted by this E and P Company.

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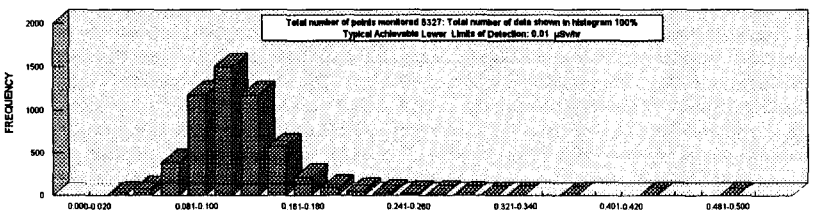
Table 1

Parameter	Equipment	Typical Lower Limits of Detection
1. Doserate	FAG FH40F2	0.01 μSv/hr
2. Gamma Counts(Scintillation)	Delta 3A + NaI probe	0.01 cps
3. Surface Contamination	ESP-2 + ZnS probe	0.03 Bq/cm^2
	Delta 3A + ZnS probe	
4. Rn/Tn Progeny	RDA-200	0.0005 WL
5. Contamination of Air Particulates	HpGe (Oxford)	^{226}Ra (0.1- 0.2 Bq/m^3)
	(30% Relative Efficiency)	^{228}Ac (0.3 Bq/m^3)
		^{214}Bi (0.01 Bq/m^3)
		^{212}Bi (0.01 Bq/m^3)
	LB 5500 (Oxford)	Gross α (0.003 Bq/m^3)
		Gross β (0.005 Bq/m^3)
6. Activity of Sludge/Scale	HpGe (Oxford)	^{226}Ra (0.01Bq/g)
	(30% Relative Efficiency)	^{228}Ac (0.01Bq/g)
	LB 5500 (Oxford)	Gross α (0.001 - 0.008 Bq/g)
		Gross β (0.002 - 0.010 Bq/g)
7. Activity of Liquid Samples	HpGe (Oxford)	^{226}Ra (0.01 Bq/l)
	(30% Relative Efficiency)	^{228}Ac (0.01 Bq/l)
	LB 5500 (Oxford)	Gross α (0.02 - 0.76 Bq/l)
		Gross β (0.03 - 1.00 Bq/l)
8. Activity of Wipe Test Samples	HpGe (Oxford)	^{226}Ra (0.001 - 0.003 Bq/cm^2)
	(30% Relative Efficiency)	^{228}Ac (0.0023 - 0.0026 Bq/cm^2)
	LB 5500 (Oxford)	Gross α (0.000026 - 0.000033 Bq/cm^2)
		Gross β (0.000046 - 0.000054 Bq/cm^2)

FIGURE: 1

GROSS EXTERNAL GAMMA DOSE RATE (IN SITU)
DATA TAKEN OVER THE PERIOD OF MARCH 1993 TO APRIL 1995



Typical onshore/offshore dose rate ranges in areas of undisturbed environment: Peninsular Malaysia 0.07 - 0.14 µSv/hr
Typical onshore/offshore dose rate ranges in areas of undisturbed environment: Sabah and Sarawak: 0.05 - 0.13 µSv/hr

FIGURE:2

GROSS EXTERNAL GAMMA DOSE RATE (TLD)
DATA TAKEN OVER THE PERIOD OF MARCH 1993 TO APRIL 1995

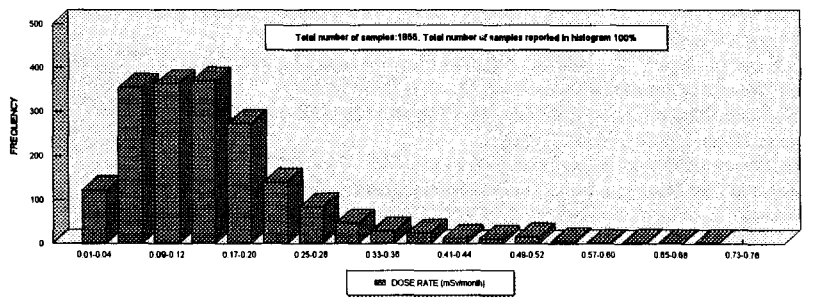


FIGURE:3

AIR SAMPLES
DATA OBTAINED OVER THE PERIOD OF MARCH 1993 TO APRIL 1995

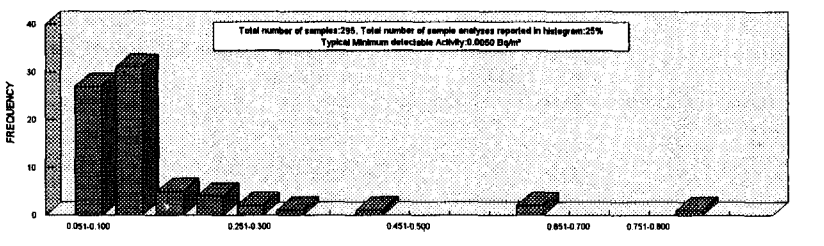
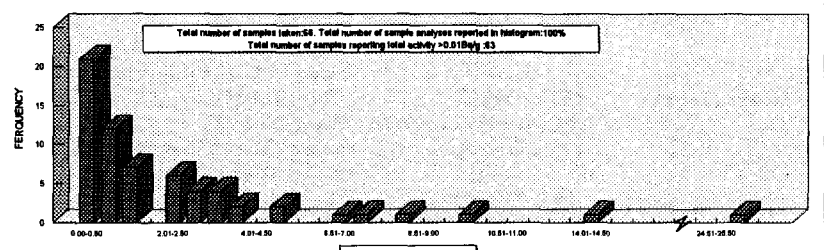


FIGURE:4

SLUDGE SAMPLES
DATA OBTAINED OVER THE PERIOD OF MARCH 1993 TO APRIL 1995



Typical total activity ranges of soil in surrounding areas of onshore facilities East coast of Peninsular Malaysia:0.69 - 1.56 Bq/g
Typical total activity ranges of soil in surrounding areas of onshore facilities of Sabah and Sarawak: 0.078 - 0.42 Bq/g

RADIATION PROTECTION IN THE OFFSHORE PETROLEUM INDUSTRY

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ABSTRACT

Radium and barium are normally dissolved in the formation water of oil reservoirs. After sea water, which contains sulfates, is injected to the reservoir to maintain production pressure radium can precipitate with barium as barium(radium)sulfate (1). This salt plates out on internal surfaces of pipes, valves and vessels and highly radioactive scale is formed. The specific activity of ^{226}Ra and ^{228}Ra in scale formed on some internal surfaces of an offshore oil production facility was measured as 250,000 Bq/kg and 300,000 Bq/kg respectively. About ten times lower specific activity of radium was measured in sludge. Naturally occurring radioactive materials in petroleum facilities are referred to as NORMs. To manage occupational radiation exposures contaminated pipes and vessels were visibly marked, all personnel on the facility attend regular training and a radiation protection monitoring program was implemented. Even though the external dose rate on external surfaces of some pipes and vessels contaminated with scale already reached 60 μSv/h it was possible to manage the maximum personal annual effective dose equivalent below 1 mSv. During shutdowns external dose rates inside contaminated vessels may reach more than 150 μSv/h and the potential alpha energy concentration of radon and thoron daughters in non-ventilated vessels may reach several WLs. Therefore, all entries of vessels contaminated by scale and protection of employees must be planned and breathing protection provided in order to keep personal doses at low levels. The paper summarized practical experience in assessment of personal doses and management of personal exposures which involve job planning and exposure scenario modelling.

INTRODUCTION

Specific activities of natural radionuclides in some NORMs may be extremely high and their build up increases with the depletion of oil in the reservoir. Therefore, the following exposure pathways are considered: a) External exposures due to radionuclides in pipes, valves and vessels containing radioactive scale, sand and sludge and external exposures inside vessels; b) Internal exposures to inhaled particles of resuspended scale; and c) Internal exposures to inhaled radon and thoron daughters inside vessels.

It early stages of NORMs build up operational occupational radiation exposures are below the public limit of 1 mSv per year. To estimate such a low average annual occupational effective dose equivalent requires to use a combination of experimental and modelling assessments methods which are similar to assessment methods used in the uranium industry.

EXTERNAL EXPOSURES

Occupational external exposures in the petroleum industry are mainly due to external gamma-rays of short-lived decay products of ^{228}Ra and ^{226}Ra which are contained in scale,

sludge and sand (Table 1). The exposure rate at surfaces of pipes and vessels, which internal surfaces are in contact with a mixture of oil and water or in contact with the produced water, were measured as high as 8 mR/h. Considering that NORMs are under normal operational conditions confined inside pipes, valves and at the bottom of vessels the external exposure rate decreases sharply with the distance from relatively small but intensive sources of radiation.

Table 1

Specific activities, a (kBq kg^{-1}), of natural radionuclides in samples of scale, sludge and sand from an off-shore oil producing facility

Sample description	Specific activity (kBq kg^{-1})					
	^{238}U	^{226}Ra	^{210}Pb	^{232}Th	^{228}Th	^{228}Ra
Scale	0.5	250	12	0.5	25	300
Sludge	-	25.0	-	-	2.6	29.5
Sand	-	8.9	0.06	-	2.28	9.60
Sand	-	0.75	-	-	0.20	0.85

Radioactive barium(radium)sulfate scale is a very brittle material. During shut downs when pipes and vessels are empty and the temperature drops scale cracks and peals from pipes surfaces. Subsequently, scale debris often cumulate in large quantities inside oil separators where the external dose rate was measured in excess of 150 μSv/h.

INTERNAL EXPOSURES

The most important radionuclides from internal dosimetry point of view, which are present in radioactive scale, sludge and sand in more significant activities, are ^{228}Th, ^{228}Ra, ^{226}Ra and ^{210}Pb. Internal exposures due to inhalation of airborne radioactive dust containing these radionuclides and due to inhalation of decay products of both radon isotopes had to be considered to cover potentially the most important occupational exposure scenarios. The effective dose equivalents per unit activity intake are presented for NORMs with several activity mean aerodynamic diameters (AMADs) in Table 2 (2). Considering that the specific radium activity in scale is about thirty times higher than the specific activity of material in Table 2 the effective dose equivalent of 1 mSv is equivalent to inhalation of approximately 300 mg of fine scale particles.

Table 2

The effective dose equivalent per unit activity intake, $h_{50,dust}$ (mSv Bq^{-1}), and amounts, m (g mSv^{-1}), of inhaled NORM dust with the specific activity of 12000 Bq kg^{-1} of ^{228}Ra, 1000 Bq kg^{-1} of ^{228}Th, 10000 Bq kg^{-1} of ^{226}Ra and 1500 Bq kg^{-1} of ^{210}Pb that give the effective dose equivalent of 1 mSv as a function of AMAD

AMAD (μm)	$h_{50,dust}$ (mSv Bq^{-1})	m (g mSv^{-1})
1	0.0041	10.0
5	0.0033	12.4
10	0.0031	13.2
20	0.0026	15.7

Internal exposures due to inhalation of radon and thoron daughters are equally important as exposures to inhaled airborne scale particles. Even though the radon emanation coefficient of scale was measured to be very low (3.5%) (3) the potential alpha energy concentrations of both radon and thoron daughters may reach in non ventilated vessels several WLs.

PERSONAL EXPOSURE MANAGEMENT

Routine operations on offshore petroleum production facilities include only a few vessel entries per year and therefore personnel are exposed more frequently to external radiation than to inhaled radionuclides. To manage external exposures at low levels was achieved through periodic training of all personnel in radiation protection. All pipes and vessels, which contain NORMs, were marked and all personnel were instructed to plan routine procedures so the occupancy time and the distance from marked components are as low as practicable. Thus, it was possible to keep annual external exposures below 1 mSv. Vessel entries are planned and special work procedures are adopted. During entries of vessels contaminated by scale all personnel use respiratory protection (supplied air is used in vessels contaminated with scale), disposable overalls, rubber boots and rubber gloves. The external exposures of personnel involved in vessel entries are measured by electronic pocket dosimeters and vessels are ventilated.

Vessel ventilation is more important in large tanks contaminated by sludge where it is difficult to use supplied air. Modelling was used to estimate the effect of ventilation on the potential alpha energy concentration of radon and thoron daughters inside large tanks. Provided the air in a tank is replaced by the outside air five times per hour the original potential alpha energy concentration of radon daughters and thoron daughters is decreased by a factor of 240 times and 45 times respectively. The results of the computer modelling were supported experimentally.

CONCLUSIONS

It is very important to be aware that NORMs may be present in production components of offshore petroleum production facilities as personnel who are unaware of their existence will receive high external and internal doses which could exceed the annual limit of 20 mSv.

The NORM detection programs should be enforced by national regulatory bodies so radiation protection monitoring programs could be implemented soon after NORMs are detected. Such monitoring programs ought to be periodically reviewed as NORMs build up faster at the end of the life of oil reservoirs.

Provided the radiation protection monitoring program is implemented it is possible to manage personal radiation exposures in the offshore petroleum industry so personal doses are at low levels.

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RADIATION SAFETY ASPECTS

IN THE OIL AND GAS PRODUCTION FACILITIES IN EGYPT

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ABSTRACT

The applications of radiation sources of various types in the oil and gas industry became an essential part in the prodution and expansion programmes in the production companies. Among these applications, oil well logging with gamma and neutron sources and tube weld radiography with gamma sources are well known.

On the other hand, naturally - occurring radioactive materials (NORM) have been detected in varying concentrations in hydrocarbon reservoirs in different areas of the world. Such materials may induce radioactive scales, sludges or gases in the production lines.

In this report, radiation safety guidelines are proposed to be implemented in the oil and gas production facilities in Egypt.

These guidelines are based mainly on the international safety standards such as the ICRP recommendations as well as the guidelines set by other countries of well recognized radiological health practices such as USA, UK and Europian Commnunity.

LEVELS OF RADIOACTIVITY IN GRIT BLASTING MATERIALS

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Introduction:

The problems of the build up of naturally occurring radioactive materials within the scales and silts accumulating in gas and oil production and processing plants are well known. (1) Whilst surveying for these materials it became apparent that very occasionally high survey readings were being produced which could not be associated with elevated levels in scales or silts in the plant being investigated. Further measurements established that the source of these elevated levels was in fact grit from recent cleaning operations which had not been properly cleaned away. It was therefore decided to look at the activity levels in grits more systematically to see if a real problem did or could exist.

Method:

Grits were obtained from a variety of commercial sources. In the UK. silica and sand are prohibited with regard to their use in blasting materials (ground quartz type rocks) Most industrialised countries have similar regulations UK (2), Belgium (3), Netherlands (4), France (5) and Germany (6)

Grits are therefore generally manufactured from slag from copper or steel processing, chilled iron particles or cast iron particles. The commercial name of some would however still indicate a direct mineral source. Unfortunately it was not always possible to obtain information as to which type a given sample was and chemical analysis results at this point are not available. Samples were subject to γ spectral analysis, generally in two centres to ensure reproducible results.

Results:

The results from the γ spectral analysis of the various samples are given in table 1.

Table 1. Activities in representative Grit Samples

Grit Number	Type of Grit	Ra-226 Bq/g	Ac-228 Bq/g	Cs-137 Bq/g	Am-241 Bq/g
1	Copper Slag	0.09 ± 0.02	0.01 ± 0.05	0.003 ± 0.001	0.003 ± 0.001
2	Cast Iron	less than 0.005	less than 0.005	less than 0.002	less than 0.002
3	Iron Slag	0.05 ± 0.02	less than 0.005	less than 0.002	less than 0.002
4		less than 0.03	0.05 ± 0.01	less than 0.003	less than 0.002
5		less than 0.05	0.10 ± 0.01	less than 0.002	less than 0.002

The samples which have been obtained from the slag arising from one of the refractory processes can all be seen to contain some of the naturally occurring nuclides. Only the activities of Radium 226 from the Uranium 238 series and Actinium 228 from the Thorium 232 series are recorded. Although measurements have not currently been made it is thought more than likely that these isotopes are in equilibrium with the respective parent radionuclides. The radionuclides lower down the series exhibited different levels of equilibrium. One sample can also be seen to contain very small amounts of both Cs-

137 and Am-241. This trapping of the natural and artificial radioactive cations by the slag is not surprising. Work carried out for other purposes shows that in practice slag is an exceedingly efficient method of fixing radionuclides and the results in table 2. show the ability of slag to trap the naturally occurring radionuclides present on scrap iron in radioactive barium sulphate scale. Over 95 % of the activity put into an electric arc furnace was trapped and immobilised into the slag produced (7). By varying the amount of slag produced it was possible to vary the activity per gramme indicating the an approximately uniform distribution was occurring.

Table 2. Uptake of Radium-226 and Actinium-228 (Radium-228) from scale contaminated iron melted in an induction furnace. From data provided by Clyde Shaw Ltd.

Melt Wt. kg	Input Ra-226 kBq	Activity Ac-228 kBq	Slag Wt. kg	Slag Ra-226 kBq	Activity Ac-228 kBq	Output Ra-226 kBq	Activity Ac-228 kBq
12,900	199	130	1310	0.15	0.11	197	138
9700	149	98	1820	0.10	0.05	182	91
8700	134	88	1320	0.12	0.03	158	40
5000	77	51	1620	0.08	0.03	130	49
4800	74	49	1540	0.08	0.03	123	46
6000	92	60	1700	0.08	0.03	136	51
6800	105	69	1500	0.08	0.03	120	45

Similar work carried out by Boynton and Lambley (8) and Gomer and Lambley (9) looking at the ability of slag to trap the radioactive cations on contaminated steel again showed that some of the cations present could be successfully moved into the slag leaving the now decontaminated steel behind. The technique removed virtually all the caesium isotopes into the slag, but the Cobalt-60 all went into the steel. From all the experiments, once the slag cooled the trapped cations were exceedingly stable and the leach rate was immeasurably small.

Discussion:

It would seem likely therefore that naturally occurring radionuclides present in the source ores or other materials present in the kiln are likely to be accumulated in the slag. Likewise many artificial radionuclides which deliberately or inadvertently find themselves in the furnace are also going to accumulate in the slag.

The activity levels at which the legislation in various countries classifies materials containing naturally occurring radioactivity as radioactive varies enormously. Some levels are given in table 3. As can be seen compared to the levels measured in the grit samples in most countries none of the grits would be considered as being radioactive. However in some the levels could be approached and in Germany the level for radium -226 would actually be exceeded by some of the grits measured.

Table 3. Brief Summary of some activity levels at which different countries consider material radioactive.

United Kingdom	0.37 Bq/g of Ra-226 (RSA 93) but under " Rare Earths" Exemption Order 14.8 Bq/g of Radium
Holland	100 Bq (total)/g
Germany	0.05 Bq/g for Ra-226
Malaysia	0.1 Bq/g (total)
Australia (South Australia)	35 Bq/g and 5 x 10^3 Bq (Ra-226)

The very use of the grits is going to produce an aerosol of fine powder. The people actually doing the blasting should be wearing sufficient protective equipment to ensure that the inhalation of the material is minimal. None of the levels measured approach the 10 Bq/g Radium-228 or 1 Bq/g Radium-226 levels the NRPB (10) recommend as operational level below which no one is likely to exceed a dose of above 1 mSv/y. Air monitoring measurements have not been made to confirm this.

Recommendations:

Our experience would indicate that non-slag type grits from smelting processes will not present any problems at all and can be used without any further tests being made. Copper or iron slag based grits may cause a problem depending on the national definition of what is classed as radioactive. Likewise mineral based grits may also cause a problem in some countries. If national limits are low then suitable analysis should be carried out on a representative sample of the grits prior to use. The levels of activity detected would indicate that the normal precautions taken during grit blasting operations are sufficient to ensure that no exposure limits for non-classified workers will ever be approached.

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RESULTS OF DECOMMISSIONING WORK UNDERTAKEN AT THE RADIOMETALLURGIY LABORATORY OF THE CEA CENTRE IN FONTENAY-AUX-ROSES

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Since 1990, CEA has been undertaking decommissioning work of the radiometallurgy laboratory for plutonium based fuels at the Fontenay-aux-Roses centre, which has been operational since 1963

The protection survey put in place on the Centre's Radiation Protection Department has enable dosimetric figures to be produced for this decommisioning operation, which generates high doses.

The collective dose of the worksite amounted to 1772 men.mSv, which makes one of the major CEA decomissioning worksites.

Doses received whilst work was in progress came from two main sources (picture 1):

- work undertaken within cells, which generated the highest dose rates
- nuclear waste stored in the premises adjacent to the cells. These doses are called "passive doses"

The dosimetric indicators used by the Radiation Protection Department are presented for each year. In particular, these are collective doses for the whole worksite, average dose and the dose received by the most exposed employee (picture 1 and 2)

One notice a significant increase in activity in 1993 and 1994 (increase in terms of work force, collective dose and even average dose)

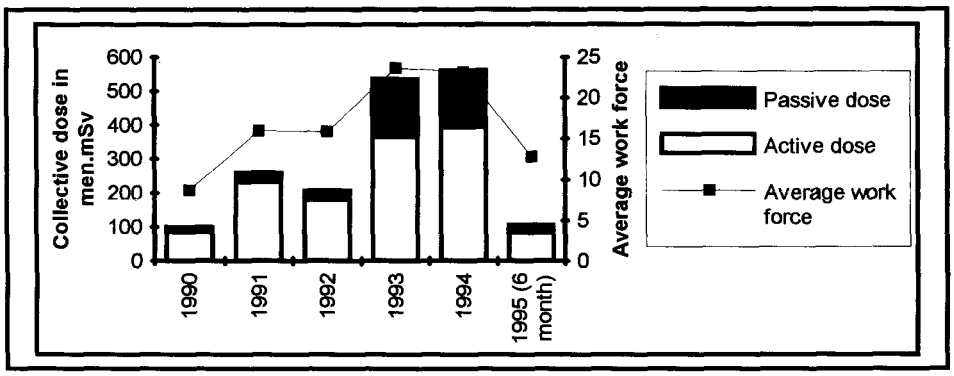
Besides the normal correlation between work force and overall dose, the increase in the average dose results from a high passive dose in the order of 30 μSv per day and per man, since irradiating waste couldn't be evacuated fast enough and was stored on the premises.

Picture 3 compares the average dose received per employee with that received if the passive dose for 1993 and 1994

A 60% reduction in the dose resulting from the use of remote control robots was assessed for an initially estimated dose of 300 men.mSv.

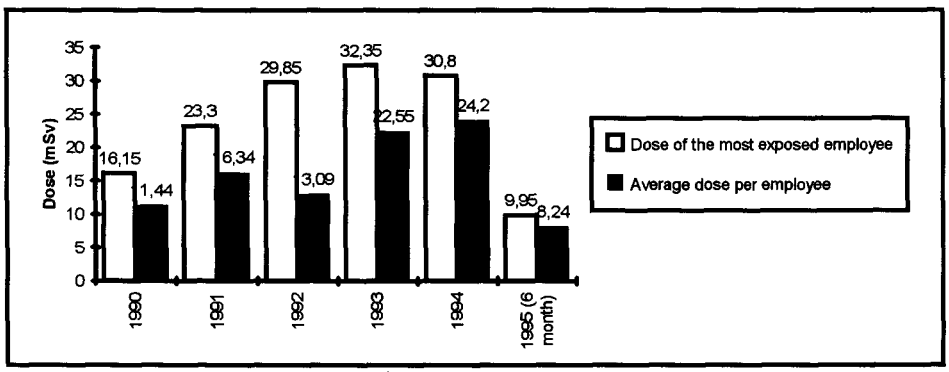
The expansion of ALARA initiative on such worksites has led to the use of remote-controlled equipment to decommission cells, automatic carriers to scrape floors and walls, cutting out internal cell protection panels by remote-control and γ-camera to make maps.

Evolution of the collective dose and the work force during work in progress



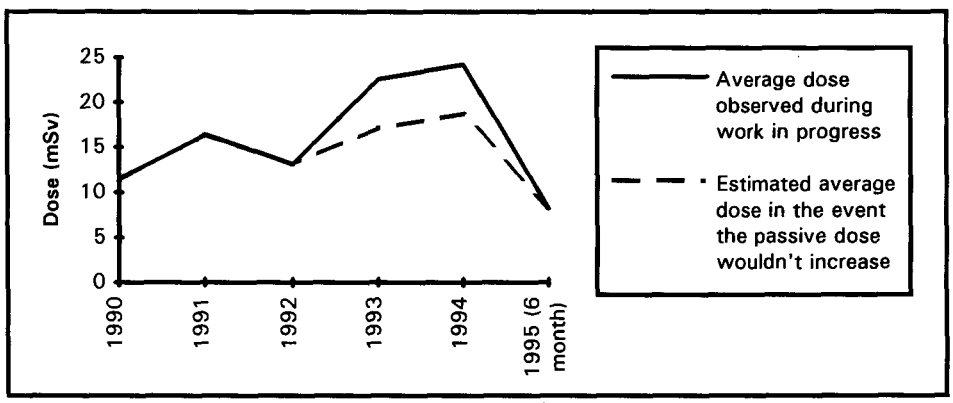
Picture 1

Dose of the most exposed employee and the average dose per employee



Picture 2

Effect on the average dose of the increase in passive dose between 1993 and 1994



Picture 3

CLASSIFICATION OF WORKERS USING OPEN/SEALED SOURCES OF RADIOACTIVE MATERIALS

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The finalisation of the European Directive on basic safety standards now means that member states of the EC must introduce legislation to implement the Directive. In the nuclear industry criteria for classification of areas could be related to the Annual Dose receive by workers in the industry.

In research laboratories, and hospital and higher education research laboratories the doses measured by workers are of the order of 2mSv per year. Therefore criteria (based on a different parameter) than dose received, needs to be considered.

The International Commission Radiological Protection[1] uses two designation of areas, controlled and supervised. Supervised areas require less administrative control but some control is necessary to ensure the doses outside the boundary are suitable for members of the general public. The ICRP in publication 68[2] recommends the use of dose coefficients e(50) which corresponds to the committed effective does E(50) resulting from an intake of 1Bq of the radionuclide.

For internal exposure it is recommended that the Annual Limit on intake should be based on a committed effective dose of 20mSv

so $$ALI = \frac{0.02}{e(50)}$$

Designation of Workers Using Open Sources

Most of the establishments above tend to designated the area and then classify the worker accordingly. Contamination is rarely present and some indication of the activity which could be handled in a supervised area would help the administrative requirements and help also to remove anomalies different establishments.

Previous papers[3] have indicated that of the total activity handled, allowing for some difference in volatilities, that of the total activity handled the fraction inhaled.

Hence allowing a factor of 3/10 for non classified workers.

Activity handled $= 10^4 \times 3/10 \text{ ALI} = 3 \times 10^3 \text{ ALI}$

$$= 3 \times 10^3 \frac{0.02}{e(50)}$$

$$= \frac{60}{e(50)}$$

where e(50) = dose coefficient

Using the dose coefficients given in ICRP8 to derive the ALI for inhalation. The table indicates the activity which may be handled, before the need for imposing greater administrative controls.

	Activity Handled Bq	e(50)	ALI Bq
$^{3}H_20$	3.3×10^{12}	1.8E-11	1.1×10^{9}
$^{3}H(OBT)$†	1.4×10^{12}	4.IE-11	4.8×10^{8}
$^{14}C_{dioxide}$	9.3×10^{12}	6.5E-12	3.1×10^{9}
^{32}P	7.5×10^{10}	8.0E-10	2.5×10^{7}
^{33}P	6.3×10^{11}	9.6E-11	2.1×10^{8}
^{35}S(vapour)	5.0×10^{11}	1.2E-10	1.7×10^{8}
^{125}I(vapour)	1.1×10^{10}	5.3E-09	3.8×10^{6}

* Dose due to the activity not absorbed through the skin is not included
† gaseous form

The administrative controls which can be relaxed include the use of Approved Dosimetry Services (UK)[4] for dosimetry purposes, rigid laboratory procedures and regulations relating to movement of radiation workers between member states[5,6].

Use of one or more radionuclide.

The activity handled would need to be reduced if more than one radionuclide is used. This is frequently the case in the type of laboratory referred to above.

The activity handled of each radionuclide can be derived quite simply by estimating the fraction of the ALI to be used.

		activity handled
^{32}P	1/10 ALI	7.5×10^{9}
^{35}S	1/5 ALI	1.0×10^{11}
^{3}H	7.10 ALI	1.0×10^{12}

Figure 2 Activities Handled in Multiple Use

The sum of the ALI's must not be greater than 1.

In the case of radionuclides with a high beta, or gamma dose rate it may be necessary to use additional shielding to reduce the workers occupational exposure. This may be a limiting factor in the activity handled.

External Exposure

The revised European Directive[7] will probably set a level of 6mSv above which workers should be designated as Category A or classified workers under UK legislation. Existing UK legislation[8] sets a dose rate of $7.5 mSvh^{-1}$ as the level above which a worker should normally be classified. The use of a time average dose rate over an 8 hour period, so as not to exceed $7.5 \mu Svh^{-1}$, allows some flexibility.

If the new European Legislation requires the designation level to be set at $6mSvy^{-1}$ corresponding to $3\mu Svh^{-1}$ the time average dose rate could be changed to $100\mu Sv$ over a 40 hour period. This would then give an average dose rate of $2.5\mu Svh^{-1}$ which would satisfy a 6mSV designation level.

This would also help reduce the potential for over classification using open sources as discussed earlier.

Conclusion

The classification of workers using open sources need to be considered in the implementation of the Directive the above could use.

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EVALUATION OF THE ENVIRONMENTAL DOSE AND THE ^{131}I CONCENTRATION IN A LABORATORY OF RADIOACTIVE MATERIALS PRODUCTION

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INTRODUCTION

Among the radioactive materials produced at the Department of Radioactive Materials Production of Instituto Nacional de Investigaciones Nucleares to be used in nuclear diagnosis and therapy in Mexico, ^{131}I which emits approximately equal numbers of medium-energy beta particles and gamma rays, has special importance notwithstanding its short half-life due its affinity with vital glands (1). Therefore, the air must be monitored because significant levels of airborne contaminators can occur in working areas (2).

In this laboratory there are working 10 workers making the dilution of $Na^{131}I$ to be used in Nuclear Medicine for medical examination and treatment of thyroid condition.

The aim of this work is to determine the ^{131}I annual incorporation levels as well as the personal and environmental dosimetry in the working area. Concentration of ^{131}I in the laboratory of radioactive materials production atmosphere and the absorbed dose produced for this concentration are determined. External dosimetry both personal and environmental were obtained with locally made thermoluminescent dosimeters.

PERSONAL MONITORING

a) Air monitoring for intake.

The inhalation route of the personal subject to potential intake of ^{131}I was monitored at the breathing zone (3). This was done with personal air samplers Victoreen 08-430 consisting of a charcoal filter and a small battery-driven pump that draws air though it. The filter was fastened to the lapel of the worker's lab coat to be located near the breathing zone. The pump were operated at a pumping rate of 6.99 liters /min.

Air samplers were used by the workers along six months. Filters were evaluated every day at the end of the work's day by means of gamma spectroscopy considering sampling times in the range of 120 to 150 minutes. Iodine-131 concentration was determined at room temperature (20°C) and at atmospheric pressure of 532 mmHg.

Radioactive concentration (Bq/liter) of ^{131}I was evaluated by means of gamma spectroscopy of the filters by dividing the net counts (counts - background) by the product of the detector efficiency (counts/gamma-ray) by the counting time (s) and by the sample volume (liters) (4):

$$C = \frac{counts - background}{E\, t_c\, Y\, V_c}$$

Dose equivalent values were calculated by multiplying the ICRP dose conversion factor, 2.9×10^{-4} mSv/Bq, by the air derived concentration in Bq/m^3, and by the volume of air breathed in over the working period under conditions of "light activity", 1.2 m^3/h x 5 h (5). In equation form this is expressed as follows:

$$D(mSv) = ADC(Bq/m^3) x 1.2(m^3/h) x 5(h) x 2.9 x 10^{-4}(mSv/Bq)$$

b) External dosimetry

Personal monitoring of the workers was performed using locally made thermoluminescence dosimenters. Personal dosimeters were constituted by four LiF:Mg,Ti+PTFE pellets placed in plastic badges of the thickness required to fulfil electronic equilibrium conditions. Monitoring devices were worn on the trunk, between waist and shoulder level at the site of the highest exposure level to which the body is subjected. Personal thermoluminescence dosimeters were worn by the workers for a month between changes because the risk of excessive exposure is low.

Environmental monitoring of the working area was performed using $CaSO_4$:Dy+PTFE locally made thermoluminescence dosimeters. Environmental dosimeters were placed at 1.60 m floor level to be exposed along one month.

Thermoluminescence readings were made in a Harshaw 4000 TL analyzer at a heating rate of 10°C/s integrating the signal from 30°C to 300°C during 30 seconds. All readings were made in N_2 atmosphere.

Previous to be used and reused, the dosimeters were submitted to an specific thermal annealing. Thermal annealing consisted in heating at 400°C during 1 hour followed by 100°C during two hours for LiF dosimeters and 300°C during 30 minutes for $CaSO_4$ dosimeters.

Doses were evaluated interpolating in a calibration curve previously elaborated by irradiating a batch of dosimeters at given doses using a Victoreen 64-764 (12.79 GBq) ^{137}Cs irradiator.

RESULTS AND DISCUSSION

Results showed that ^{131}I derived air concentrations received by occupationally exposed personal and the absorbed doses resulting from these concentrations were lower than the limit established by the ICRP, except that obtained for a worker, labouring during two months at the radiopharmaceuticals packing area, which received a dose three fold the established limit.

Personal dosimetry revealed that two workers received an absorbed dose two fold higher than the limit due that these workers were labouring at the radiopharmaceutical packing area.

Environmental monitoring results showed that two points, located at the packing area and at the labelled radiopharmaceuticals area, registered radiation levels 5 times higher than the typical environmental values.

CONCLUSIONS

These results suggested that the hot cells in the radiopharmaceuticals packing area could have any radiation leakage which was corrected immediately reducing these radiation levels until admissible values. However, these areas must be classified as controlled areas.

ACKNOWLEDGEMENTS

The authors express their gratitude to E. Quintero of Environmental Radiation Protection Laboratory, ININ for her technical assistance in the gamma spectroscopy measurements and to J. Lezama of Radioactive Materials Production Department, ININ for providing facilities to make the field measurements.

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The on-line operationnal dosimetry of workers as a key issue for neutron radioprotection optimisation.

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Handling and processing of various chemical forms of Plutonium leads to neutron irradiation, for example due to (α,n) reactions. Under some circumstances, workers are therefore subject to occupationnal exposures.

Regulation regarding individual dosimetry is very stringent to implement. Furthermore, individual dosimetry is not well adapted for operationnal measurement because of the overwhelming response time, specially in case of on-line neutron monitoring on the spot. R.T.L. dosimetry , P.G.P.-D.I.N. type, carried out by differential measurements using ^{6}LiF and ^{7}LiF detectors, was the first achievement. Since then we have been using BD 100R bubble dosimeters. These detectors are located nearly the most critical work places as far as neutron radiation is concerned.

This paper deals with neutron characteristics of the different work places in term of spectrum and intensity. It focuses on bubble dosimeter operation : calibration, drift of the sensibility and reading frequency, function of reset operations, etc... Our main goal is to use in the near futur this dosimeter for the on-line monitoring of neutron. This approach will enable us to optimize the neutron doses delivered to the most critical work places and to make workers sensitive to health physics issues.

Preliminary results are reported as well as our health physics policy in respect to the A.L.A.R.A. principle.

RADIATION PROTECTION AT A RADWASTE TREATMENT AND STORAGE FACILITY IN THE NETHERLANDS

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1 INTRODUCTION

Since 1984 the radioactive waste produced in the Netherlands is managed by COVRA, the Central Organisation for Radioactive Waste. This dedicated organisation has been set up as a joint venture between the three major radwaste producers in the country and the government.
All kinds of radwaste (LLW, ILW and HLW) that will be generated in the next 100 years will be treated and stored in an above ground facility.
At this moment COVRA's activities are collection, treatment and storage of LLW and ILW. The conditioned HLW is expected to arrive in the year 2001.
The treatment and storage facility for LLW and ILW is located in the south-west corner of the Netherlands near the town of Vlissingen (Flushing).
The construction of a storage building for HLW in Vlissingen will start in 1997.
The methods of radwaste treatment are chemical cleaning by means of a precipitation technique of liquids, incineration of organic liquids, supercompaction of solids, reducing large items and conditioning treated radwaste by cementation.
At this moment, in the storage facility there are approximately 20,000 drums of conditioned LLW stored with a total volume of 7500 m^3 and a total activity of 0,5 PBq. This waste has been collected in the period 1982 - 1995.
The new facility is in operation since 1992.
Radiation protection during 1992 - 1995 is quantified by monitoring the radiation during treatment and storage, dosimetry of the work preformed by radwaste workers and measuring radiation levels at the fence of the facility.
In the facility near Vlissingen there are some 45 employees working of which 30 are radiological worker.

1.1 Producers of radwaste

In the Netherlands the radwaste producers are research institutes, hospitals, ore-processing and manufacturing industry, and nuclear facilities (two nuclear power plants and two nuclear research reactors).
The annual amount of LLW produced by the users of radioactive materials in research, medicine and industry equals roughly the amount of ILW produced by the nuclear facilities. This amount is after treatment, conditioning and packaging some 500 m^3.

1.2 Radwaste management organisation and facilities

In general terms the task of COVRA is to manage all Dutch radwaste, in order to protect man and the environment against the hazardous effect that could be caused by this radwaste. This means more specifically:

- collection and shipment of radwaste
- treatment and conditioning of radwaste
- storage of all kinds and categories of conditioned radwaste
- conventional and radiological protection

- monitoring the radwaste before treatment
- monitoring the radwaste after conditioning
- monitoring the environmental impact of treatment and storage

2 ACCOUNTANCY OF RADWASTE TREATMENT AND RADIOLOGICAL ASPECTS

Before shipment an administrative and radiological control of the radwaste is preformed by radiation protection monitors (RPM's) of the Health Physics department of COVRA. These measurements are executed to check if there is compliance with COVRA's standards of LLW and ILW.
After shipment there is an extra administrative and radiological check of the radwaste at the receptionbay of COVRA and after agreement the radwaste is transported into a bufferstorageroom in the treatment building.
During treatment and storage of the radwaste there is a daily and a weekly program of monitoring preformed by the RPM's. They measure the doserate and the level of contamination in the treatment- and storage buildings.
The results of these measurements should be in accordance with the radiological zone classification of IAEA.
The emission of aerosols and gasses (α, ß, γ, SO_2, NO_2 and CO) in the ventilation air is accounted for by sampling and analyzing airfilters from the ventilation systems and the ventilation stack.
The emission of radioactivity in the wastewater is accounted for by analyzing the radionuclides and the emission of chemical substances (heavy metals, MAC, EOX and COD) in the treated wastewater.
The limits of emission in air per year permitted by the Dutch authorities are for α-emitting nuclids: $2.5*10^4$ Bq, for ß/γ-emitting nuclids: $2.5*10^{10}$ Bq, non-ionising gasses SO_2: 40 mg/Nm^3, NO_x: 70 mg/Nm^3 and CO: 50 mg/Nm^3.
The limits of emission in water per year permitted by the Dutch authorities are for α-emitting nuclids: $1.5*10^7$ Bq, for ß/γ-emitting nuclids: 10^{11} Bq, non-ionising heavy metals: 0,5 mg/l, MAC: 0,5 mg/l, EOX: 0,1 mg/l and COD: 1100 mg/l.
These limits were during the years 1992 - 1995 never exceeded.
The doserates at the fence of the COVRA-facility are monitored. These measurements are used for the accountancy of the dose caused by COVRA's activities. The contribution to the doses at the fence of the facility (figure 1) was in **1992** ; 0,01 μSv, in **1993** ; 6 μSv, in **1994** ; 8 μSv and in **1995** 6 μSv (**1991** is used as a referencepoint).
The radiological protection of the workers, contractors and visitors of the treatment and storage facility at COVRA are monitored by electronical dosimeters. The doses of the workers of COVRA are also monitored by thermoluminescence dosimeters (TLD). These TLD's are once a month controlled by an Approved Dosemetry Service and the results are reported back to COVRA and the results are also stored in a centralized databank system (NDRIS).
The collective doses of the radiological workers of COVRA in 1992 was 10,5 manmSv (32 man), in 1993 it was 7 manmSv, in 1994 it was 7 manmSv and in 1995 it was 16 manmSv (figure 2).
The collective doses of contractors in 1992 was 4,6 manmSv (3 man) and in 1993 it was 0,7 manmSv (3 man). COVRA made no use of contractors in the

period 1994 and 1995.
During the period 1993 - 1995 approximately 2000 people visited the facility and during those visits a personaldose of 6 μSv was never exceeded (mostly however 1 - 2 μSv).
The radiation protection during collection and transportation of radwaste from the producers is preformed by RPM's of the Health Physics department.
The operators are accompanied by a RPM during treatment and storage of radwaste. The RPM preforms the radiological measurements of the radwaste and is responsible for the radiation protection.
Persons leaving the controlled zone are controlled on α/ß contamination and on γ contamination.
A reading device for the electronical dosemeters gives the person a read-out of his personal dose collected during that day.
Twice a year the workers in the controlled zone of COVRA are informed by the radiation protection officer (RPO) of their personal dose .
The RPO is mandated by the license holder COVRA and is responsible for the radiation protection at COVRA.

Figure 1

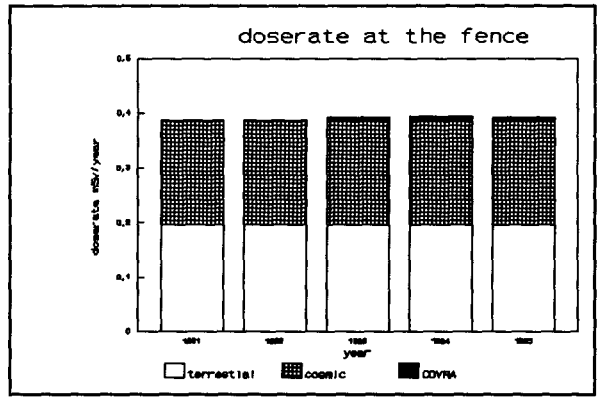
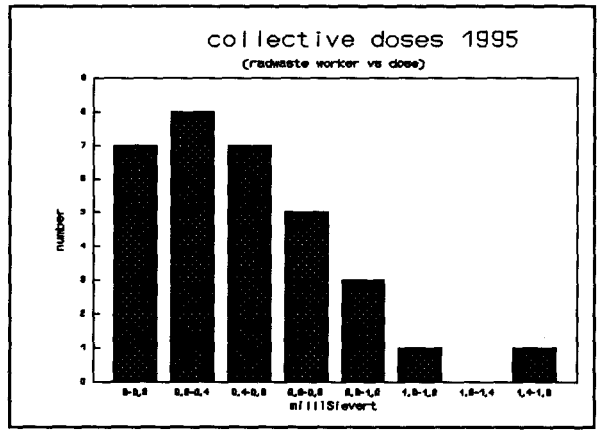


Figure 2



METHODOLOGY FOR RISK ASSESSMENT FOR WORKERS HANDLING UNSEALED RADIOACTIVE SOURCES

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INTRODUCTION

Occupational handling of unsealed radioactive substances may result in internal or external contamination of the worker or in releases to the environment.
We developed a methodology for risk assessment for workers who handle unsealed sources and for the estimation of releases to the environment. Results of risk assessments are helpful to establish derived limits regarding the quantities of radioactive substances that can be handled within defined dose constraints, appropriate for the actual working scenarios.
This study is supported by the Dutch Department of Labour, with the special objective to develop constraints on the source activities in laboratory work and other fields of application. In this paper, we give some highlights of the study, focussed on the dose consequences for the worker.

MODEL ANALYSIS

In essence our study is concerned with the model analysis of the relationship between the quantities of the unsealed radioactive substances that are handled in a particular practice and the potential dose consequences for the workers. A major factor in this modelling is the description of the characteristics of these scenarios, in particular the working techniques, the physical and chemical properties of the material and the protective measures that are taken.
As regards the working conditions we distinguish regular working conditions, which indicate that the progress of work is as it was planned, and incidents that are perceived as small mishaps which can occur and for which it is reasonable to assume that they will occur every now and then.
The model analysis that we developed also provides the option to calculate dose consequences in accident situations. In accident situations, however, the dose consequences for the worker are not the only factor that should be considered. The probability of accidental events plays an important role as well. Therefore, the risk assessment for accidents is beyond the scope of this paper.

The dose consequence for the worker in terms of committed effective dose can be described as a function of the activity intake and the dose coefficient DC for the pathway under consideration. The intake can be described as the product of the source activity ACT and the fraction of this activity, which is transferred to the worker – the transfer fraction to the worker – TFW:

$$\text{DOSE CONSEQUENCES} = \text{INTAKE} \times \text{DC} = \text{TFW} \times \text{ACT} \times \text{DC}$$

From this expression, it is clear that the transfer fraction for the worker TFW is an unambiguous measure for the consequences for the worker. In principle, this transfer fraction should be determined for all relevant pathways. However, some pathways cannot be described in a simple way with a mathematical model. Furthermore, the practical importance of the various pathways is different. Exposure of the worker due to the air pathway can happen under normal working conditions and as a result of incidents or accidents. Exposure of the worker due to contamination of the skin, ingestion or injection is always an incident or accident.
Therefore, we developed a mathematical model for the air pathway. This mathematical model is based on a general algorithm that solves first order compartment models, including recycling. Our model makes calculations of transfer fractions possible, not only under regular working conditions

but for incident and accident scenarios as well. After calculating the transfer fractions for the air pathway, the other relevant pathways are judged retrospectively.

Restrictions to the activity of radioactive substances can be described in terms of restrictions to the product of activity and dose coefficient. In actual situations, an upper bound for this product (ACT × DC) can be derived from the dose constraint for a certain kind of practice.

$$TFW \times (ACT \times DC) \leq DOSE\ CONSTRAINT$$

The product (ACT × DC) is a very elegant way of describing the upper bound for the activity, because it is a measure for the 'radiotoxicity' of the activity handled. When the transfer fraction and the dose constaint are known, the product (ACT × DC) is the same for all radionuclides.

The transfer fraction and the dose constraint are different for regular working conditions and for incidents. If we want to derive guidelines for constraints of activity, it is necessary to choose a dose constraint for regular working conditions and for incidents. Developing dose constraints was not the objective of this study. However, the study can provide insight into the meaning of dose constraints.

The mutual difference in consequences for the worker in different scenarios can be expressed by the ratio of respective transfer fractions. Assuming that the consequences in a certain actual scenario A and in a so-called *reference scenario* R are identical, it follows that:

$$TFW_R \times (ACT \times DC)_R = TFW_A \times (ACT \times DC)_A$$

Once an upper bound of (ACT × DC) for the refence scenario has been derived (or chosen), the upper bound of (ACT × DC) in the actual scenario can be derived, assuming identical consequences for the reference scenario and the actual scenario:

$$(ACT \times DC)_A = \frac{(ACT \times DC)_R}{TFW_A / TFW_R}$$

The ratio of the transfer fractions TFW_A/TFW_R can be regarded as the correction factor needed to derive the upper bound of (ACT × DC) for the actual scenario.

RESULTS: ABSOLUTE MEANING OF TRANSFER FRACTIONS

We calculated transfer fractions for a variety of scenarios. In this paper, we present a summary of the results for a 'standard' laboratory. The results are given for working in a fume hood, for one month of regular working conditions and for the 'maximum reasonable incident', respectively.
The maximum reasonable incident is failure of the fume hood. The transfer fraction for one month of regular working conditions is calculated by multiplying the transfer fraction for a full working day by 20. The transfer fraction for the incident is calculated over a period of one hour after the incident.
From this summary of results it can be shown that the transfer fraction for the worker during one month of regular working conditions is always less than the transfer fraction for the maximum incident. Furthermore, it may be expected that the dose constraints for regular working conditions and for incidents, although numerically unequal, are in the same order of magnitude. Taking into account that it is prudent to assume that an incident will occur once a month for planning purposes, this means that the transfer fraction for the incident situation is dominant in the risk evaluation. Therefore, derived limits should be based on the incident circumstances. An adequate level of protection under regular working conditions then is implicitly guaranteed.

A very important conclusion from this study is that it is not effective to improve the ventilation qualities of a fume hood beyond a certain level. Provided that a certain level of protection by the fume hood is guaranteed, the upper bound of (ACT × DC) is determined by the incident situation, in which the fume hood is not working at all. On the other hand, it is advantageous to improve the operational safety of the fume hood in order to prevent incidents.

RESULTS: RELATIVE MEANING OF TRANSFER FRACTIONS

We compared the calculated transfer fractions for a number of actual scenarios (for working in a fume hood) to the transfer fraction in a chosen reference scenario. This ratio of transfer fractions can be taken as the correction factor TFW_A/TFW_R for determining the upper level of $(ACT \times DC)_A$ in the actual scenario, once the upper level of $(ACT \times DC)_R$ for the reference scenario is known.

The comparison refers to incidents, because the incident circumstances determine the maximum activity that can be handled in a certain scenario.

We have chosen the *handling of an aqueous solution in a fume hood in a standard laboratory* as the reference scenario.

SUMMARY OF RESULTS FOR WORKING IN A FUME HOOD IN A STANDARD LABORATORY

TFW – one month regular working conditions	*TFW – maximum incident*	*SCENARIO*	*TFW_A / TFW_R for incidents (rounded)*
4,6 E-3	1,4 E-1	Boiling of a volatile or an aqueous solution*	2000
4,4 E-3	8,1 E-2	Boiling of a moderately volatile solution*	1000
1,7 E-3	1,1 E-2	Working with very fine dust	150
4,4 E-5	1,4 E-3	Spattering processing of a volatile solution*	15
2,2 E-4	1,2 E-3	Gas or vapour in a holder or working with dust	15
4,4 E-5	8,1 E-4	Spattering processing of an aqueous solution*	10
4,4 E-6	1,4 E-4	Working with a volatile solution*	2
1,7 E-5	1,2 E-4	Spattering processing of a moderately volatile solution*	2
4,4 E-6	8,1 E-5	Working with an aqueous solution* (reference scenario)	1 (by definition)
1,7 E-6	1,2 E-5	Working with a moderately volatile solution*	0,2
4,4 E-7	8,1 E-6	Preparation of syringes with an aqueous solution*	0,1
2,2 E-9	1,2 E-8	Solution of a non-volatile material	0,0002

*It is assumed that the radionuclide in the solution is equally volatile as the solvent

OTHER IMPORTANT RESULTS

The methodology for calculating transfer fractions allows us to derive upper bounds for the activity handled, and not only for working in laboratories, but for other situations as well (nuclear medicine, working with animals, radionuclide production and other industrial applications).

The method provides an insight into the differences between the consequences of different working conditions, different physical properties of the material handled and different protective measures.

The methodology is also a useful tool in optimisation studies, because it can provide insight into the effect of additional protective measures (e.g. an extra containment).

The method also gives us an opportunity to derive transfer fractions to the environment for the air pathway. This makes it possible to derive upper levels of activity for unsealed radioactive sources, based on upper bounds for the emissions of activity.

RADIOACTIVE CONTAMINATION CONTROL PROGRAMS IN RADIOACTIVE INSTALLATIONS OF BIOLOGICAL RESEARCH

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INTRODUCTION

Radioactive compounds are nowadays widely used in Biological Research. There is a great quantity of analytic radioisotopic techniques since they yield evident advantages like specificity, versatility and sensitivity in comparison with non radioactive protocols. Nevertheless, it is obvious that manipulation of radioactive unsealed sources may generate risks for exposed workers.

The aim of our work has been to evaluate the probability and incidence of several contamination types that can occur. In order to achieve this purpose, we have make an exhaustive analysis of the most common radioisotopic techniques (table I) used in our Research Centers. Also, we have taken into account physical characteristics and radiotoxicity of the radioisotopes used in such techniques.

Type of mollecule	Most Common techniques	Labeled compouns mainly used
Nucleic acids	Northern, Southern, "in situ" hybridization	nucleotids-^{32}P, -^{33}P
	Sequencing	nucleotids-^{35}S, -^{32}P, -^{33}P
	Activity RNA-polimerase based methods	nucleotids-^{32}P, -^{33}P
	Footprinting, bandshift	nucleotids-^{32}P, -^{33}P
	Activity of genetic promoters	chloramphenicol, acetyl CoA-^{14}C
	DNA synthesis	Thyimidine-^{3}H
Proteins	Western blot, radioimmunoassays	Na^{125}I
	Binding hormone-receptor assays	ligands-^{32}P, -^{35}S, -^{125}I, Na^{125}I
	Enzymatic kinetics	several-^{32}P, -^{35}S, -^{125}I, -^{3}H, -^{14}C
	Proteins synthesis	methionine, cysteine-^{35}S
Carbohydrates and lypids	Action mechanism of steroydic hormones	ligands-^{14}C, -^{3}H
	Metabolism of lypids, cholesterol and carbohydrates	cholesterol, precursors-^{14}C, -^{3}H

Table I: Radioisotopic techniques most commonly used in Biological Research

The main factors that we have considered in relation to the probability and incidence of possible contamination are: maximum activity used per assay (assays carry out "in vivo" or "in vitro"), physicochemical characteristics of radioactive compounds used, and of his main and residual compounds; technical complexity (manipulation type and equipment needed). This information has been quite useful to set up the contamination control program that we expose below.

MATERIALS AND METHODS

In order to determine contamination levels in the different techniques indicated below, we have performed the following measures:

-Surface contamination: surfaces and diverse material.

-Enviromental contamination: coal filter, coal samples used during the manipulation of volatile compounds (labeled with S-35 or I-125 and/or I-131), liquid effluents.

-Personal contamination: external (protection clothes and skin) and internal (these measures haven't been done because of not having the necessary resources).

The measuring time depends on the kind of equipment or surface to measure. The distance between the monitor and the surface or equipment has been about 1 cm.

In the radioisotopic techniques using P-32, I-131 and/or I-125 the measures have always been directly taken.

In the case of S-35, C-14 and P-33, the measures have been both direct and indirect, and always indirect measurement in case of H-3.

The equipments used in the direct measurements were:

-BERTHOLD:LB-122 and 123.

-ROTEM: Ram-Gene1, Ram-Surf, Ram DA-3 with PM-10 ang GM-10 probes

-MINI-INSTRUMENTS: E, EP-15 and 44A.

The indirect measurements have been performed by smearing with filter paper wet in decontaminant solution are afterwards counting in a liquid scintillation counter.

We have evaluated the factors considerated in the introduction and according to them, we have been able to provide the specific hazards of each technique .All these data together with bibliographic references,have decided the necessary protection protocols.

* Radiolabelling of biological molecules with I-125 or I-131

* *Asociated risk.*

Volatility of radioactive iodo during the labelling experiment. This can produce personal, external and internal, contamination and superficial contamination.

* *Protection measures.*

- Storing the labelling compounds at room temperature.
- Avoid the use of strong oxidants.
- Work always into a fume hood with charcoal filters.
- Use the Hamilton syringes for dispense the radioactive compounds.
- Take advice to put the plates or filters of active charcoal into the fume hood and into the work place to prevent airborne contamination.

* Radiolabelling of proteins with S-35 amino-acids.

**Asociated risk..*

Releasing of ^{35}S--labeled volatile sub-products. Although this compounds have not yet been identified, likely candidates are methyl-mercaptan (CH_3-^{35}SH) as the major volatile product of radiolytic decay of methionine, with hydrogen sulfide H_2S and sulfur dioxide as minor components. Netherless, some authors suggest that less volatile sub-products are released from ^{35}S-cysteine than from ^{35}S-methionine.

Release of volatile radioactive compounds from ^{35}S-amino acids happens during storage, protein-labelling experiments and incubations.

These compounds are readily soluble in water, for this reason the incubators used in cell culture could become highly contaminated.

* *Protection measures*.

- Stored at -80º the 35S-aminoacids in dry ice
- Traw the products in a fume hood with charcoal filters
- Place the filters of activated charcoal in the incubator (this filter does not affect the CO_2 equilibrium in the incubator).
- The water inside incubator should be changed after each labelling.
- Use a sufficient volumen of tissue culture media (75µCi/ml).
- Add an effective stabilizer agent to the tissue culture media (tricina 50 mM). This agent should not be toxic.

*Radiolabelling of biological molecules with H-3

** Asociated risk.*

- The tritiated compounds used for biological research wiith high risk of contamination are tritiated water, tritiated thymidine and some organic compounds .
- The biological risk highly depends of the tritiated compounds used. Tritiated thymidine presents a high risk of internal contamination because these compounds are quickly metabolized into the genetic material .
- The tritiated water (especially in high specific activity) causes contamination of the frost in freezers when this compounds is stored.
- Tritium gas, tritiated water and some tritiated organic solvents pass through the walls of plastic containes easily.
- Compounds stored in aqueous solution may give rise to tritiated water by exchange which can then result in contamination of the frost in freezers.

** Protection measures .*

- Experiments with tritiated water or tritiated compounds must be carried on in a fume cupboards (with charcoal filters) to prevent airborne contamination.
- The refrigerators and incubators where tritiated compounds are stored should be checked to detect possible contaminations .
- Sealed glass vials in wich labelled compounds have been stored for long periods should be opened with great care, preferably in a fume cupboard or glove box.
- Tritiated water should be stored at room temperature in sealed containers.
- Wear two pairs of gloves.

CONCLUSIONS

Due to the analysis carried out on the application of radionuclides in biological research we have established a program for contamination control and surveillance and we have set up contamination Reference Levels in these radioactive installations:

REFERENCE LEVELS	Bq/cm^2
Surface contamination	1
Skin	0,37

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RADIATION PROTECTION IN THE RADIOACTIVE INSTALLATIONS OF THE AUTONOMOUS UNIVERSITY OF BARCELONA

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INTRODUCTION

In the Autonomous University of Barcelona (UAB) there are numerous investigation and teaching activities that imply the use of equipment generating ionizing radiations or the manipulation of radioactive isotopes. The existence of disciplines so different as medicine, veterinary, biology, physics or chemistry gives rise to a broad diversity of applications, concerning the techniques carried out as well as to the equipment and radioactive isotopes that are used. Other notable characteristic of the university radioactive facilities is its spatial dissemination, being located in different faculties and departments distributed all over the campus.

The Spanish legislation (1) establishes the regulation on the protection against ionizing radiations according to the guidelines of the EU (2). Furthermore, it exists a regulatory organization in radiological safety matter that it is the Spanish Nuclear Security Council (Consejo de Seguridad Nuclear: CSN) (3). This organization checks and authorizes the performance protocols of all the nuclear and radioactive facilities of Spain. In order to coordinate the inherent activities to the management of the ionizing radiations in the UAB, it was necessary to create a Radiological Protection Service (RPS). Its most important task is to remind of the risks limitation principles in the use of the ionizing radiations to the involved professionals and to control the fulfillment of the radiological protection regulations in use.

The basic document of the RPS is the Radiological Protection Handbook that must be approved and authorized by the CSN. All the procedures and guidelines concerning the organization and safety management of the ionizing radiations used in the UAB investigation and teaching activities are written in this document. Between these procedures are emphasized those which are referred to the environmental radiological vigilance, concerning radiation levels and contamination both inside and outside of the installations, and also to the control of the radioactive waste management. Within this last, one way to manage the liquid radioactive waste is dilute it and make a controlled evacuation through the sewer that guarantee that the activity levels are below the limits authorized by the CSN (1). The activity limits authorized for release in a controlled way the liquid radioactive effluents are found explicitly collected in the Radiological Protection Handbook of the UAB (4).

The RPS of the UAB periodically carry out radiological controls in the different radioactive installations of the campus. Such controls envisage the environmental radiation level measurement and the detection of the radioactive contamination in the different work surfaces. Furthermore, it has been performed a campaign to control the liquid waste release in the sewer net of the campus in order to verify that the limits established for the controlled liquid waste pouring away are sufficient to assure a good radiological quality of the residual waters. The knowledge of the radiation and contamination levels will permit the optimization of the techniques used, specially in view of the next incorporation of the new recommendations of the ICRP (5) to the Spanish legislation. Furthermore setting up a liquid waste sampling and measurement method results indispensable for the evaluation of the impact of a hypothetical accident that could happen in any installation of the university campus. The methodology used and the results obtained from these controls are described in the present paper.

INSTALLATIONS, EQUIPMENT AND RADIOISOTOPES

The UAB holds 9 radioactive installations that develop investigation and teaching activities. The denomination of these facilities and the radioactive sources used are detailed in Table 1.

Table 1. Denomination and radioisotopes used in the UAB installations

Code	Denomination	Faculty	Radioisotope
BQ-C	Biochemistry	Sciences	^{32}P, ^{35}S, ^{125}I
BQ-V	Biochemistry	Veterinary	^{3}H, ^{14}C, ^{32}P, ^{35}S, ^{125}I
GFR-C	Radiation Physics	Sciences	enclosed sources
G-C	Genetic	Sciences	^{32}P, ^{35}S
IBF	Biochemistry	Institute	^{3}H, ^{14}C, ^{32}P, ^{35}S, ^{45}Ca, ^{125}I
SRE-C	RIA Service	Sciences	^{3}H, ^{32}P, ^{125}I
UB-M	Biochemistry	Medicine	^{3}H, ^{14}C, ^{22}Na, ^{32}P, ^{45}Ca, ^{86}Rb
UF-V	Physiology	Veterinary	^{3}H, ^{14}C

The nature and characteristic of the radioactive isotope used by the personnel of the UAB installations are related in Table 2. Furthermore, the faculty of veterinary counts on one installation of radiodiagnostic equipped with an X-rays device (maximum characteristics: 100 kVp, 100 mA and 1000 ms). The RPS gives coverage to a total of 140 persons including students and professionally exposed personnel. These persons are subject to the corresponding medical and dosimetric control. The medical control is performed at the UAB in a center authorized by the CSN. The Spanish legislation establishes that the professionally exposed personnel of the radioactive facilities must accredit a license for working with ionizing radiations and at present there are 17 Supervisor licenses and 6 Operator licenses in the UAB campus. In each installation, the Supervisor is responsible of the radiological safety routine controls without prejudice of the RPS actions.

Table 2. Characteristics of the radioisotopes used in the UAB.

Radioisotope	Radiation Type	Maximum Energy (MeV)	Half Life
^{3}H	β	0.0186	12.3 years
^{14}C	β	0.156	5730 years
^{22}Na	γ	1.28	2.6 years
^{32}P	β	1.71	14.3 days
^{35}S	β	0.167	87.4 days
^{45}Ca	β	0.083	163 days
^{86}Rb	β	1.77	18.66 days
^{125}I	γ	0.035	60 days

METHOD AND MEASURING SYSTEMS

The environmental radiological vigilance that carries out the RPS with a semiannual periodicity include the control of the environmental radiation levels and the detection of the surface contamination in the different installations. The radiation levels are determined using an ionization chamber ROTEM RAM ION connected to a portable computer that stores in a data base the levels of each controlled installation. The detection of the contamination in the surfaces is carried out with the same counter and, in the case of being detected a level significantly higher than the background, the activity by surface unit is determined with the counter FAG CONTAMAT. This counter is provided with a proportional probe for beta-gamma radiation, that is calibrated according to a library of 25 different radioisotopes. When the access to the surface is difficult, as for example drain, the measure is carried out by rubbing the surfaces with an absorbent paper that is after measured. The efficiency of the ionizing chamber RAM ION is 30% for energy of 600 keV and its response is compensated for a range of 10 to 1000 keV. For the superficial contamination counter the manufacturer indicates an efficiency of 34% for the ^{90}Sr ^{90}Y. The ionization chamber is calibrated each two years by an official metrological laboratory and the counting response of surface contamination probes is verified annually using a set of calibration sources by the own RPS.

The control of the liquid waste activity levels has been carried out through a specific campaign that have take in account the most meaningful installations concerning the use of radioisotopes. The selected installations have been: the SRE-C, the UB-M, the UF-V and the IBF. In each installation has been taken residual water samples in points of the sewer corresponding to the drain of the facility. This sampling has been accomplished during the controlled pouring of the liquid wastes previously diluted according to the protocol established by the RPS (4). For each installation were taken five samples of a liter, the first before beginning the release and the remainders each ten minutes. Beta-emitters were measured by WALLAC 1220 QUANTULLUS low background liquid scintillation counter. For counting by means of liquid scintillation system, the samples were conditioned following the preparation method established for the environmental samples measurement by our Group (6). The determination of ^{125}I activities was accomplished in a WALLAC 1271 CLINIGAMMA solid scintillation counter (7). The ^{22}Na level was measured using a ORTEC GMX high resolution gamma spectrometer provided by a HPGe detector linked to a 8k multichanel analyzer (8).

RESULTS

The periodic controls carried out by the RPS of the UAB have shown levels not significantly different from the environmental background value, namely 0.1 μSvh^{-1} for irradiation and 13 cps (beta-gamma average) for surface contamination. Higher irradiation levels, up to 0.5 μSvh^{-1}, have been observed only in a few occasions, corresponding to situations in which wastes or radioactive material in use were found inside the working zone. Furthermore, in one control over the working surfaces, contamination was detected in the absorbent cover paper reaching levels of 5.5 $Bqcm^{-2}$ of ^{32}P. The obtained results confirm the need of hold the wastes and radioactive material always outside of the working zone and prove the effectiveness of the periodic radiological controls. In the other hand, the control of liquid radioactive wastes poured through the drain has involved ^{125}I, ^{22}Na and beta total determination from samples taken from the sewers. In the case of the ^{125}I activities slightly higher than the background level have been observed just in the middle of the sampling decreasing to the background value in ten minutes approximately. These values must be investigated using a counter with a lower detection limit in view to determinate the activity as far as the ^{125}I has the lowest authorized limit of the radioisotopes analyzed. In the case of the ^{22}Na, the levels of the samples from the UB-M were below the minimum detectable activity of the detection system, namely 10 $mBql^{-1}$. With respect to the beta total determination, samples analyzed were found to have activities below the minimum detectable activity of the counting system, namely 0.1 Bql^{-1}. This result shows the goodness of the beta liquid effluent release protocol. Radiological survey in the radiodiagnostic installation has showed the following levels of irradiation: 0.1 μSvh^{-1} in the vicinity of the operator console and 0.7 $mSvh^{-1}$ for the leakage radiation test (placing the detector at 1 m from the X-ray tube and working with maximum generator parameters). The results prove that the installation was well-designed and the leakage radiation verify the legal prescription, i.e. < 1 $mSvh^{-1}$.

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EVALUATION OF OCCUPATIONAL EXPOSURES IN THE CZECH REPUBLIC

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ABSTRACT - The recent situation in the Czech Republic (CZ) concerning the evaluation and recording of occupational radiation exposures is described. The individual monitoring is based on the interpretation of the personal dosemeters responses and the evaluation of the other special dosimetric methods (measurement of excreta, whole body counting, etc.). The evaluation of occupational radiation exposures is carried out by five approved dosimetric services, which control about 20,000 workers. Record keeping of overexposures is based on two systems, which principles are explained. Based on long time analysis of occupational radiation exposures in CZ it can be present that the average values and trends of individual and collective effective dose equivalents are comparable with those in developed countries. The distributions of the radiation exposures for the important occupational groups of workers are presented.

INTRODUCTION

The individual monitoring of workers with sources of ionizing radiation (SIR) in CZ is based on measurement, evaluation and registration of occupational radiation exposures (ORE) using personal dosimeters or the other special methods (measurements of organ doses, of excreta, the whole body counting, etc.). The licensee's duty, covered by law in CZ[1], is to secure the control, monitoring and recording of ORE of its workers.

Recent legislation in CZ, in accordance with political and economical changes in the country, is going through an extensive reconstruction. A proposal of the Law on Peaceful Use of Nuclear Energy and Ionizing Radiation (new "Atomic Act") [2] has been already elaborated (it is assumed to come into force during 1996). The Atomic Act and related regulations are based on ICRP Report 60 [3] and New International Basic Safety Standards [4], i.e. licensees will be responsible not only to designate controlled areas where specific protective measures are required for:

* control of ORE and prevention of the spread of contamination during normal working conditions,
* prevention and limitation of the extent of potential exposures,

but shall maintain exposure records for each worker for whom an assessment of ORE is required. Simultaneously, licensees will be responsible for the communication with the Central Registry of Occupational Exposures (CROE). CROE is created [5] by the State Office for Nuclear Safety (SONS) for the registration and evaluation of ORE on the national level.

The evaluation of ORE is carried out by five approved dosimetric services in CZ - National Personal Dosimetric Service, two Dosimetric Services of NPP at Dukovany and Temelin, Dosimetric Service of Uranium Industry and Dosimetric Service of Nuclear Research Institute in REZ. Recently these services control about 20,000 workers (Table 1).

METHODOLOGY

The film dosemeter is a basic type of dosimeter for the evaluation of ORE from external irradiation. All services proceed from filtration analysis for the estimation of dosimetric quantities. A qualitative estimate of contamination of worker and of individual (surface and penetrating) dose equivalents (with recording level 0.2 mSv for one month period of control)

are made by the evaluation of dosemeter. The estimated value of individual penetrating dose equivalent H_p is considered to be an acceptable (usually overestimating) estimate of the effective dose equivalent from external irradiation. Further dosimetric investigations (from H_p > 1.25 mSv) are carried out, considering the presumed geometry of irradiation , if an overexposure is suspected.

Since 1976 personal neutron dosemeters (based on plastic track detectors Malyar with the couple of fission foils, enriched uranium and natural thorium [6]) are used (recording level H_p > 1.25 mSv for month as well as for year control periods is used at the selected workplaces (about 600 workers) with significant neutron sources (industrial radiographers, well-loggers, etc.).

Some workplaces, especially NPP's, are equipped with additional, operational dosimeters (mostly TLD, now different types of electronic detectors) for evaluation of ORE in the course of special tasks, somewhere finger TLD are used.

As far as the internal contamination is concerned, the licensees are responsible for securing of monitoring of workers or workplaces and recording of the results. When individual monitoring is inappropriate, inadequate or not feasible, ORE are assessed on the base of the monitoring of the workplaces, information about given location and duration of exposure. CROE registers ORE from internal contamination exceeding the recording level (0.1 of annual limit of intake).

The evaluation of annual ORE and their recording in dependence on different factors (occupational group, working time, sex, age, etc.) are introduced into practice of dosimetric services a fter a revision of their methodology of ORE evaluation (implementation of the new ICRP and ICRU recommendation) during 1990-91 years.

OVEREXPOSURES RECORDING

Record keeping of overexposures is based on two systems:

- if there is suspicion on overexposure at a workplace, licensee is obligated to estimate of the severity of given event; an investigation shall be conducted with the aim to identify the circumstances and assess and record the relevant doses and their distribution in the body. In the case that the annual limit of the effective dose equivalent $H_{E,L}$ (for internal contamination - 1/10 of ALI), the event must be reported to CROE. For serious incident (with possible medical consequences) National Radiation Protection Institute in Prague is involved in evaluation of incident consequences;
- the second system is based on co-operation of CROE with dosimetric services; these services report directly to CROE immediately after the evaluation of dosimeters for given control period any excess of the $H_{E,L}$ (or determined investigation levels). Identified events are then investigated by inspectors of SONS directly at the workplace where the event occurred.

183 persons were involved in 65 minor accidents and in indicated events detected in CZ (and former CSSR) during the 1954 - 1994 years. 102 of them were exposed by external and 81 by internal radiation. Only 15 from these events have led into health consequences (dermatitis with skin defects, cataracts). In some events surgical interventions - amputation of fingers, removal of a local deposit were used. In the period 1975-94 years 107 dosemeter readings were reported by the second system as exceeding the value 50 mSv ($H_{E,L}$). Following reinvestigation it appeared that in most cases the primary evaluation was falsely positive; it was concluded that only in 45 of indicated findings, the values of $H_{E,L}$ were exceeded.

DISTRIBUTIONS OF ORE, CONCLUSIONS

The long term evaluation of ORE in CZ covering period 1975-1990 has undertaken in 1993 year [6]. Several basic characteristics are illustrated in Table 1.

Based on mentioned analysis and recent results of ORE evaluation, we can present that:

- the average values and trends of individual effective dose equivalent (H_E) are comparable with those in developed countries (only 20-30 % of values ORE were higher than 0.2 mSv, the recording level),
- the significant part of the annual collective dose equivalent (S) arises form operations in the nuclear industry, when values of H_E from radon, its daughters and from ore dust constituted about (60-80)% of its total value in the uranium industry (UI);
- average individual values of H_E in NPP Dukovany are lower or comparable with those in the other NPP s PWR - type. In period 1985-94 (operational time of NPP Dukovany) the H_E values have been registered in the interval (0.55 - 0.82) mSv.

Table 1: The numbers, sex and age distributions of monitored workers in the Czech Republic.

NUMBER OF WORKERS (THS)	YEAR				
	1 975	1 980	1 985	1 990	1 993
NPP				1.5	2.6
URANIUM INDUSTRY	11.2	10.3	10.3	7.9	3.3
INDUSTRY	3.5	5.4	6.8	7.8	5.7
MEDICINE	3.7	5.6	7.1	8.2	11.1

	AGE DISTRIBUTION [YEAR]					
	< 25	25 - 30	30 - 40	40 - 50	50 - 60	> 60
NUMBER OF WORKERS (N) [THS] [1)]	1.3	1.6	4,4	5.7	2.8	0.9
COLL. EFF. DOSE EQUIVALENT (S)[1)][Sv]	0.5	1	3.2	4.4	2.3	0.5
S/N[1)] [mSv]	0.4	0.6	0.7	0.8	0.8	0.6

	DOSE EQUIVALENT DISTRIBUTION H_E [mSv]						
	< 0.2	0.2-1.0	1.0-2.0	2.0-10	10-20	20-40	> 40
NUMBER OF WORKERS N(THS)[1)]	9,66	3,93	1,61	1,37	0,09	0,02	0,01

GROUP	SEX DISTRIBUTION [RATIO M/W]
INDUSTRY	5,13
MEDICINE	0,54
RESEARCH	1,17

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Mass Optimization Studies of Gamma Shield Materials for Space Nuclear Reactors

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MAJOR SCIENTIFIC TOPIC NUMBER 4 (see page 7)

ABSTRACT (See instructions overleaf)

An optimization study of the total gamma shield mass requirements for a space nuclear power system is presented. The minimization of the mass of the gamma shield for a space nuclear power system is of interest because it represents a large fraction of the overall spacecraft mass, typically between 20% and 40%. Optimization of the shield mass can thus result in significant mass savings which translate into reductions in launch costs.

The gamma radiation source and reference dimensions were based on the Russian TOPAZ-II nuclear electric power-generating reactor. The potential shield materials analyzed were aluminum, steel, lead, tungsten, depleted uranium, lithium hydride, and concrete. The required shield masses were normalized to the lowest value, providing a "mass index" that was used as a basis of comparison for the different shield materials. A gamma source of 10^{16} γ/sec originating from a thin disk-source model of the reactor, provided a uniform beam of monoenergetic gamma rays incident on the cone shield. A value of 50 mSv/hr (5 mrem/hr) was chosen as the desired dose equivalent at the reference dose point. The computations were performed with QAD-CGGP, a three-dimensional, point-kernel gamma shielding code. Depleted uranium had a mass index of 1.0 and provided the required shielding with the lowest mass requirement. Mass indices between 1.2 and 1.7 were characteristic of tungsten and lead, making them potential substitutes for depleted uranium in the case of smaller reactor power levels.

REENTRAINMENT OF $^{239}PuO_2$ PARTICLES CAPTURED ON HEPA FILTER FIBERS

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ABSTRACT

The collection efficiency of the air filter is the most important parameter for removing of radioactive aerosols. At the same time, retention of the captured particles is very important for contamination control not only during use but also after use. Reentrainment of $^{239}PuO_2$ particles collected on HEPA filter fibers was investigated under various conditions of air flow direction, air flow rate, air flow pattern and dust loading. Activity median aerodynamic diameter (AMAD) of Pu aerosols was about 0.45μm, and the geometric standard deviation (GSD) was 2. Collected activity on the filters ranged from 10^7 to 10^8 Bq/m^2. Dispersal rate under nominal direction flow was measured as less than 2 x 10^{-7} /hr. Under the inverse direction flow, higher dispersal rates were observed, but dispersal basically decreased with increasing sampling run. High air flow rate increased the dispersion. There was no significant differences among air flow patterns, continuous flow, intermittent flow and mechanical shock flow. Dispersal rate strongly depended on the dust loading. For example, three times loading of atmospheric airborne particles against the initial pressure drop was found to increase the dispersal rate by over 300 times to 10^2 /hr at maximum. Even if experimental conditions were considered extremely severe, it makes us to take reentrainment from a high loading filter more seriously, since it indicates that there is a possibility of air contamination due to handling of the spent HEPA filter.

INTRODUCTION

Reentrainment of particles collected by the air filter as well as a filtration mechanism has been a very important subject. The recoil phenomenon of alpha-emitting nuclides such as plutonium(Pu) has been recognized to affect retention of alpha active particulate on filters. There are many papers on the recoil of radon progeny from a substrate from the perspective of basic science and in practical application. However, there are few studies on the recoil of Pu particles. Fleischer and Raabe noted the possibility that recoiling fragments of Pu particles due to alpha decay relates to dissolution in liquids[1]. McDowell et al. investigated the effects of recoil on penetration through HEPA filters and reported that alpha-emitting particulate matter does indeed penetrate high efficiency filter media much more effectively than nonradioactive or beta-gamma-active aerosols[2),3)]. However, Wen and Kasper investigated the kinetics of reentrainment from surfaces for nonradioactive particles and explained particle reentrainment as a result of a competition between adhesion forces and fluid dynamic lift forces[4)].

In our previous studies[5),6)] on HEPA filter, decontamination factors against Pu aerosol particles were almost the same or higher than the levels expected from experiments using nonradioactive aerosol particles, but these studies did not clarify retention efficiency. For filters in long-term use and spent filters containing such plutonium-containing dust, retention efficiency is very important along with collection efficiency. In this work, we focused on the retention of alpha active particulate, distinguishing from apparent retention with re-collection and net retention.

MATERIALS AND METHODS

Experiments were performed with filter media not in-place HEPA filters. The filters used in this work had an effective diameter of 37 mm. Basic collection performance was examined in our previous studies[6)]. Two types of test filters with and without dust loading were prepared prior to loading of radioactivity for dispersion experiments. Atmospheric dust was loaded up to 1.7 and 3.0 times the initial pressure drops of the filter. Radioactive source filters were prepared by collecting plutonium oxide ($^{239}PuO_2$) aerosol particles[5)] at an air flow face velocity of 7.8 mm/s, which corresponds to about one-third of the nominal flow velocity of HEPA filters. The particles were polydisperse aerosols with AMAD (Activity Median Aerodynamic Diameter) ranging from 0.4 to 0.5 μm and GSD (Geometric Standard Deviation) ranging from 1.9 to 2.1. Activity densities on the source filter surface ranged from 10^7 to 10^8 Bq/m^2.

Reentrainment of Pu particles captured on HEPA filter fibers was examined under dynamic conditions with air flow rather than static conditions. Source filters and

clean filters for sampling of the dispersed particles were mounted in a multistage filter holder as shown in Fig.1. Experimental parameters on air flow through the source filter consisted of two air flow directions, two air flow velocities and three air flow patterns. The air flow directions were a nominal direction from front to back and an inverse direction from back to front. The air flow velocities were a nominal velocity of 23 mm/s and high velocity of 1.2×10^2 mm/s. The air flow patterns were (1) continuous flow, (2) intermittent flow that stopped for five seconds per half minute and (3) mechanical shock flow by free fall from a height of 50 mm. Three runs of five minutes sampling were repeated every ten days so that nine samples were obtained for each source filter.

SSD for alpha counting and phoswich detector for LX-rays of ^{239}Pu counting were used for activity measurements of filter samples. The minimal detectable activity during 2.5×10^5 sec counting was 1.2×10^{-2} Bq.

RESULTS AND DISCUSSION

Effects of air flow directions through the source filter on reentrainment of Pu particles were examined by intermittent flow at high air flow velocity. In the case of nominal direction flow, no significant activities in sampling filters were detected except one measurement among total 27 sampling runs. The dispersal rate per hour was very small and was estimated to be 2.3×10^{-7}. However, dispersed activities were observed in most runs with inverse flow from back to front. This indicates that particle reentrainment with nominal direction flow is not denied. Dispersed particles would be re-collected by fibers before penetrating a filter. The maximal dispersal rate per hour reached 2.4×10^{-5}. Dispersal rates scattered among source filter samples and even among successional runs using the same samples. Overall tendency showed a decreasing trend with the sampling runs. Next, effects of air flow patterns through the source filter on reentrainment of Pu particles were examined with inverse direction flow at high air flow velocity. Three source filter samples for each air flow pattern, continuous flow, intermittent flow and mechanical shock flow, were used for dispersion experiments. As shown in Fig.2, the measured dispersal rates ranged from 4×10^{-5} to 10^{-7} /hr. There were no differences in dispersal rates among air flow patterns. Thirdly, effects of air flow rates through the source filter on reentrainment of Pu particles were examined with intermittent flow in an inverse direction. The dispersal

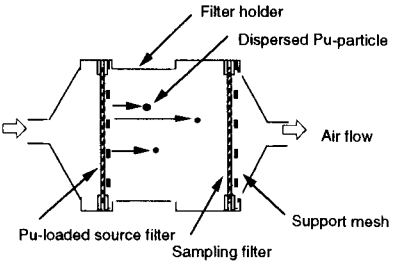

Fig.1 Structure of filter holder with multistage for sampling the dispersed Pu particles

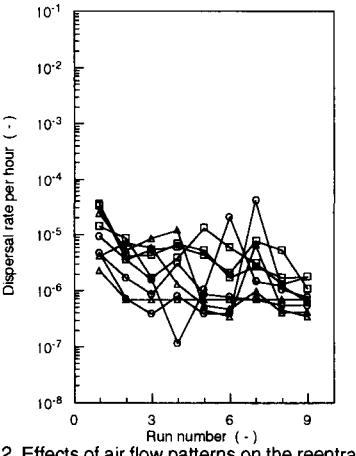

Fig.2 Effects of air flow patterns on the reentrainment of Pu particles at inverse flow of 1.2×10^2 mm/s (○: continuous flow, △: intermittent flow, □: mechanical shock flow)

rate depended on the air flow velocity. The measured dispersal rates at nominal velocity were smaller than those at high velocity. The maximal dispersal rate was 3.1×10^{-6} /hr and it was approximately one-tenth that at high velocity.

For source filters loaded with atmospheric aerosols, reentrainment was examined in the same way as for the pre-loaded source filters described above. The experiments were made in the inverse direction flow through the source filters at high flow velocity. Air flow patterns were continuous flow, intermittent flow and mechanical shock flow. As shown in Fig.3, high dispersal rates were observed for the source filters with dust loading. The maximal dispersal rates were 5.6×10^{-4} /hr with low dust loading and 1.0×10^{-2} /hr in high dust loading. These were 23 - 4.2×10^2 times higher than the rate for the filter without dust loading. There were no differences of the dispersal rates among the three air flow patterns. The dispersal rate decreased with increases in sampling run number. The dependency was

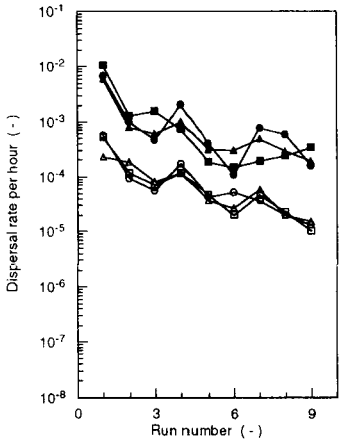

Fig.3 Effects of low and high dust loading on reentrainment of Pu particles at inverse flow of 1.2 x 10^2 mm/s (○ : continuous flow, △ : intermittent flow, □ : mechanical shock flow, open symbol: low loading, solid symbol: high loading)

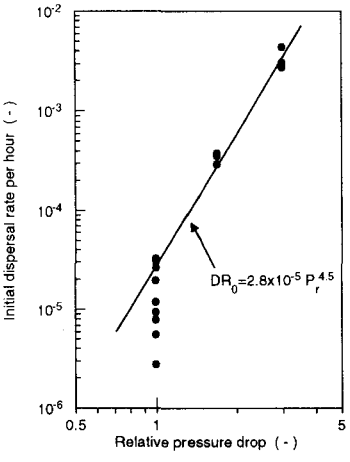

Fig.4 Dependency of the initial dispersal rate on the dust loading level of the source filter

relatively more clear than that for filters without dust loading. The fitted curves were expressed as an exponential function defined by

$$DR = a \exp(b N)$$

where DR is the dispersal rate per hour, N is the number of sampling runs, and a and b are the constants. b in the low and high loading filters were estimated as 0.36 and 0.34, respectively. The difference between these two was very small, so that factor b was assumed to be common (b = 0.35) for all experiments including those using filters without dust loading. Initial dispersal rate, DR_0, is defined by extrapolating to the point where the sampling run number is zero. The DR_0s for filters with low and high dust loading and those for filters without dust loading were estimated and plotted against the relative pressure drop, P_r, of the source filter to initial pressure, where the P_r is one of the indices indicating the dust loading levels. The DR_0 linearly increased with the P_r in a semi-logarithmic graph as shown in Fig.4. The relationship was expressed by

$$DR_0 = 2.8 \times 10^{-5} P_r^{4.5}.$$

For example, the dispersal rate at two times dust loading is calculated to have increased by 23 times. It was confirmed that the reentrainment was strongly affected by dust loading. The loaded dust particles play the role of a Pu carrier in radioactive particle reentrainment.

CONCLUSIONS

The reentrainment of submicron-sized Pu particles captured on HEPA filter fibers was experimentally investigated, and the results are summarized as follows:

(1) With nominal direction flow from front to back, the dispersal rate was estimated to be below 2.3 x 10^{-7} /hr at a high velocity of 1.2 x 10^2 mm/s. This dispersal rate was much lower than the maximal penetration of the HEPA filter with a minimal collection efficiency of 99.97 % for 0.3 μm sized particles. Therefore, it was concluded that Pu particles captured on filter fibers near the front surface hardly penetrate the filter.

(2) With inverse direction flow from back to front, reentrainment phenomena of Pu particles were observed in most experiments. The dispersal rate increased with air flow velocity and dust loading to the order of 1 x 10^2 /hr. In handling of a spent filter contaminated by Pu, marked attention should be paid to preventing air contamination.

(3) Dispersal rates were not constant in the succeeding sampling runs and decreased with large scattering. The fitted curves were expressed as an exponential function for all experiments.

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IMPROVEMENT OF RADON SUPPRESSING AEROSOL MONITORING TECHNIQUES

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ABSTRACT

Subject of this presentation is the improvement of the effectiveness of different radon-suppressing techniques for Alpha aerosol monitoring in working areas.

Due to the very high dose factors of Uranium and Transuranium Aerosols the limit of detection for airborne concentrations derived from statistical considerations has to be very low, i. e. about 100 mBq/m³ after a few hours.

ABPD (Alpha/Beta Pseudo Coincidence Discrimination)- and AERD (Alpha Energy Range Discrimination)- techniques can reach that goal when the Alpha emitting radon decay products do not change their equilibrium. When measuring continuously in real conditions, this is disturbed by systematic errors, such as the influence of Po 214, which sometimes changes its equilibrium with the suppressible Po 218.

As a result of our investigation we changed the ABPD- and the AERD- method by stopping the air flow through the measuring filter for a short time and watching the decay and build up of Po 214 by means of an on-line least squares fit.

INTRODUCTION

The implementation of the newer findings about the high biological effectiveness of Alpha-radiation, mainly of Uranium and Thorium, resulted in lower limits of intake and lower limits of airborne radioactivity concentration.

The permanent presence of an airborne Alpha concentration of 0.2 Bq/m³ would be equivalent to the yearly limit of intake, as it is given by the German Radiation Protection Ordinance of 1989. Accordingly high are the demands on aerosol-monitoring, which should detect the presence of such an airborne Alpha concentration as early as possible. In that case continuous monitoring is the appropriate method.

When monitoring Aerosols, it is an unwelcome side effect that the radon-followers give a varying contribution to the Beta-signal as well as to the Alpha-signal. This varying contribution presents the essential restriction to the lower limits of detection, mainly when measuring continuously.

Apart from continuous Alpha-spectrometry the most important measuring techniques to minimise such a restriction are the ABPD (Alpha/Beta Pseudo Coincidence Discrimination)-technique (1) and the AERD (Alpha Energy Range Discrimination)-technique (2). These methods can also be combined (3).

Both these techniques use the specific detectability of Po 214 of the Radon decay chain (resp. Po 212 of the Thoron decay chain). Po 214 (resp. Po 212) is representative for the natural activity (including e.g. Po 218 resp. Po 216), which can then be deducted from the total activity to get the artificial (Uranium, Transuranium) Alpha activity as a result.

This presentation was developed in the course of the research project: Room Air Monitoring of Uranium and Transuranium, supported by the Ministry of Environment (Bundesministerium für Umwelt) of the Federal Republic of Germany with the Project Number StSch 4055

The ABPD-technique uses the short lived succession of the decay of Bi 214 and Po 214 in the Radon decay chain (resp. of Bi 212 and Po 212 in the Thoron decay chain) whereas the AERD-technique takes advantage of the range of the Alphas of Po 214 (resp. Po 212), which is high compared to the case of long lived Alpha-Emitters, like Uranium or Transuranium nuclides.

The following simplified part of the nuclide chart explains it:

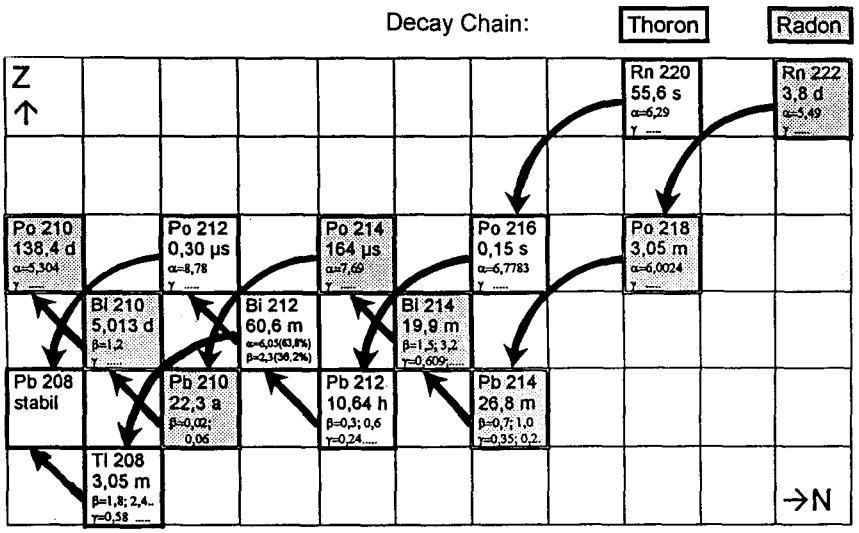

Figure 1: Extract of the Karlsruhe Nuclide Chart with the Radon and Thoron decay.

INTERFERENCES

The theoretical detection limits of these radon-suppressing techniques are about 0.2 Bq/m³ when that concentration is present for at least 1 hr in an atmosphere of 10 Bq/m³ Radon Gas in the state of equilibrium of the Radon daughters (4). When comparing these theoretical detection limits with the ones achievable in real cases, it shows that there are systematic interferences due to the disturbances of the equilibrium of the radon decay products. One example should explain this:

If there is a system for circulating and filtering air, that works optionally, the long-lived radon decay product Po 214, which is the basis for the compensation, is filtered out, while the short-lived Po 218 is rebuilding quickly. So switching on that system leads to an undercompensation of the Radon daughters, if the monitor's compensation was adjusted before - or, even worse - switching off that system leads to an overcompensation of the adjusted system.

It is obvious that similar effects appear, after doors of monitored rooms are opened ore closed.

SOLUTION

We have tried to separate the influence of Po 218 by stopping the air flow through the measuring filter of an ABPD-monitor regularly for a short time and watching the Alpha-signal. The part of the signal due to Po 218 falls exponentially with a half-life-time of 3.05 min. The count rate, collected every 10 seconds, is fitted by a least squares fit using a transputer system to get the Po 218 concentration. The same method,

but using the rebuild of Po 218 is also used, when the air flow is started again. The Po 214 concentration is known by the pseudo coincidence rate.

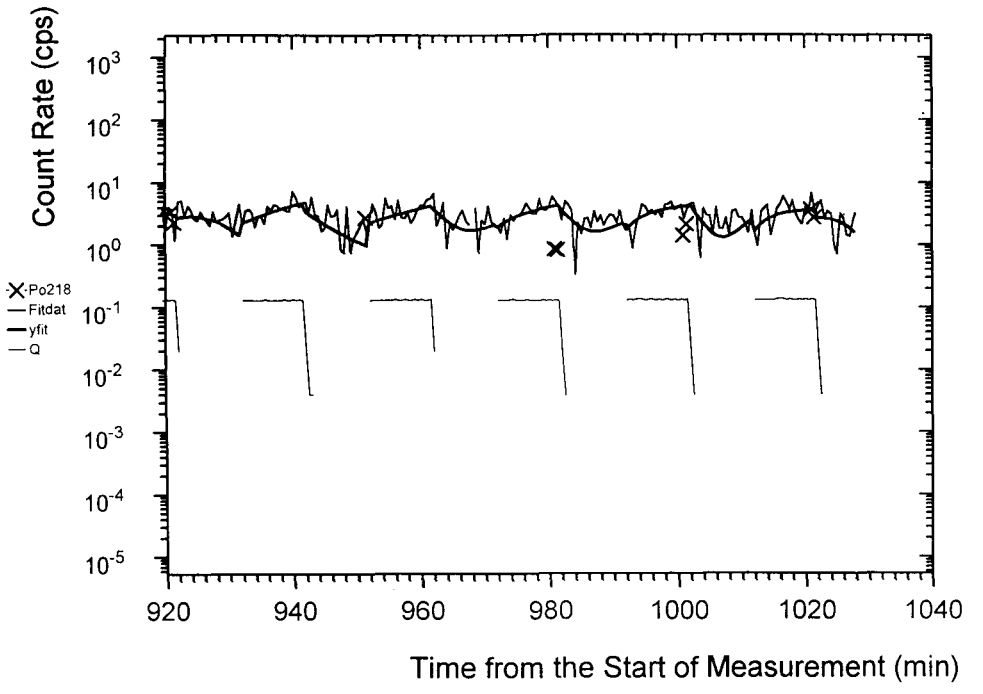

Figure 2: Results of the first on-line measurements with the extended ABPD-Monitor
The Po 214-compensated count-rate is fitted to get the equilibrium count-rate due to Po 218
The measured flow rate through the monitor is given as Q and is about 42 m³/h

With that system, working in intervals of 10 min for accumulation and stopping, the problem of wrong compensation is solved, but a contamination burst in the stopping time cannot be seen. It is planned to upgrade the system, so that it only stops the accumulation if a malcompensation or significant airborne contamination is obvious. In this case the system can decide by doing the Po 218-decay fit.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE RADIATION PROTECTION AT WORKPLACES. THE FRENCH INSTITUT CURIE EXPERIENCE 1958 - 1994.

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ABSTRACT (See instructions overleaf)

RADIATION PROTECTION AT WORKPLACES. THE FRENCH INSTITUT CURIE EXPERIENCE ; 1958 - 1994 - JM COSSET, B PERDEREAU, B. DUBRAY, F. CAMPANA, H. JAMMET
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From 1958 to 1994, 434 persons were referred to the radiopathology Unit of the Institut Curie - Paris, France -, for a suspicious or proven accidental irradiation.
In 250 cases (58 %), the accident occurred in **industrial facilities**, 32 patients being referred from foreign countries lacking specific radiopathology units.
In 138 cases (32 %), various **research laboratories** were involved. 11 cases were referred from abroad.
In 46 cases (10 %), the accident occurred in **medical facilities.** Therapeutic accidents or complications in cancer patients were excluded. No such cases were referred from other countries.

Detailed follow up is available for 420 cases (97 % of the total number).
In a majority of cases (309 ; 74 %) a complete survey, as well as operational and/or biological dosimetry actually demonstrated that the putative victim did **not** receive any significant irradiation.
In 26 cases (6 %) a very low-dose localized or whole body irradiation did not warrant any treatment, and the patients were only assigned to regular follow-up.
85 patients (20 %) presented with some type of complications or sequelae (including two lethal cases).
Details about the type and severity of the sequelae will be presented.

STRATEGY FOR THE ASSESSMENT OF THE RADIATION EXPOSURE AT WORKPLACES DUE TO RADON AND RADON DECAY PRODUCTS

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INTRODUCTION

According to the future EU basic standards the exposures to radon decay products have to be taken into consideration. Therefore it is necessary to investigate

* which working operation could be connected with an increased exposure to radon decay products,
* which workplaces have to be considered,
* how many employees could be afflicted,
* which methods should be applied for the investigation and - as for as necessary - for supervision.

On the basis of the current radiation protection legislation in the new Federal Lands working operations which are connected with radon exposures have already been monitored. For instance, in 1994 3095 persons were supervised, 2422 of them were engaged in clean-up operations at Wismut facilities (including underground work) and 673 at non-uranium mines and facilities (conventional mining), water treatment plants, show caves and spas /1/. The extension of the monitoring to similar workplaces in the old Federal Lands is going to discussed in implementation of the EU standards.
Beyond it further workplaces affected by radon and radon decay products exist. Therefore it is helpful to get an overview about the kinds of workplaces and the number of afflicted employees.

DIMENSION OF THE PROBLEM

In regions with increased radon potential especially in the mining regions of East German Federal Lands measurements of the indoor radon concentration with an exposure time of 24 hours were conducted from 1991 to 1993. 2447 institutions participated in these investigations and more than 8000 measurements were made there. The measurements were conducted in 716 producing firms (especially small and middle-class institutions which are typically for the investigation regions), 237 education and research departments, 525 social institutions, 651 public utilities and 318 shops, banks etc..
The investigations included a broad spectrum of working rooms in different institutions like conventional bureaus, lecture rooms, production rooms etc..
Although the information by the voluntary participants about the realistic exposure conditions were often insufficient, the results are useful for an overview assessment. Table 1 shows the evaluation of measured results related to typical workplaces. It should be taken into account that these results represent an overestimation of the long-term situation because short-term measurements are made under stringent conditions (closed windows and doors). It has been shown that factors between 4 and 10 (ratio of short-term against long-term measurents in the range of higher radon concentrations) observed at dwelling measurements. They will be used for the further analysis /2/.

Because the aim of the screening measurements was the area-wide analysis of the situation in regions with a high radon potential, only a rough characterization of the investigated rooms was necessary, particularly there were no influence on their selection by the institutions.

Therefore the number of "other" is high, a majority of these rooms has occupancy times below 3 - 5 hours per day.
Against the expectation the results show that production rooms can not be neglected in the investigations.

Table 1: Results of short-term measurements of the radon concentration in radon potential regions at typical workplaces independent of the institution /3/

Using of rooms	Number of investigations	Average	Maximum	Median	Share of workplaces with short-term radon concentration above	
					4000 Bq/m³	10000 Bq/m³
		Bq/m³	Bq/m³	Bq/m³	%	%
office	1246	390	79000	110	1,4	0,5
production	696	430	13000	120	1,6	0,7
store	719	460	26000	130	1,9	0,1
other*	4002	700	170000	140	2,5	0,7

* class rooms, waiting-rooms, shops, workshops, engineering rooms etc.

To estimate the number of afflicted employees, it was assumed that
1. the percentage of investigated workplaces (rooms) with higher radon concentrations can be transferred on the number of employees who are working in these rooms and
2. only about 10% of the German area have a higher radon potential and the distribution of employees is uniformly.

Outgoing from these assumptions and measured results, the information of the Statistisches Bundesamt Wiesbaden on the total number of employees in Germany (36 mill.) and on the distribution of occupation, it can be estimated that nearly 20000 up to 60000 employees are working at places of different categories with long-term radon concentrations higher than 1000 Bq/m³, recommended by the IAEA Basic Safety Standards for a chronic exposure at workplaces. This rough estimation justifies and requires a detailed investigation of the relevance of radon exposure at these types of workplaces.

APPLICATION OF AN ACTION LEVEL

The ICRP in the publication 65 as well as the IAEA in its Basic Safety Standards recommend to establish an action level for workplaces. If the long-term radon concentration exceeds the action level, remedial measures should be initiated to reduce the radon exposure.
In the international literature there was a consent to a complex evaluation of radon concentrations in institutions (schools, stores, hospitals hotels, kindergarten etc.). However, in recent British and Finnish publications the individual workplaces or kind of workplaces in an institution, respectively, are considered /3/. This objective of the analysis of radon exposures at workplaces seems to be supported by our measurements at different kinds of workplaces and determines further investigations.
According to the current standard of measuring techniques, measurements of the annual average of radon concentration at all workplaces could be done effectively only by means of

integrating methods. However, these methods imply an overestimation of the real exposure, because the measurements result in a mean radon concentration for 24 hours but the time, in which the worker is exposed, is lower - normally 8 hours. That means an unjustified high expense of remediation or number of employees to be included into a supervision.

A first investigation shows that the parameters working time, daytime and using behaviour connected with ventilation rate are the main parameters requiring the detailed study of typical workplaces. The results of radon concentration measurements at some different kinds of workplaces (schools, police stations etc.) show deviations of mean working time exposure and integrated 24 hour exposure hitherto by factors of two to five. One example is a teacher room in a primary school:

daily average (24 hours):	290 Bq/m^3
working time average (7 a.m. to 4 p.m.):	120 Bq/m^3
night time average (4 p.m. to 7 a.m.):	315 Bq/m^3

Taking into account the variation of the above mentioned parameters, it can be assumed that even factors >10 are possible caused by the individual behaviour at the workplace. Because at present only long-term integrating radon measurements are feasible for broad investigation the methods should be adopted to ensure a realistic assessment of the radon exposure at workplaces.

A classification of workplaces into groups of similar features may be useful to estimate the ratio between working time average and daily average of radon concentration. The integral measured exposure can then be adjusted to a real exposure by correction factors which are typical for these groups. Existing regulations related to dusts, vapours, chemical noxious substances and air-conditioning will be taken into account, too, and their relevance to the radiation exposure due to radon and radon decay products is investigated.

If the approach to estimate the real working time exposure due to radon by correction factors on the basis of long-term measurements for classified workplaces is unsuitably then specific measuring methods have to be developed to monitor the radon concentration during the working time for a multitude of working places.

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Internal BfS-report (in preparation)

RADIATION ENVIROMENTAL INSPECTION IN RUSSIA

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The Serpukhov region is situated to the south of Moscow. The local extraction and production of building materials in the Serpukhov region are produced in considerable amounts.Besides that some quantity of building materials and construction units are imported to Serpukhov from other regions. According to the geological data the mean radon gas concentrations in soil are estimated as 20-40 kBq/m , but the maximum radon concentration reaches 1000 kBq/m .Developed building industry and geological peculiarities of the region make it necessary to carry out radiation control during all stages of building production. Durin years 1993-1994 in Serpukhov we controled region factors at the following objects: 19 schools, 37 kindergartens, 3 technical colleges. 11 many-storey houses, 6 sand-pits, 3 hlants of building materials. The main gavva- and radon gas control measurements were done to nation regulation [1], but in the special cases we took into account the methodical data [2] and recomendations of Institute of Radiation Hygiene (St.Petersburg). The additional radiation factors of alfa-beta contamination were measured for some rooms at schools and in the kindergartens.

The following devices have been used:

- DRG-01T, SRP-68 - for the internal gammas;
- RGA-01T, RGG-01T - for the radon gas;
- NC-302B, BDZA - for alfa contamination
- MKS-01R, NC-302B - for beta contamination;
- AMA-03F analyzer and DGDK-80B semiconductive detector for gamma spectrometry.

The summary results of our inspections are following:

1. The natural radionuclides in raw materials and building constractions. The specific concentration of Ra, Th and K-40 were measured by gamma-spectrometry method. The values of effective radioactivities Aeff were calculated according to [1] and presented in Table 1.
2. Equivalent equlibrium specific radioactivity of radon gas in indoor air A .During radiation inspections the momentary and dayly integrated values of A were meas,,ured by filtration (paper filters) and sorbtion (carbon co lumns)methods. The summar results are presented in Table 2.
3. Gamma dose rate (P).The gamma dose rates were measured inside each room of all buildings. The value of gamma dose rate lay in interval 7...33 R/h for all cases. Summary results of radiation measurements are given in Table 3 (the gamma and cosmic background not subtracted).

Table 1. Comparison Charecteristic of Building Materials

Site of sample-taking	Staff	Aeff,Bq/kg
Sand-pit "Syanovo",Serpukhov	sand	15
Sand-pit "Oka", Serpukhov	sand	32
Sand-pit "Oka", Serpukhov	sand+gravel mixture	90
Sand-pit "Serpukhov # 9"	sand	110
Sand-pit "Serpukhov # 1"	gravel	310
Sand-pit "Serpukhov # 1"	road metal	170
Sand-pit "Kuzmitshevo",Serpukhov	sand	88
Sand-pit "Kuzmitshevo",Serpukhov	road metal	260
Sand-pit "Dashkovka",Serpukhov	clay(fit for building)	280
Burnt Clay Plant,Serpukhov	burnt clay	300
Sand-pit of Brick-making Plant,Serpukhov	clay	170
Brick-making Plant,Serpukhov	brick	190
Heat and Power Plant, Kashira	ashes	200
Krivoy Rog (Ukraine)	road metal	544
Karelia	road metal	755
Ignatpol (Ucraine)	road metal	459

Table 2.Distribution of Number of Objects due to A Range of A , Bq/m

Floor	0...40	40...100	100-150	150-200
3-d floor	86%	14%	-	-
2-d floor	94%	6%	-	-
1-st floor	67%	28%	4%	1%
basement	49%	30%	18%	3%

Table 3. Distribution of number of objects due to p .

Object	No rooms with p >20 R/h	Rooms with p : 20<p <30 R/h	p > 30 R/h
Schools,colleges	79%	21%	-
Kindergartens	74%	22%	4%
Many-storeyed houses	80%	20%	-

On the basis of obtained data we conclude:

1. The products of local building industry correspond to 1-st class of radiation quality [1] (Aeff < 370 Bq/kg), this is confirmed by our annual sertification measurements. The imported raw building materials correspond to 2nd or even 3rd class of radiation quality, what depends on location of sand-pits.
2. On the factor of external gamma radiation all investigated objects correspond to "radiation free" objects. Nevertheless all new buildings must be subjected to radiation inspection because local building industry uses the imported raw materials and building products.
3. n the factor of radon gas in indoor air the radiation situation is more or less normal for all investigated objects. But some rooms in the kindergartens show the radon gas consentrations in indoor air close to thereshold levels. In these cases we recomended to take measures directed to decreasing the radon gas concentrations in indoor air

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April 14-19,1996
Vienna, Austria

17 [illegible] 1995

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Author 10635
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Mini-Presentation

PAPER TITLE Occupational Exposure Assessment in China

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MAJOR SCIENTIFIC TOPIC NUMBER (see page 7)

ABSTRACT (See instructions overleaf)

This paper describes the occupational exposure assessment in China, including nuclear industry, medical dignosis staff and undergroun mineers in coal and nonferrous metal industry. The collective effectiv dose are 1.0E+2 man·Sv·a^{-1}from nuclear industry which is 0.17 percent of total collective effective dose, 2.2E+2 man·Sv·a^{-1}from application of nuclear and radio-technology--0.37 percent of total collective effective dose, 5.0E+4 man·Sv·a^{-1}from underground coal-mines--85.15 percent of total collective effective dose, and 8.4E+3 man·Sv·a^{-1}from underground nonferrous-mines--14.3 percent of total collective effective dose. The total collective effective dose is 5.872E+4 man·Sv·a^{-1}. The collective effective dose from non-nuclear activities reaches 99.45% of total collective effective dose, i.e., non-nuclear activities is the largest contributor of total collective effective dose.

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Vienna, Austria

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PAPER TITLE A Radiation Protection Perspective on the 1996 Revision of the IAEA Regulations for the Safe Transport of Radioactive Material

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MAJOR SCIENTIFIC TOPIC NUMBER 5.7. (see page 7)

ABSTRACT (See instructions overleaf)

The IAEA Regulations for the Safe Transport of Radioactive Material, Safety Series No. 6, serve as the basis for worldwide safety requirements and are adopted by Member States as well as international transport organizations. In order to ensure that the regulations remain up to date with the latest technological developments and radiation protection principles, they undergo a comprehensive review and revision approximately every ten years. The 1996 revision of the Transport Regulations is nearing completion and several significant radiation protection issues were addressed during the revision. These issues, such as the incorporation of principles of the IAEA Basic Safety Standards, will be examined in detail in the full paper.

Effect of U.V. and Gamma radiation on Rn222 permeation through polyvinyl chloride (PVC). Application to the packaging of radium sources for the purpose of storage.

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ABSTRACT

Mining of uranium and thorium, as well as the use of radioactive sources in radiotherapy, especially during the first half of the 20th century, have yielded waste with a various concentrations in long-lived radionucleides (particularly radium). This waste cannot be stored in traditional storage sites which will return to the public domain after 300 years, due to, on the one hand, the radium period (1,620 years) and, on the other hand, the build-up of radon, its gaseous daughter.

One solution, in order to optimise the packaging and storage of such products, could be to use successive barriers, made of polymer membranes, intended to limit radon emission. Laboratory tests have shown that polyvinyl chloride (PVC) greatly reduces radon emission from a radium source. However one should take into account the damage of the polymer with the time, due to the radioactive waste storage itself over long periods of time. Therefore, in order to check the durability of such barriers, PVC samples have been subjected to different accelerated ageing processes by exposure to ultra-violet (UV) radiations or gamma rays.

We have determined the effect of such radiations on the samples. It seems that the UV irradiation causes structural changes in the PVC as a function of the duration of irradiation . This leads first to an increase in the efficiency of the polymer as a "radon barrier", by a factor of 10 then to a long-term weakness. The gamma irradiation (dose rate : 1.05 $kGy.h^{-1}$, dose : 0.71 MGy) has the same effect, but increases the efficiency of PVC as a radon barrier only by a factor 2.7. We have observed the same behaviour for samples of PVC irradiated with a dose rate 4000 time smaller (dose rate 27×10^{-2} $Gy.h^{-1}$; dose 4.6×10^{3} Gy).

1. INTRODUCTION

Today, radon is considered as the major source of radiological exposure of man to natural radiation (1).
In order to limit the radon emission from radium-waste packaging, and comply with the different regulations in the storage facility, specific packagings should include intermediary enclosures, such as polymer membranes used as radon barriers.
This work consists of studying a polyvinyl chloride membrane (a polymer often used in the nuclear industry), and in evaluating the effect of different radiations (UV and gamma) on the ageing of this polymer.

2. EXPERIMENTAL STUDY OF THE EFFECT OF UV AND GAMMA RADIATION ON THE TRANSFER OF Rn222 THROUGH PVC

2.1 UV radiation

2.1.1. Principle

We are studying radon transport in PVC membranes which have been exposed to UV radiation over different periods. After irradiation, these membranes are arranged in a stainless steel chamber (2) included in an experimental system.
We follow continuously the evolution of radon concentration in the downstream part of the system (if we refer to the membrane) due to the transfer through the membrane of gas present in the upstream part of the system at constant concentration. All the experiments are carried out at 20°C, at atmospheric pressure, with a zero differential pressure between the downstream and the upper stream of the membrane.

2.1.2 Results

In order to qualify the radon transport phenomenon inside the polymer, we have calculated the permeation factor by modelling the experimental evolution of radon activity concentration dowstream of the membrane (2). Figure 1 shows evolution of the permeation coefficient of radon in PVC versus UV dose. The permeation factor falls when the exposure period increases. In other words, the more irradiated is the membrane, the more Rn222-proof it is.

2.2 Gamma irradiation

2.2.1 Principle

We have developed two experimental device to study the effects of gamma irradiation on radon transfert through PVC.

- In the first configuration, we are following continuously radon transfer through PVC while irradiating the membrane with cobalt-60 gamma sources (dose rate 1.05 $kGy.h^{-1}$). The temperature of the experiment is contained between 20 and 25°C.
- In the second configuration, PVC membranes are first irradiated with cobalt-60 sources (3). The membranes are submitted to a gamma dose rate of 2,7 10^{-3} $Gy.h^{-1}$(4000 times smaller than in the first configuration) After irradiation, the radon transport through the gamma irradiated membranes is studied with the same experimental device that is presented in the chapter 2.1.1.

2.2.2. Results

In the first configuration, radon activity concentration downstream of the membrane evolves up to a maximum value, then decreases in successive steps to reach a minimum value.

In the second configuration, we observe an increased of the radon concentration until a steady state wich level is function of the dose taken by the membrane.

We have had more difficulties to make a model to describe the transport of radon for gamma irradiated membranes. However, we used the model proposed by Giridhar (4) to determined the radon permeation factor in PVC in these conditions. The permeation factors estimated by Giridhar law, in the two configurations, falls when the total gamma dose increase (figure 2).

3. INTERPRETATIONS

For the UV and gamma irradiations, we observed an increase of the PVC efficiency as a radon barrier, especially for UV-irradiated PVC in our experimental conditions.

Numerous studies on radiation effects in linear polymers have shown that the chemical changes which occurs are either crosslinking or degradations (5-6). It is well known that PVC crosslinked under irradiations. In order to explain our results (the decrease of the radon transport process in PVC in the UV or gamma irradiated PVC), we have to remember that the diffusion process of gas inside the polymer occurs by small displacements of side groups or segments of chains of the polymer. As a polymer crosslinks, the coordinated segment motion available for permeation is reduced, with a significant reduction in the diffusion coefficient. It is also likely that the extend of crosslinking zones inside the polymer structure, due to the effects of radiation on PVC, increases the effectiveness of the polymer as a radon barrier

4. CONCLUSION

The objectives of this preliminary study were to investigate the effect of UV and gamma radiation on the Rn222 transfer through PVC, with the purpose of knowing more about this polymer as a radon barrier, and its potential use for containment of radium-containing waste destined for storage in sites for long-lived radioactive waste. To observe the effect of UV radiations, we have exposed samples during different periods, then studied the radon transfer into the irradiated samples. We have been able to determine a permeation factor in each instance, with good repeatability, using a theoretical model. We have found a decrease by a factor of 10 in the permeation factor between an unexposed membrane and a membrane exposed in excess of 368 hours. To study the effect of gamma radiation, we have had two approaches:

-observed the radon transfer while irradiating the polymer at high gamma dose-rates,

-irradiated the membranes at weak gamma dose rate and study the radon transfer in irradiated polymers after irradiation

As for UV irradiation, we have found a decrease of the permeation factor, but by a factor of 2,7 only between a non irradiated membrane and a membrane having received a gamma dose of 0.71 MGy (dose rate: $1.05.kGy.h^{-1}$)

When the membrane is irradiated with a dose rate of 2.7 10^{-3} $Gy.h^{-1}$ (dose : 4600 Gy) the permeation factor decreases of a factor 1.8.

Future experiments should allow us to determine a limit in using the polymer as an effective radon barrier.

FIGURES

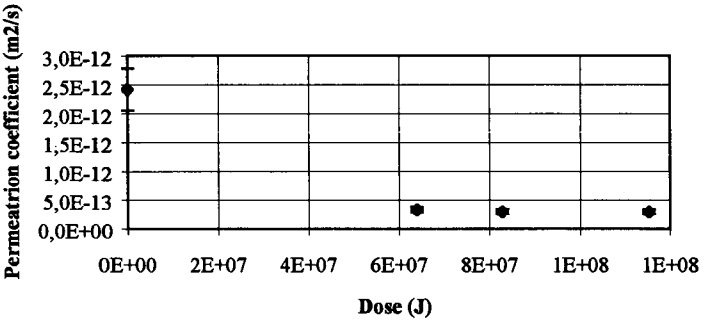

Figure1. Evolution of the permeation of radon in PVC versus UV dose

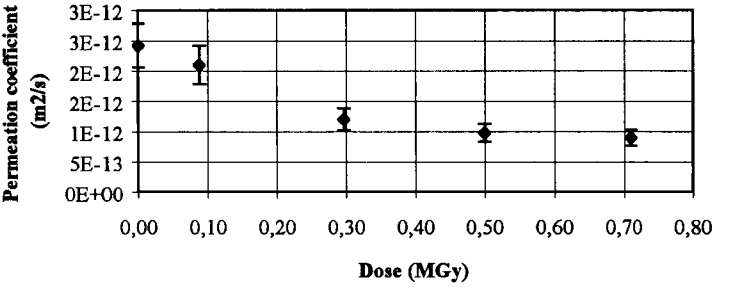

Figure 2. Evolution of the permeation of radon in PVC versus gamma dose (dose rate : 1,05 k Gy.h^{-1})

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QUALIFICATION OF RADIUM SOURCE PACKAGING BY THE MEASUREMENT OF THE RADON ESCAPE RATE

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ABSTRACT

The extensive use of radium sources in radiotherapy and in industry during the first half of the 20th century has yielded waste of different types (needles, tubes, luminescent paints etc.) with various concentrations of radium. These waste products have the characteristic to produce radon, a radioactive gas and its progeny radionuclides. These are of natural origin, but they still require precaution to be taken in order not to increase significantly the radiological risk to workers and to present and future populations.The management of these waste products implies the use of specific packaging that reduces as much as possible the radon transfer from the radium source to the storage facility, in order to comply with the different regulations.

So, the efficiency of the packaging as a radon barrier must be controlled. We have developed a test method to estimate the radon leakage of different containers by the measurement of radon degassing from the packaging. With this methodology, we can measure radon leakage rates as low as 10^{-5} Bq.s^{-1} .

INTRODUCTION

Some industries produce radium rich waste (treatment of ores to extract uranium, phosphates, rare earths). One of the special features of these products is that they give off a rare radioactive gas, radon, with variable emission rates depending on the intrinsic properties of the material (humidity, grain size, radium content etc.). The health risk from the radon is, in fact, due to its short life progeny which, once inhaled, can be fixed in the lungs and be harmful. That is why there are Annual Intake Limits (L.A.I.) for potentially exposed workers.
Radium containing waste can be packaged in container of different types before being stored on suitable sites. It is very difficult to design a long term, absolutely radon-tight package.
Therefore, a certain rate of leakage is tolerated, a function of the final storage conditions at site (particularly the presence or absence of ventilation).
That is why it is important to be able to quantify as precisely as possible the rate of radon leaking from the packaging of the radium waste, and why a strict methodology for testing the packaging has been developed.

1 - PRINCIPLE

To find the radon leakage rate of different type of packages, it is necessary to have what is called a "reference" radium source and to know its radon emission rate. This is put into the packaging to be tested. Everything is then put into a confinement chamber. The leakage rate of the packaging depends on the measurement of the radon collected in the chamber.

2 - DESCRIPTION OF THE TEST APPARATUS

The test apparatus comprises (figure 1):

- a stainless steel, perfectly sealed confinement chamber,
- a radon 222 measuring instruments (silicon detector, scintillation flask),
- a circuit sweeping the confinement chamber which ensures ventilation rate of 1 per hour.

3 - CHARACTERIZING THE LEAKAGE FROM RADIUM SOURCE PACKAGING

3.1 - Studying the "reference" source

In order to determine this radon emission rate of the "reference" radium source, it is put into the confinement chamber. The change in radon activity $A_{(t)}$ in it is measured. Under these conditions, the radon concentration follows a law of the following type:

$$A(t) = A_{\infty}(1 - e^{-\lambda t})$$

in which: $A(t)$: radon activity in time t ($Bq.m^{-3}$)
A_{∞} : radon activity reached at $t \rightarrow \infty$ ($Bq.m^{-3}$)
λ : radioactive constant of Rn 222 ($2.1\ 10^{-6}\ s^{-1}$)

Now,
$$A_{\infty} = \frac{E}{\lambda V}$$

in which E is the radon emission rate of the source ($Bq.s^{-1}$), and V is the volume of the confinement chamber. The "reference" radium source is preferably one of high activity ($\sim 10^7$ Bq) with a radon emission rate around 10 $Bq.s^{-1}$.

3.2 - Testing the packaging

The radium source, qualified in this away, is put into the packaging to be tested. This is then put into the confinement chamber which is closed and the circulation starts. As previously, the change in activity of the radon inside the chamber is continuously followed.

In general, the radon concentration in the confinement chamber changes as shown in Figure 2.

Let: A_v : volumetric activity of radon 222 ($Bq.m^{-3}$) in the chamber
V : confinement volume (m^3); V = [(volume of the chamber) - (volume of the packaging to be tested)]
λ : constant for decrease in radon 222 ($\lambda = 2.1\ 10^{-6}\ s^{-1}$)
t : time (s)
P_F : radon produced in the confinement chamber by leakage from the packaging ($Bq.s^{-1}$).

This can be written as follows:

$$\frac{dA_V(t)}{dt} = \frac{P_F}{V} - \lambda A_V$$

with $A_V = 0$ for $t = 0$. So that P_F can be estimated using the formule :

$$P_F = \frac{A_v(t)\lambda V}{1 - e^{-\lambda t}}$$

Thus, for a radon activity in the chamber of 100 $Bq.m^{-3}$, P_F is in the region of $2.5\ 10^{-5}$ $Bq.s^{-1}$.

4 - CONCLUSION

A methodology has been developed in order to quantify as precisely as possible the radon-tightness of containers intended for storing sources of radium 226. With this technique, it is possible to measure radon leakages as low as 10^{-5} $Bq.s^{-1}$. We can also estimate the reduction of the radon emission rate of the "reference" radium source due to the packaging.

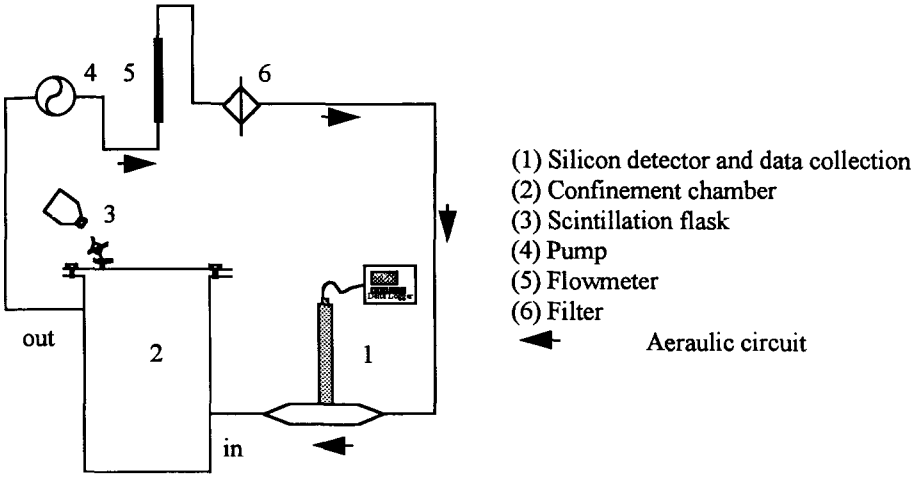

Figure 1 - Testing Apparatus

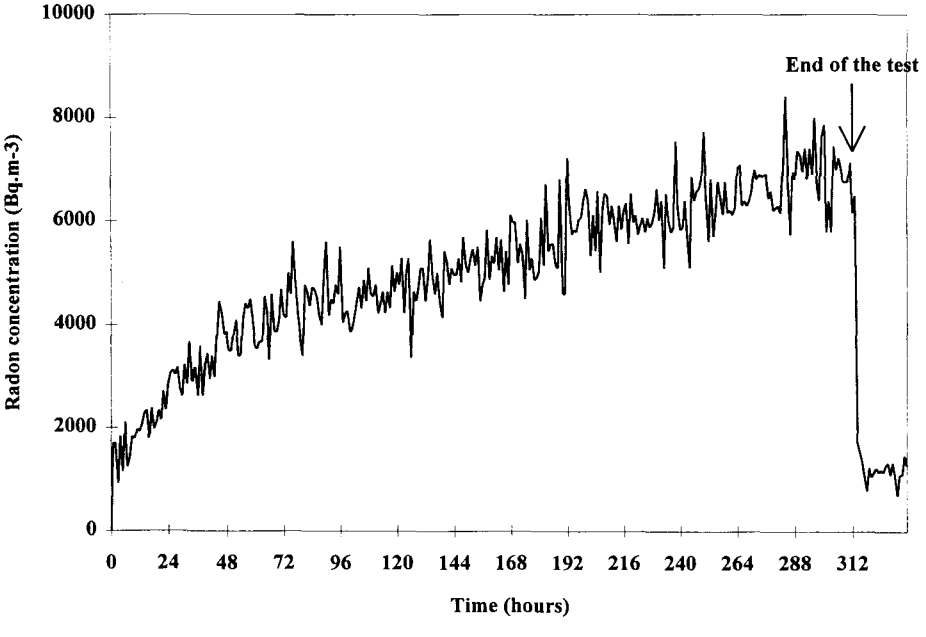

Figure 2 - Test of radium packaging: evolution of the radon concentration inside the confinement chamber due to the radon leakage from the packaging.

Effect of the New Neutron Quality Factor (ICRP 60) on the Transport of Radioactive Material

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The aim of this paper is to assess the effect of the new neutron quality factor, given by the publication ICRP 60, on the transport of radioactive material in France.

Neutron sources, mainly transuranians, are transported in containers with walls of a thickness allowing to meet the regulatory limits of equivalent dose rate on the package surface and at different distances. These limits will be more stringent when the new definition of the neutron quality factor defined by the publication ICRP 60 is implemented.

Faced with this regulatory evolution, packaging designers and carriers of radioactive material may resort to different solutions such as increase cask shielding and qualification of the new package model, limit the payload, or fit additionnal shielding on the vehicle.

Because the new regulatory limits can lead to a modification of package models, of the equipment used for the transport, and to more numerous transports of radioactive material, this paper proposes some examples of technical solutions, and gives indications about expected changes in the design and possible increase in the number of transports in France.

THE APPLICATION OF EXEMPTION VALUES TO THE TRANSPORT OF RADIOACTIVE MATERIALS

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INTRODUCTION

The principles and methods for establishing exemption values have been published by the European Commission DG XI (Radiation Protection 65) (1) and endorsed by the International Atomic Energy Agency (Basic Safety Standards) (2). These documents contain activitiy and activity concentration below which reporting is not required. The main objective of this study is to examine the relevance of the BSS exemption values to the international transport regulation.

The basic dose criteria in all feasible situations for exemption used by the BSS are 10 μSv/y for an individual effective dose in normal conditions (1 man-Sv/y for a collective effective dose), 1 mSv for an individual effective dose in accident conditions and 50 mSv for an individual dose to skin for both conditions. To derive the BSS exemption values, various exposure conditions were identified for normal and accident conditions leading to radionuclide specific exemption values expressed in term of activity and activity concentration.

Currently the transport regulation (3) includes a definition of radioactive material such that material with a specific activity below 70 kBq/kg is outside of the scope of the regulation. At the IAEA, the transport community has discussed the application of the BSS exemption values to transport and supported the dose criteria used, but was not convinced that all transport scenarios had been adequately addressed.

This paper[1] presents the calculations of exemption values for transport scenarios for a selection of twenty representative radionuclides. Those scenarios are derived from transport scenarios extracted from the BSS and from added specific transport scenarios. These results are compared to the corresponding BSS exemption values and to the 70 kBq/kg existing definition of the radioactive material of the transport regulation.

SCENARIOS AND METHODS

Several scenarios used in the BSS have been considered to be relevant for transport. In addition four basic scenarios were considered to be relevant for normal conditions of transport.

- A postman or courier delivers a package containing radioactive material to a laboratory or a hospital after having carried it during their delivery round.
- A driver, either an employee or member of the public, transports bulk material or packages in a truck or van.
- An employee or member of the public loads bulk material or packages into a truck or van.
- A member of the public travelling in an aircraft is exposed to radioactive materials being transported in the hold of the aircraft.

[1] This study has been supported by the European Commission DG XI.

Within the four basics scenarios, there are several sub-scenarios. They are described in the report « The Application of Exemption Values to the Transport of Radioactive Materials » (4).

A scenario based on the Q system (IAEA Safety Series n°7) (5), revised with the dose criterion of 10 μSv/y, was used. This scenario considered the main exposure pathways associated with accident situations of transport.

The methodology used to calculate exemption values was consistent with that used in the determination of the BSS values and is detailed in the report « The Application of Exemption Values to the Transport of Radioactive Materials » (4).

RESULTS AND DISCUSSION

The results for the transport specific scenarios were compared with the relevant ones used for the BSS exemption values to provide limiting values. One of the restrictive scenarios concerns the external exposure of a truck driver transporting 20 m^3 of bulk material for a duration of 400 hours per year. The restrictive activity values come from several scenarios but many of the results are similar. The values obtained from application of the Q-system are not limiting.

A comparison of the basic dose criteria of 10 μSv and the doses associated with the 70 kBq/kg is illustrated in Figure 1. The figure 1 reveals that a dose of up to 2.2 mSv could be expected with 70 kBq/kg of Co-60 and doses greater than 1 mSv could be expected with 70 kBq/kg of Ra-226, Th-232 and U-238. On the other hand, the 70 kBq/kg is too stringent for radionuclides such as C-14 or S-35. A single value of 70 kBq/kg for all radionuclides is not compatible with the results from the agreed transport scenarios. For the radionuclides listed in the BSS, more than two thirds of them have an exemption value less than 70kBq/kg or result in a dose greater than 10 μSv for an activity concentration of 70 kBq/kg.

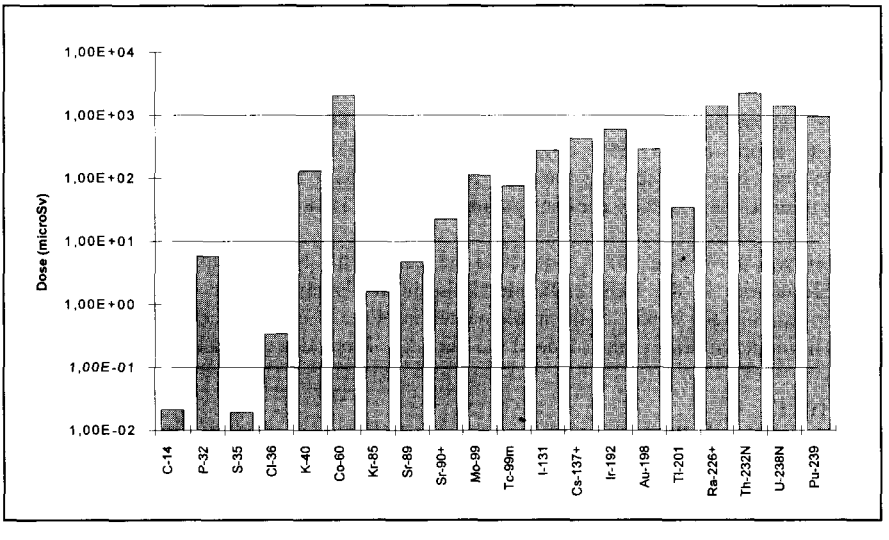
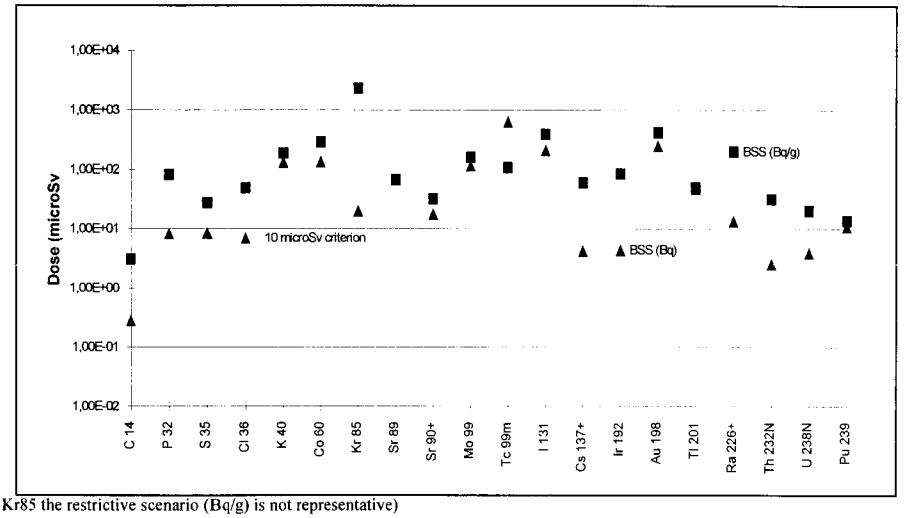


Figure 1 : Radiological impact of the 70 kBq/kg exemption level.

Figure 2 presents the radiological impact of the BSS exemption levels and shows that most are within one order of magnitude. Only Kr-85 is two orders of magnitude different. It can be argued that krypton is not transported as 20 m^3 of bulk material. For activity values the greatest difference is for Tc-99m, where a factor of 47 lower is observed.



(for Kr85 the restrictive scenario (Bq/g) is not representative)

Figure 2 : Radiological impact of the BSS exemption levels.

The transport specific exemption values are generally rather more restrictive than the BSS exemption values. However, these two sets of values are much closer to each other than the transport values are to the single value of 70 kBq/kg.

CONCLUSION

The activity concentrations and activities for the agreed twenty radionuclides in the transport scenarios are within two orders of magnitude of the BSS values. Such a range is insufficient to establish transport values different from those of the BSS. The generic BSS figures are therefore representative for transport.

A single exemption value is not consistent with the adopted dose criteria. Then, it seems reasonable to adopt, in the 1996 Edition of the international Transport Regulations, the BSS exemption values below which the transport regulation would not apply. In practice, it could be appropriate to restrict the activity per consignment because it is more operationally convenient.

This work was performed under contract to the EC and has been discussed by the international transport community at the IAEA. On the basis of this study, further developments could be envisaged to examine the impact of the implementation of the BSS exemption values to the transport practices. This could be done in the scope of the Radiation Protection Program.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Exclusion, Exemption and Clearance - A Clarification of Concepts

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MAJOR SCIENTIFIC TOPIC NUMBER 8.1. (see page 7)

ABSTRACT (See instructions overleaf)

The terms exclusion, exemption and clearance are defined and used in the International Basic Safety Standards for Protection Against Ionizing Radiation and for the Safety of Radiation Sources (BRSS). All of the terms are concerned with specifying the types of radiation sources which do not need to be regulated. However, the reasons that regulatory control is not needed are different in each case. Exclusion refers to sources which are not amenable to control, for example, cosmic radiation at ground level and most natural sources in the environment. Exemption and clearance apply to sources which would normally be expected to be under control but which present such a low risk that it would be a waste of resources to exercise control by regulatory means. Exemption applies to sources which never enter the regulatory control regime, while clearance applies to sources which are released from regulatory control. While the broad concepts are clear, it has become evident that some clarification and elaboration of their application is necessary.

EXPOSURE TO RADON IN CAVES AND ABANDONED MINES

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Abstract

The significance for health of exposure to radon daughters during leisure excursions in disused mines or caves is considered for visitors with interests in speleology, mineral collecting, mining history and youth training. General members of the public might also enter tunnels. Potential annual exposures based on radon measurements in various mines and caves are estimated and it is concluded that the annual exposure of individuals who undertake frequent and prolonged trips to underground systems might exceed 10^6 Bq h m^{-3}. Exposures to general members of the public are likely to be much lower.

The National Radiological Protection Board has developed proposals for a coherent and comprehensive scheme to promote dose limitation in a wide range of circumstances. These are described, with current initiatives on consultation with National Associations, local Government and many special interest groups.

1 Introduction

Most studies of radon daughter exposures are concerned with the exposures received in homes or at work in buildings or mines[1,2]. Significant potential also exists, however, for people who undertake leisure excursions into caves or disused mines[3], including, for example, people with interests in speleology, mineral collecting, mining history or youth training. Some of these activities can involve prolonged exposures in remote regions of caves, called wild caves, or in abandoned mine tunnels, where there is a potential for radon to reach very high concentrations. The potential for exposure has been investigated amongst UK groups, and a scheme has been developed to promote voluntary dose limitations during such activities.

2 Potential for exposure

The radon level in a cave or mine depends largely on the rate and direction in which air is flowing through the system. Broad trends in concentration can be assessed from a knowledge of how changes in temperature affect natural ventilation, and some systematic variations in radon concentrations can be explained, at least qualitatively[4]. Considerable uncertainty arises, however, in trying to predict radon levels under particular conditions, and the potential for exposure is most reliably estimated from measurements of radon levels and duration of exposure.

Radon levels have been investigated in various studies of UK mines and caves. The potential for exposure in showcaves of varying complexity has been evaluated both in the outer regions, where members of the public have access, and in the more remote locations which are often used for adventure caving by youth groups and others. Average radon levels in cave systems were up to 22 500 Bq m^{-3} , with a maximum concentration of 30 000 Bq m^{-3}. Extensive measurements have also been conducted by various caving groups during extensive and prolonged trips in several wild cave systems[5]. The measurements showed that average levels in systems could reach 35 000 Bq m^{-3} with a maximum concentration of 300 000 Bq m^{-3} at one location. The arithmetic mean values of all measurements in these two studies were 7 000 and 10 000 Bq m^{-3} respectively. A limited programme of measurements was conducted by NRPB in several abandoned mines that were known to be visited by casual visitors and those pursuing various hobbies. Concentrations were measured in the outermost 50 m of the tunnel and were found to be similar to the levels measured in natural systems. Although access to these tunnels is unrestricted, there is generally less scope for exploration of inner regions, since tunnels are often blocked or endangered by unprotected vertical shafts and broken rock. An exceptionally high concentration of 800 000 Bq m^{-3} was measured at one location about 20 m inside a tunnel entrance, but access was very difficult, through deep water, and it is probably very rarely visited. The results of the measurements are summarised in Table 1.

The duration of a trip underground is likely to vary considerably with the purpose of the visit. The groups who might enter systems are listed in Table 2. Members of the public who enter cave or mine tunnels out of curiosity are likely to spend only short and infrequent periods underground, perhaps an hour or so in a year, although

this might in some cases reach ten hours. The exposures accumulated by regular cavers and mineral collectors on the other hand might be substantial as a result of prolonged excursions underground or frequent entry into areas with high concentrations. Some of the highest exposures are likely to be received in wild caves or abandoned mines and might easily reach or exceed 100 hours annually for particularly enthusiastic individuals.

Radon levels in some mines or caves clearly reach a value at which a significant exposure could be accumulated during a single trip of several hours underground, and those who enter mines or caves regularly could accumulate excessive exposures. On the assumption that visitors are exposed to an average level of 10 000 Bq m^{-3} during trips underground, the annual exposure received by a regular and frequent visitor could reach 10^6 Bq h m^{-3} or more if trips are predominantly to systems with high levels. In the worst case, an exposure of this magnitude could be received during a trip of only a few hours. Exposures received by those who enter systems less often would be unlikely to exceed one-tenth of this value, although the ease of access to areas with high levels should be recognised.

3 Voluntary dose limitation

It is a general principle of radiological protection that radiation exposures should be as low as reasonably practicable and it is clearly desirable to apply this principle irrespective of the source of exposure. The first requirement, therefore, is to inform visitors about the potential for exposure so that the need for surveillance can be considered. Unfortunately, there are considerable practical difficulties in measuring radon levels in remote parts of underground systems, and information about the potential for exposure will often not be available or easy to obtain. In such circumstances, personal dosimetry can provide a suitable and convenient means of obtaining accurate estimates of dose[7]. Experience has been gained in the UK with the use of personal dosemeters for people who enter caves for leisure purposes.

Recognition of the significance of exposures that might be received in caves and mines prompted NRPB to propose a voluntary scheme for dose limitation. Essential elements of the scheme include advice that individual exposures should not exceed 10^6 Bq h m^3, that data on radon levels should be reviewed where available and that personal monitoring should be employed to assist people to remain within the annual guideline. It is also expected that information about radon levels and exposure limitation will be developed and made widely available, particularly amongst caving groups.

The scheme has been developed in consultation with the National Associations that represent and co-ordinate activities in caves and mines and it is expected that the various organisations involved will monitor and review exposure limitation practices and encourage improved procedures. Exposures received by members of the public who enter the outer parts of tunnel systems are likely to be very small; nevertheless, it is recognised that access into areas with high radon levels might be quite common in some localities. Local Authorities have certain responsibilities for health and safety in places with public access and are developing exposure limitation programmes for specific sites in their areas.

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Table 1 Summary of radon studies in UK caves and abandoned mines

	Type of Site		
Parameter	Showcaves	UK wild caves	Abandoned Mines
Number of systems	9	8	5
Number of measurements	44	350	17
Highest mean value in an individual system, Bq m^{-3}	22 500	35 000	50 000
Average for all sites, Bq m^{-3}	7 260	10 000	12 000
Median for all sites, Bq m^{-3}	6 200	11 300	1 600
Maximum concentration, Bq m^{-3}	30 000	310 000	80 000*

*Excludes exceptional value of 800 000

Table 2 Visitors to mines and caves by duration of exposure

	Duration of exposure	
	Generally short	Potentially long
Caves	Members of public Adventure cavers	Guides Explorers
Abandoned mines	Members of public	Mining historians Mineral collectors

Abstract for IRPA 9 (1996)

INTERNATIONAL REPORTING SYSTEM OF UNUSUAL EVENTS (RADEV)

Authors: P. Ortiz, Robert Jarrett

A questioning and learning attitude by managers and operators of radiation sources and devices used in medicine, industry, research and teaching, combined with information on accidents, their initiating events and contributing factors is an efficient way to drastically reduce the probability of further accidents. The list of contributing factors can be readily used to test the vulnerability of a given facility.

The number of reported events in a single country is in most cases insufficient to provide significant body of lessons in a reasonable time. Therefore, an compilation of accidents at international level, would allow all countries to benefit from the lessons from each one of them. Moreover, unusual events which did not culminate in an accident can build up a body of knowledge to avoid real accidents. Therefore, it should be possible to obtain information on cases (near misses) that otherwise are not published in any scientific journal and would never reach the interested community.

For this reason, an International Reporting System of Unusual Events has been set up by the IAEA. The reporting system will consist of a database with protected fields for those confidential data which should not be included in the reports. The goals of this project is to compile the information, to revise it periodically, to extract lessons, to disseminate them so that they this knowledge can fed into the training programmes of different professionals such as: regulatory authorities on radiation safety, health authorities, managers of facilities where radiation sources are used, manufacturers and maintenance personnel and operators of these facilities.

The sources of information are expected to be national authorities on radiation protection and ministries of health, along with professional associations such as radiotherapy, medical physics radiation protection, industrial gammagraphy (non destructive testing) and industrial irradiators among others.

THE NATIONAL CENTRAL REGISTRIES OF OCCUPATIONAL AND MEDICAL EXPOSURES IN THE CZECH REPUBLIC

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ABSTRACT - This paper describes recent situation in the Czech Republic concerning the registration and evaluation of occupational and medical radiation exposures. Since 1993 the creation of the Central (national) Registries of Occupational (CROE) and Medical Exposure (CRME) has been started. One of the main functions of these registries will be to provide statistics to guide policy making on a national basis. Authors give more detailed information on the structure of creating programs and discuss some actual arising problems.

INTRODUCTION

The actions leading to the creation of CROE and CRME were opened in 1993 by Radiation Hygiene Centre of National Institute of Public Health under the financial participation of the Ministry of Industry and Trade and Ministry of Health. In accordance with the changes in the structure of radiation protection in our republic this year, registries are now created in new National Institute of Radiation Protection which is supervised by State Office of Nuclear Safety (Fig.1).

Creating central registries, the Czech Republic (CZ) follows an international recommendation and world-wide trends in this field. CROE enable us to follow and control not only the individual doses of workers, but also to follow a radiation history of workers, estimate collective doses and time trends in different occupational categories, evaluate the efficiency of ALARA methods applied into the practice, help to find groups of workers significant from the point of view of radiation protection and collect data for epidemiological studies.

The survey and evaluation of medical radiation exposure (MRE) exists as a part of all reports of UNSCEAR from 1958. The aim of these world-wide studies is an estimation of world-wide dose, analyse of frequencies and dose distributions also in relation to age and sex of patients and determination of time trends in this area. Results of these studies help to find reasons for regional differences in doses for the same procedures, search the way for decreasing patient's exposure and to evaluate their effectiveness.

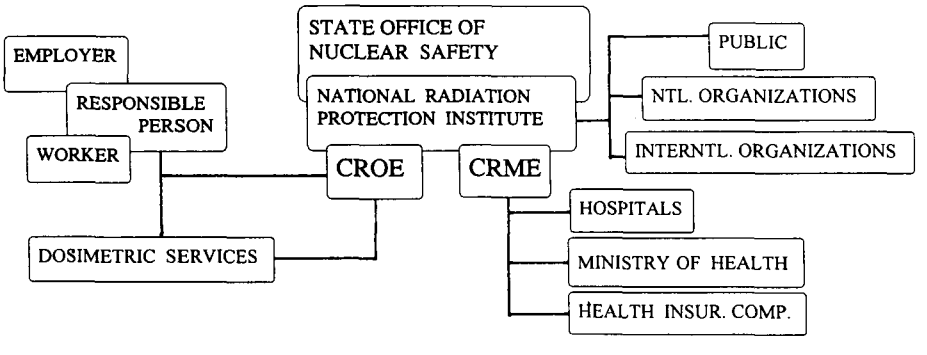

Fig.1: Organizational scheme of CROE and CRME in the Czech Republic

THE CENTRAL REGISTER OF OCCUPATIONAL RADIATION EXPOSURE IN CZ

The base of a registration and evaluation of OE is an individual monitoring of classified workers. The employer's duty, covered by the Regulation No.59 / 1972 in CZ, is to secure individual monitoring of radiation workers and record results of this monitoring. There is also legal duty of licensee to provide CROE with data in new prepared law on Peaceful Use of Nuclear Energy and Ionizing Radiation (e.g. " Atomic Act "). Dosimetric provision and evaluation of OE in CZ is the subject of a separate report to be presented at IRPA 9 Congress by Prouza , Petrová .

Databases of CROE contain detailed identification of employers including their activity categories (Tab.1) and the dose records for all monitored workers with details of their age , sex, occupational category (Tab.2), type of handling radiation source. The data will be reorganized annually and individual dose assessment will be maintain for the current year and previous five years. Earlier data will be archived. The system use special identification number for workers (birth number) and employer (random number) and all data are treated as confidential. Recently the registration cards serving for a contact between CROE and dosimetric services and employers are distributed. The cards will provide for CROE entrance data of all radiation workers in CZ and any changes in their registration. The database system is using ORACLE and operating in HP computers.

The contact with the International System on Occupational Exposure (ISOE/OECD) was opened up in 1994 and the created national system of ORE registration is built in the harmony with the recommendations and demands of this international system. This is a reason for instance for such detailed structure of occupational categories in NPP.

Tab.1: Employer's activity categories in CROE

1.0. health service (3)	6.0. defence
2.0. education, research	7.0. agriculture, food
3.0. energetics (3)	8.0. transport
4.0. general industry (4)	9.0. specialized facility (2)
5.0. uranium industry	

The start of operation of CROE is planned for 1996 . CROE will provide the new employer with summarized information of an individual's dose history, to guarantee a right annual dose calculation for workers with two or more employers, to provide statistics to national regulatory authorities and also another subjects interested in, according to CROE statute.

Tab.2: Occupational categories in CROE (the number of items for each category in parentheses)

1.0. Defectoscopy (2)	5.0.0. NPP (6)	6.0.0. Medicine (4)
2.0.Well logging (2)	5.1.0. Inspection, control (3)	6.1.0. Radiodiagnostics (4)
3.0.Radioisotopes (7)	5.2.0. Radiation protection (3)	6.2.0. Nuclear medicine (3)
4.0.Uranium industry (5)	5.3.0. Operation, maintenance (8)	6.3.0. Radiotherapy (5)
	5.4.0. Chemistry (3)	6.4.0. Veterinary medicine (1)
	5.5.0. Fuel handling (1)	7.0. The others specialized
	5.6.0. Waste disposal (1)	workers (4)

THE CENTRAL REGISTRY OF MEDICAL RADIATION EXPOSURE IN CZ

In CZ (10 mil.inhabitants) there are recently 350 radiodiagnostic, 52 nuclear medicine and 40 radiotherapy workplaces. There was performed about 9.5 mil radiodiagnostic, 250 ths

nuclear medicine and 22 ths radiotherapy procedures per year (numbers from Institute of Health Information Systems of Ministry of Health of CZ, in1993). This is big amount of data and it is impossible to sort all of them according to all demanded parameters . There are three main sources of data :

- the regular annual statistic survey of Ministry of Health which is managed by Institute of Health Information Systems (IHIS), but there is no possibility to sort patients according to their age and sex ,and there is also problem with clear definition of individual examinations,
- the organization of a national survey and use of information systems of individual health utilities - there is problem with co-operation and organization of this survey,
- the co-operation with health insurance companies - there are twenty companies in CZ now, but one of them - General Health Insurance Comp. (GHIC) is the biggest one which cover recently about 80 % of our population.

Database GHIC obtains birth number of each patient from that it is possible to determine age and sex of patient and there is unified list of all examination. CRME is operating only with a part of birth number of patient, the system will not operate with confidential personal data. In principle we would like to collect data by both way in co-operation with GHIC and with selected representatives health utilities.The data from IHIS will serve for better approximation of collected data to a national level.

GHIC provided CRME with first data which are related to the region with 1,2 mil. inhabitants for 1994. The first results for nuclear medicine procedures are showed in Tab.3 .

Tab.3: Sex and age distribution of the patients undergoing a nuclear medicine procedures in 1994 in the Czech Republic - first results of CRME

females	0-10	11-20	21-30	31-40	41-50	51-60	61-70	71-80	81-90	90+	total
bone	85	600	1417	1071	960	213	69	138	154	42	4749 (34%)
renal	16	156	418	363	353	235	132	467	710	216	3066 (22%)
thyroid	28	93	254	207	313	169	133	67	4	1	1269 (9%)
liver+gb	8	63	161	106	192	115	49	127	56	21	898 (6%)
brain	14	64	113	72	105	54	30	55	21	2	530 (4%)
lung	167	503	598	265	288	90	42	44	18	9	2024 (15%)
heart	3	14	64	73	163	100	33	66	35	5	556 (4%)
tomo sci	4	20	120	120	181	60	13	10	0	3	531 (4%)
other	1	18	43	33	40	36	14	21	13	0	219 (1.5%)
total	326	1 531	3 188	2 310	2 595	1 072	515	995	1 011	299	13 842 100 %

males	0-10	11-20	21-30	31-40	41-50	51-60	61-70	71-80	81-90	90+	total
bone	64	402	826	441	273	100	54	145	135	69	2509 (26%)
renal	22	129	371	466	369	208	163	453	434	271	2886 (30%)
thyroid	4	16	108	143	91	106	57	52	175	138	890 (9.3%)
liver+gb	7	20	81	118	114	68	47	109	62	13	639 (6.7%)
brain	1	37	91	92	79	53	49	62	67	30	561 (6%)
lung	60	197	382	228	218	125	51	22	35	13	1331 (14%)
heart	0	25	140	219	214	104	59	90	50	4	905 (9.3%)
tomo sci	4	16	121	144	168	102	45	11	43	41	695 (7.3%)
other	1	2	31	57	58	92	48	13	7	0	309 (3.2%)
total	163	844	2 151	1 908	1 584	958	573	957	1 008	579	9 525 100 %

Acknowledgement: This work is carried out under the financial support of Ministry and Industry and Trade and Ministry of Health of the Czech Republic

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Mini-Presentation

PAPER TITLE

STRATEGIES DEVELOPED FOR CONTROLING SOURCES IN PERU FROM 1982-1994

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MAJOR SCIENTIFIC TOPIC NUMBER 8.7 (see page 7)

ABSTRACT (See instructions overleaf)

The peruvian regulatory control is based in the system that includes registers, licences, inspection and enforcement. Its programm was designed in two stages, taking into account risk and economic matters.

First stage was addressed to the priority for locating all of the sources - inventory -, and to carry out inspections for knowing the current protection of individuals. This stage, marked by a persuasive than a coercitive control, allowed to meet an important reduction of occupational doses (from 9 mSv/year to an average of 6 mSv/year in only four years) and solving basic aspects in the safety of sources and installations. However, results in the formal aspects were poor because only a few installations and sources were formal licenced.

In this sense, the second stage has started three years ago, with priorities on registration and licencing of sources and installations. The process, that is acomplished by coercitive actions of users, has been succesful and the results show an increase of four times in the registers and licences issued, in just three years. Besides, the inspection activities has been enfaced by trained inspectors, not belonging to authority body, increasing the quantity of sources inspected in almost two times. Results in occupational doses show a reduction to an average of 3 mSv/year. Additionally, the regulatory expenses has been reduced. This results and methodology will allow to enface others aspects as enhacing the patient protection in the short time.

DEPARTURE FROM ICRP-60 IN CANADIAN RADIATION PROTECTION REGULATIONS

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INTRODUCTION

The Atomic Energy Control Board (AECB), Canada's nuclear regulatory authority, is preparing new regulations based on ICRP-60. The proposed regulations have some features which are different from ICRP-60, e.g., the limit for pregnant workers was set at a higher level than that proposed in ICRP-60. The AECB, through its consultative process, received many responses from female staff in nuclear medicine. The concern was that it would be very difficult to prove compliance with such low doses, particularly for internal doses. There was an additional concern that hospital administrators would respond to the reduction of the current pregnant worker limit from 10 mSv to the ICRP-60 recommended value of 2 mSv by discriminating against hiring women. A new proposal for the dose limit for pregnant workers has been made which will limit the dose to the mother by means of a combining formula that will take both external and internal doses into account.

DISCUSSION

The original proposal of the AECB was identical to the ICRP 60 recommendation, i.e., 2 mSv to the surface of the abdomen and an intake of 0.05 ALI. This was criticized in the comments from the public as being unnecessarily low. In addition, it was difficult to demonstrate compliance in view of the difficulty of measurement of doses at such a low value. Therefore, a series of public meetings was held to obtain more direct comments from the female radiation workers themselves. They were asked to advise the AECB on what they thought the limit should be. Considerable effort was made to present the technical information on risk in a form that could be readily understood by the participants. Representatives of professional associations, employers and trade unions were given the opportunity to make presentations.

The response from the public meetings was to set the limit at 4 or 5 mSv for the pregnancy period, i.e., between the current limit of 10 mSv and the ICRP recommendation. The technical presentation at the public meetings had indicated the risk to the foetus from 2 mSv, and compared this to the risks to the foetus from natural causes such as spontaneous abortions, congenital anomalies, etc. The radiation risks were taken from a consultant's report to the AECB(1). This report was prepared by Dr. D. Myers, who also gave the technical presentations at the public meetings.

According to his report, a dose of 2 mSv to the foetus, which would be the approximate dose if a pregnant woman received the ICRP 60 recommended limit, would carry a risk of 4:10,000 total health detriment including fatal and non-fatal cancers. For childhood cancer alone, 2 mSv would carry a risk of 1:10,000. These risks were compared, at the meetings, to

the risks which the foetus faces from natural causes, as mentioned above, which are much higher. The participants at the meetings were of the general opinion that slightly higher risks, i.e., about double that from 2 mSv, would be acceptable when compared to the risks to the foetus from natural causes.

The meeting participants were presented with data from the Canadian National Dose Registry (NDR) for doses received by pregnant workers. Since 1986, 300 to 400 pregnant workers have been monitored for external radiation doses each year. During those years, the maximum dose to any individual, recorded for the duration of the pregnancy, varied from 1.3 to 2 mSv. These doses are in relation to the current limit of 10 mSv during the pregnancy. The NDR data for the category of nuclear medicine technologist (not those that are pregnant) indicates that 70% of these workers receive less than 2 mSv per year, and only 5% receive more than 5 mSv per year, in relation to the current limit of 50 mSv per year. Therefore, the current doses to nuclear medicine technologists would not be a major source of concern, even for those who do not choose to declare pregnancy. It should be pointed out that these doses do not include any contribution from internal contamination. What information which is available at present suggests, is that this is not likely to be a major contribution to total dose and a study on the subject, sponsored by the AECB, is nearing completion, and should give more definite information.

CONCLUSION

The limit of 4 mSv was chosen because:

a) this was the consensus of the meetings, based on acceptable risk considerations;
b) the risks from a dose at this limit are small compared to the risks to the foetus from natural causes;
c) with a current limit of 10 mSv per pregnancy, dose levels for pregnant workers are generally below 2 mSv, which indicates that ALARA is being practiced by most licensees. A limit of 4 mSv would make those that are not, review their operations;
d) we also took into account advice received from the Canadian Human Rights Law Section of the Canadian Department of Justice about the Charter of Rights implications of treating pregnant workers differently from other workers.

REFERENCE

1. "Comments on ICRP 60 Rationale for Dose Limits for the Pregnant Worker" (INFO-0421). Atomic Energy Control Board, Ottawa, Canada, 1992.

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1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

02 AUG 1995

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Abstract No. 96050
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PAPER TITLE IMPLEMENTATION OF DOSE CONSTRAINTS : ANALYSIS OF THE REGULATORY ENVIRONMENT

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MAJOR SCIENTIFIC TOPIC NUMBER 8.3 (see page 7)

ABSTRACT (See instructions overleaf)

The concept of dose constraint was put forward by the International Commission on Radiological Protection (ICRP) 5 years ago. Consensus has been reached on most of its characteristics, but it is not yet applied in the European Union regulations. Under the auspices of DG XI, a study has been carried out in order to look at the practical difficulties that may be encountered in doing so. It aimed at identifying the practices which involve "restrictions" on doses, below the regulatory limits.

Both in the field of occupational and public exposures it has been found that many restrictions apply in the national regulations, or in the less formal exchange between the authorities and the industry.

Most of those restrictions have many common points with actual constraints but few of them possess all the characteristics of a dose constraint as defined by ICRP. For example, limits for gaseous or liquid discharges do aim at doses below the limit for the public and they do take into account other sources, but they have the statute of a regulatory limit. Some authorities add "objectives", below the authorized levels, which are almost real constraints, but the practice of optimisation does follow very seldom. As regards occupational exposures, the analysis of the safety cases showed that the management often proposes dose objectives to the authorities.

The paper will present the results of this investigation and propose solutions for an introduction of the dose contraint concept that would make profit of existing practices.

A POSSIBLE APPROACH TO ASSESS COMPLIANCE WITH A LIMIT BY MEASUREMENTS*)

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Abstract

A proposal is made to how to prove or not compliance with a limit by radioactivity measurements, taking into account the statistical nature of radioactivy. The proposal has also some influence to the requirements for the quality of the measurement.

1. PROBLEM

Usually figures are set in standards /IAEA 1994/ or by authorities with the objective to control doses. These figures are established for different exposure conditions as emergency or chronic exposure. Such figures might be called guidelines, reference levels, action levels, limits etc., and have to be conceptually in terms of dose. For practical reasons and therefore more frequently, however, derived quantities are used as activity concentration for a given radionuclide i in a given material, e.g. in foodstuffs or in air. In the following, the activity concentration $A_{c,i}$ is denoted as A.
In any case, an apparently very precise figure is established, and the compliance with this number is achieved by checking whether the assessed activity concentration is below or above this figure. In the most frequent case, the assessment of the activity concentration by measurement, a statistical quantity results. In this case, two different kind of figures have to be compared: a deterministic and a statistical quantity.

The figures are:

a) a preset limit, indicated by a singular figure, usually expressed with one or two significant digits, e.g. 0,2 kBq/kg
b) a statistical quantity, to be expressed by a mean and a variance.

When samples are subject of measurements, a question as
" can you assure (or even guarantee) that the sample complies with..... "
is raised by authorities, by media and by the public. From a scientific point of view, striktly spoken, a positive answer can not be given at all. This is because radioactive decay does not behave to fulfil the properties of a deterministic quantity. However, the question is still asked

frequently, although being an unscientific one by definition. Although a foolish question usually deserves a foolish answer, a realistic approach which might satisfy nonscientific formalists and being acceptable from scientists is presented below.

2. BASIC CONSIDERATIONS

The fulfilment of the condition, say

$$A < L \qquad (1)$$

where: A... activity, L.. limit

can be stated only with an additional information, namely the confidence interval of the assessment. Therefore, the correct statement is

$$A < L \text{ , with a given probability} \qquad (2)$$

linking three values:

a) mean A (estimated from the measurements)
b) uncertainty σ (estimated from the measurements)
c) limit L (constant)

Taking into account the well known properties of statistical distributions, a statement:

$$A < L \text{ , with 100 \% probability}$$

can not be achieved by definition, although being required by some groups who think that the setting of limits solves all problems.

3. PROPOSAL

In the following, a possible solution of the question raised above is presented. In order to obtain a resonable answer, the question on the purpose of the assay of radioactivity has to be formulated in a more precise form.

3.1 Formulation of the questions

There are three possible questions, which have to be clearly expressed in advance:
a) what is the activity concentration?
b) is the activity concentration of the sample lower than a given value and hence in compliance with an upper limit (e.g. a possibly contaminated food sample)?
c) is the activity concentration of the sample higher than a given value and hence in compliance with a lower limit? (not considered in detail here)

3.2 Answers

ad a) The first question is answered by measurements, the answer being an activity (concentration) A and a standard deviation σ expressing the statistical uncertainty:

$$A \pm \sigma$$

This formulation of the question is not associated with compliance with limits, but rather with the assessment of the activity as such.

ad b) This is the most frequent case, namely to check whether an activity in a sample is less than or equal to a preselected figure.

It is proposed to consider a sample in compliance with the limit when the following condition is fulfilled:

$$A + n.\sigma \leq L \qquad (3)$$

The choice of the number of standard deviations n is subject of discussion and by definition somewhat arbitrary. When n is chosen too small, the measurement is uncertain. When n is large, a lowering of the limit is associated with. To obtain a compromise, it seems therefore reasonable to adopt n = 1.

4. DISCUSSION

This concept presented above has the following properties:

i) the activity of the sample is by definition less than the limit. Hence, a possible blame of a hidden increase of the limit is unfounded.

ii) The definition has an inherent property which might be even more important than the condition i) above: The acceptable uncertainty of the measurement in a given case depends on the activity level and the quality of the measurement, i.e. on the variance and the detection limit. This implies that an inexpensive measurement with a large variance is sufficient to prove compliance with the limit for samples well below the limit. On the other hand, when the activity approaches the limit, an accurate measurement permits the confirmation of the compliance of samples with the limits with almost arbitrary accuracy. Details can be found in /ÖN 94, Ts 95/

References:

/ÖN 94/ ÖNORM S 5250-1 : Zählstatistische Aspekte bei Radioaktivitätsmessungen
/IAEA 94/ Safety Series 115-I (1994) International Basic Safety Standards
/Ts 95/ M.Tschurlovits et al: Proc. IRPA regional meeting Portoroz Sept.1995, in press

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PAPER TITLE Limitation of Radioactive Discharges from NPP Based on Radionuclide Specific Monitoring

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MAJOR SCIENTIFIC TOPIC NUMBER 8.3 (see page 7) or 8.7

ABSTRACT (See instructions overleaf)

Monitoring of nuclear power plants (NPP) discharges into atmosphere and hydrosphere based on particulate-iodine-gase (PIG) measurements and gross beta or gamma plus tritium measurements respectively is being improved by nuclide specific measurements employing semiconductor gamma and alfa spectrometry and radiochemical methods. In connection with this progress new concept of authorized effluent limits and the related regulations implemented in Czech Republic is presented. Activity of all principally contributing to the effective dose radionuclides discharged during a year multiplied by a Sv/Bq conversion coefficient based on a standard model are summed up and the effective dose is compared with the new limit. These limits should be for discharges into atmosphere authorized as per caput collective committed effective dose in the NPP surrounding and into the hydrosphere as committed effective dose to the critical group member.

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Vienna, Austria

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PAPER TITLE A case for applying temporary dose limits

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The Basic Standards for Radiological Safety enforced in Argentina in 1994, include the ICRP Publication 60 recommendations and are consistent with the International Basic Safety Standards of the IAEA. Studies performed show that no relevant difficulties should appear in accomplishing with the new occupational dose limits in the nuclear and radioactive facilities in the country, except for the Atucha I Nuclear Power Plant. It was said in previous studies that in this plant it would be necessary a significative effort for the optimization of radiological safety procedures in order to diminish the occupational doses, and to change the fuel channels by new ones free of cobalt. At the moment it have been changed 59 channels, almost 25% of all.

The Argentine Basic Standards consider special circumstances as those where radiation protection in a practice has been optimized and occupational exposures still remain above the new dose limits, and it can be predicted that reasonable efforts can, in due course, bring the occupational exposures under the limits. In this case the Regulatory Authority may exceptionally aprove a temporary change in a dose limitation requirement of the Standards -only if formally requested by the registrant or licensee-, with the constraint of 50 mSv in a year and for no more than 5 years.

It have been identified individual and collective doses due to the change of fuel channels, and the workers for whom a temporary change in the dose limits could be asked for.

RATIONAL SYSTEM OF RADIATION DOSIMETRY

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INTRODUCTION

Radiation doses are the most important subject to the sciences relating to the effects of ionizing radiation on matter. Since any science at all must stand on the quantitative description of causality, uses of physical quantities as the measures are indispensable.

The current system of radiation dosimetry is built on the fundamental dose of the absorbed dose, which is defined as the "energy imparted" density, and for practical convenience, various weighted absorbed doses are introduced as subsidiary. However, it has been pointed out that these quantities lack adequacy inherently as a measure of causes, in addition to the insufficiency of the specification of the concept(1-3).

Firstly, separation of the quantities of radiation field and of the dose is not possible, since both quantities are deeply related to the same microscopic constituents, i.e., electrons(4,5).

Secondly, the value of the absorbed dose cannot be fixed at the moment of irradiation. Since the absorbed dose is a quantity of interaction product of radiation and matter, the values of the dose cannot be fixed instantaneously.

Thirdly, it is not easy to envision the physical entity of the quantity from the definition. The form of existence of the "imparted" energy is not clear in the current definition (1).

These defects can not be removed as long as an interaction product is adopted as the dose quantity. In this paper, the authors present a prescription to solve these problems.

PROPOSAL OF A RATIONAL DOSE CONCEPT

Being the system of radiation dosimetry rational, the quantity adopted as the dose should meet the following requirements.

First of all, the quantity of measure (radiation dose) is to be chosen as point specific not only in space (coordinates) but also in time (coordinate). Secondly, the value of dose should be fixed at the moment of incidence. Thirdly, the dose is to be of local field quantity.

In order to guarantee the measurability, one must define the quantity point specific in space-time coordinates. Though the problem with the space has been discussed (4), the temporal structure of energy transfer contributing to the dose has not been mentioned so far. The value of the dose defined with interaction products is not definitely settled for a given moment, since the events of energy transfer do not in general occur simultaneously.

Since the dose is to be used for the purpose of predicting future phenomena of radiation effects, its value should be clearly determined at the instance of irradiation. As far as an interaction product is used as a dose, its value for the future can only be estimated as a mathematical expectation, though its value for the past is definitely established. Thus, the dose should not vary temporally thereafter.

With the current system of dosimetry, values of the dose as well as the field are influenced simultaneously by the change in configuration of materials surrounding the point of interest. The difficulty arises in separating the field and the dose of radiation (4).

In order to meet above stated requirements, we believe that THE SOLUTION is to define the dose with a radiometric quantity, the particle fluence, not with a dosimetric quantity. Because radiometric quantities are point specific both spatially and temporally, their values are defenitely settled at the moment of incidence of radiation, and their physical entities are clearer in general. However, there is room for discussion whether particle fluence is superior to energy fluence or not.

RELATION BETWEEN THE CURRENT AND PROPOSED SYSTEMS AND PROBLEMS

Usually, very drastic changes to a prime system are not favorable. Because its influence to the society must be tremendous. However, even today, practical dosimetry is not always made in a way faithfully which follows the definition of the current system of dosimetry, e.g., neutron dosimetry, high energy particle dosimetry. In these cases, doses are evaluated through the measurement of the particle fluence using corresponding conversion coefficients. So, one has only to use these conversion coefficients as the weighting factors to the particle fluence when one wants to use the current doses subsequently in the new system.

Moreover, any dosimetric quantity, in principle, can be expressed as weighted particle fluence. No undesirable consequences are found for the moment.

For an illustration of the situation, relations of the effective doses of photons and of neutrons, an important dose in the field of radiation protection, and their particle fluences are shown in the Figs. 1 and 2 . As seen in the figures, the impact of the particle is an increasing function of its kinetic energy, which seems desirable at least from a pedagogical view point.

Dosemeters for practical uses can be so designated with the intention of adjusting their responses to these curves. In case of photon dosimetry, such a procedure can provide a more direct method than the traditional one via a third quantity such as exposure, kerma or absorbed dose.

The unit of the new dose is one of particle fluence, $[m^{-2}]$, in the sense of metrology, but the familiar names of gray or sievert may be succeedingly used as SI special units, if so desired.

CONCLUSION

It has been shown the current system of dosimetry has inevitable defects, as far as the physical quantity of the dose is not changed. Our proposal of adopting particle fluence instead might be rather drastic but it is shown that the currently used practices can be transferred to the new system without any serious difficulty.

To use particle fluence as the basic dose, the stability of the quantity is guaranteed.

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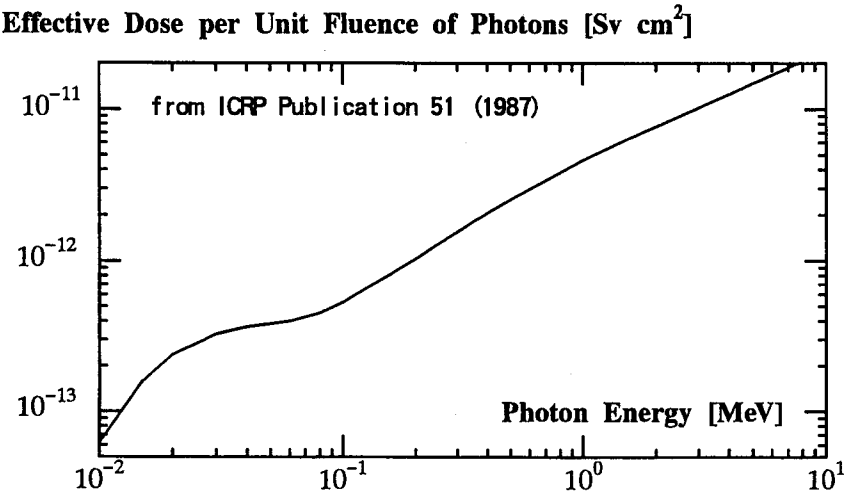

Fig.1 Relation between Effective Dose of Photons and Particle Fluence

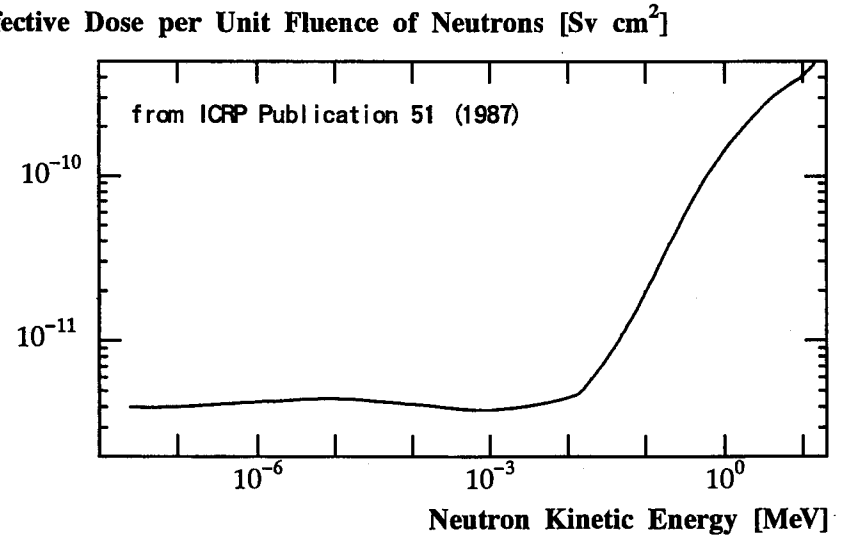

Fig.2 Relation between Effective Dose of Neutrons and Particle Fluence

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PAPER TITLE Hazard Assessment of Long-Lived Radionuclides

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MAJOR SCIENTIFIC TOPIC NUMBER 8.10 (see page 7) 2nd choice (8.6)

ABSTRA(The presence of long-lived radionuclides, such as Pu-239, in radioactive wastes has been perceived to pose a hazard of unprecedented severity and duration. This perception has caused serious difficulties in developing acceptable waste management programs. However, technical assessment indicates that concerns related to long-lived radionuclides are largely unwarranted.

Since specific activity is inversely proportional to half-life, it is clear that the longer the half-life of a radionuclide, the less radioactive it is. Fears directed toward the hazard of long-lived radionuclides might more appropriately be directed toward the stable toxic elements, such as Pb and Hg, which persist forever. Recent evidence on proton decay indicates that the distinction between stable and radioactive could be somewhat vague. It is suggested that certain misconceptions might be avoided if nuclides were classified as either radioactive ($T_{1/2}$=<10^6 yr), radiopassive ($T_{1/2}$= >10^6<10^{12} yr), or radioquiescent ($T_{1/2}$ = >10^{12}yr).

Special problems related to dealing with the long-lived radionuclides are discussed and evaluated. These include: (1) specification of ALI's for certain radionuclides, which considering their low specific activity, would be impossible to attain, (2) concerns directed toward certain nuclides, such as I-129, which could cause no significant doses, and (3) implementation of controls which would likely result in greater risks than those due to the long-lived radionuclides being controlled.

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PAPER TITLE NEW RELATIONSHIPS IN BASIC DOSIMETRY

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MAJOR SCIENTIFIC TOPIC NUMBER 3.2. (see page 7)

ABSTRACT (See instructions overleaf)

NEW RELATIONSHIPS IN BASIC DOSIMETRY

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In this paper four new relations are derived, for the first time, based on the definitions of the fundamental dosimetric quantities given in the last ICRU Reports. These relations have the general form $A.\dot{B} = \dot{A}.B$, where A and $\dot{A}$, B and $\dot{B}$ are the dosimetric quantities and their rates, respectively.

Thus, the author found $\Psi.\dot{\Phi} = \varphi.\dot{\Psi}$, for a monoenergetic beam of photons, in any point of this beam and at any time. Here Ψ is the energy fluence rate, φ particle fluence rate, Ψ energy fluence, and Φ the particle fluence. The second relation is $K.\dot{\Psi} = \dot{K}.\Psi$, where K and $\dot{K}$ are kerma and kerma rate. The third relation, for absorbed dose D, and its rate $\dot{D}$, is $D.\dot{\Psi} = \dot{D}.\Psi$. Finally, for electron fluence Φ_e, and electron fluence rate φ_e, the corresponding relation is $D.\dot{\varphi}_e = \dot{D}.\Phi_e$.

At the end of paper the author states interesting remarks regarding an exhaustive description of an ionizing radiation beam or radiation field closely related to their interaction with matter.

HOT-CELL OPTIMIZATION USED IN PRODUCTION AND DISTRIBUTION OF ^{153}Sm

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ABSTRACT

The purpose of this paper is to optimize the shielding lead thickness necessary to the hot-cell used in the production and distribution used in the ^{153}Sm - label compound, ethilenediaminetetramethilene phosphoric acid (EDTMP) that will be constructed in radioisotope production plant. of Instituto de Pesquisas Energéticas e Nucleares", of "Comissão Nacional de Energia Nuclear" (IPEN-CNEN/SP). For this study, we considered the following attributes: number of people exposed, the magnitude of individual doses due to normal exposures, collective dose and activity handled. For the optimization we will utilize the analytical quantitative decision-aiding techniques such as cost-benefit analysis and differential cost-benefit analysis, for taking account of all protection and safety attributes in a systematic way.

INTRODUCTION

The ^{153}Sm-EDTMP is a label compound that is absorbed in the bone and has been applied to patients as a relief for pain. The IPEN-CNEN/SP is assembling a hot cell for production and distribution of this label compound to support weekly quantities of 1.85 GBq (5Ci).

We have considered as basic case for the hot cell shielding thickness, the option of protection one (01), that attends the dose limit established by CNEN[1] , and the actions taken led to the introduction of shielding in the hot cell frame, in a way to keep the annual dose lower than 40 mSv. This value was calculated considering a 25% error in the measurements.

Starting with the basic case, we have identified other options, trying to reduce the dose to satisfy the 20 mSv annual medium for five (05) years established by ICRP[2], the 15 mSv per year to cease the individual monitoring, the 4 mSv to cease the routine area monitoring and 1 mSv to not exceed the dose limits for public exposure.

The major radioprotection problems in optimization studies are related to the main attributes: the observed dose and the costs associated to reduce this doses. The adoption of other option different from that elected as optimum case, should consider the availability of financial resources.

RESULTS

The optimization study in ^{153}Sm-EDTMP production hot cell is shown in Table I where has been built with the aid of cost-benefit analysis decision-aiding techniques.

From the Table I results, we can see:

1) If we take the international standard alpha value, adopted in Brazil, of US$10000/person-Sv, the optimum option is the number eigth (08).We can also observe in column six (06), that the individual dose for this option is 0.55 mSv per year, which is about half of annual dose allowed for the general public.

2) Considering that IPEN-CNEN/SP has a hot cell with a shielding thickness considered in the option thirteen(13), we could verify the activity capacity of this hot cell for the ^{153}Sm, in a way to give the same dose observed in the optimized cell, option eight (8), column six (6). In option thirteen (13), column four (4), Table I, we can verify that the hot cell capacity is 18.5 TBq (500 Ci), what means 100 times bigger than the today's prevision.

We find interesting to reproduce the calculations that generated the Table II, using the aiding technique for decision making named cost-benefit differential analysis, because their results practically show as the study sensibility of alpha value.

From the Table II, we can see:

3) The optimum option is still number 8, as it should be, because the results are independent of the technique used.

4) If we had used the real alpha value for our country, which is US$3000.00/person-Sv, the optimum option the number 05 (five) with an annual dose of 1.9 mSv, bigger than the annual limit dose for the public but lower than the value necessary for the workplace monitoring with routine function and the saving in the radiological protection expenses should be US$36.20 per year.

5) If we use the european alpha value, US$20000.00/person-Sv, the option 11 (eleven) should be the optimum and the annual dose would be 28.8 μSv. This value is very close to 10 μSv/year, the exemption value, but the annual cost , X, should be US$16.1 per year, greater than that of the option 08 (eight).

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TABLE I - Cost-Benefit Analysis

Option	Shielding Material	Thickness Shielding (cm)	Activity (TBq)	H_E/t (μSv/h)	H_E (mSv)	$H_E.N$ (person.mSv)	α.S = Y (US$)	X (US$. person Sv)	X+Y (US$)
01	Pb+Lucite	0.1 + 1.0	0.185	194.60	38.92	194.60	1946.00	128.74	2074.74
02	Pb+Lucite	0.2 + 2.0	0.185	90.00	18.00	90.00	900.00	247.77	1147.77
03	Pb+Lucite	0.5 + 2.0	0.185	76.68	14.74	73.70	737.00	262.17	999.17
04	Pb+Lucite	1.7 +2.0	0.185	15.59	3.12	15.60	156.00	326.17	482.89
05	Pb+Lucite	2.0 + 2.0	0.185	9.50	1.90	9.50	95.00	340.89	435.89
06	Pb+Lucite	2.5 + 2.0	0.185	3.93	0.79	3.93	39.30	366.79	406.09
07	Pb+Lucite	2.6 + 2.0	0.185	3.28	0.65	3.28	32.76	371.94	404.70
08	Pb+Lucite	2.7 + 2.0	0.185	2.72	0.55	2.72	27.24	377.09	404.33
09	Pb+Lucite	2.8 + 2.0	0.185	2.26	0.45	2.26	22.62	382.24	404.86
10	Pb+Lucite	2.9 + 2.0	0.185	1.88	0.37	1.88	18.75	387.39	406.14
11	Pb+Lucite	3.0 + 2.0	0.185	1.55	0.31	1.55	15.50	393.19	408.69
12	Pb+Lucite	5.0 + 2.0	0.185	0.03	0.01	0.03	515.39	515.39	515.68
13	Pb+Lucite	5.0 + 2.0	18.5	2.88	0.58	2.88	0.29	515.39	544.19

TABLE II - Differential Cost-Benefit Analysis

Option	X (US$)	ΔX (US$)	S (person.mSv)	ΔS (person.mSv)	ΔX/ΔS (US$/person.Sv)
01	128.74		194.60		
02	241.77	119.03	90.00	104.60	1137.95
03	262.17	14.40	73.70	16.30	883.43
04	326.17	64.00	15.60	58.10	1101.55
05	340.89	14.72	9.50	6.10	2413.11
06	366.79	25.90	3.93	5.57	4649.91
07	371.94	5.15	3.27	0.65	7874.61
08	377.09	5.15	2.72	0.55	9329.71
09	382.24	5.15	2.26	0.46	11147.17
10	387.39	5.15	1.87	0.39	13307.49
11	393.19	5.80	1.55	0.32	17846.15
12	515.39	122.20	0.01	1.54	80331.32

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PAPER TITLE A MODEL FOR THE DETERMINATION OF MONETARY VALUES OF THE MAN-SIEVERT

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MAJOR SCIENTIFIC TOPIC NUMBER 8.4... (see page 7)

ABSTRACT (See instructions overleaf)

In order to implement protective actions to reduce radiological exposure "as low as reasonably achievable", the use of cost-benefit analysis has been progressively promoted among the practitioners, especially for the management of occupational exposures in nuclear power plants. Applying this procedure to the selection of radiological protection options, a monetary valuation of the avoided exposure has to be adopted (the so-called monetary value of the man-sievert), allowing for the expression of the benefit of protection in the same unit as the protection costs. The value of the man-sievert has to reflect the detriment associated with radiological exposures. As stressed by the International Commission on Radiological Protection (ICRP) in its Publication 37, it is necessary to evaluate not only the "objective" aspect of health detriment, but also the "subjective" one like the perception of risk by individuals. Moreover, in its Publication 60, the ICRP has recommended to take into consideration the dispersion of individual exposures. The integration of the equity aspects in the value of the man-sievert means that one accepts to pay more in order to avoid a unit of exposure when the individual level of exposure increases, and, moreover, that this increment of the monetary value of the man-sievert is more and more important.

This paper aims at presenting a model for the determination of the monetary values of the man-sievert dealing with the three following objectives: reduction of the collective exposure, reduction of the dispersion of individual exposures and reduction of dispersion in priority in the highest individual levels of exposure.

THE STANDARD FOR JUSTIFICATION OF RADIATION PRACTICE

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INTRODUCTION

Justification is the first principle for the system of radiological protection proposed by ICRP. Other principles, the optimization of protection and individual dose limits, have been well studied quantitatively. The justification of a practice, however, has been scarcely discussed. The reason is obviously that the justification is related to the evaluation on both the risk and the benefit of a practice. There are social scientific problems in the discussion on the benefit and especially on the comparison between the benefit and the risk. Without the basis of the justification, the purposes of the optimization of protection and individual dose limits become ambiguous.

JUSTIFICATION OF ACTUAL RADIATION PRACTICES

Any existing practices must be justified consciously or unconsciously by some ways. The actual radiation practices, the mass screening tests for tuberculosis and stomach cancer, the use of a smoke detector with ^{241}Am, the use of an artificial tooth containing uranium and a lens of eyeglass containing thorium, and occupational radiation work are observed how they have been justified, mainly in Japan.

MASS SCREENING FOR TUBERCULOSIS: Tuberculosis had been a national disease in Japan more than 40 years ago. Many young people were infected and died, and the mass screening was wanted nationally. The buses for the X-ray inspection went around offices, schools and communities. Gradually the infection rate has fallen and tuberculosis became a geriatric disease. Then the justification of the mass screening has been controverted. In 1975, WHO recommended the stop of the X-ray mass screening for tuberculosis. The benefit has been decreasing with years, and that for the people younger than 20 years old becomes below the risk(1). Even for adult people the cost for one life saving becomes very high. The X-ray mass screening for pupils and students has been decreased in stages.

MASS SCREENING FOR STOMACH CANCER: The mass screening for stomach cancer systematically began at 1960 in Japan. The benefit of the mass screening has been compared with the risk of the X-ray exposure, and further compared with the endoscopic mass screening and with no mass screening(2). Methods for medical decision making have been applied(3), and the economical balance of the mass screening has been discussed in the scales of the mutual society of a company or a local government. The benefit of the mass screening on health and economics increases with the incidence rate which increases with ages. Furthermore in order to neglect practically the genetic effect of the exposure, the X-ray mass screening for stomach cancer is recommended for people above 40 years old.

SMOKE DETECTOR WITH ^{241}Am: The benefit of a smoke detector is the saving of life, house, and others from a fire. The exposure from the ^{241}Am source and the pubic exposure for its waste has estimated and justified from the comparison of the dead from a fire(4). The developments of electron techniques made an electron smoke detectors which is comparable in the efficiencies on fire prevention for that with ^{241}Am. The ^{241}Am smoke detectors in use have drastically decreased.

AN ARTIFICIAL TOOTH CONTAINING URANIUM AND A LENS OF EYEGLASS CONTAINING THORIUM: The justification of these goods is very difficult, because their

benefit is adornment and cannot be compared directly with the risk of the exposure. In practice, the uranium or thorium content is restricted by the government or the guild of manufacturing companies and their goods are on the market. A leaflet from UNEP(5) wrote that the use of these goods was not justified because the benefit is only adornment which is obviously worse than the health risk from the exposure. In such cases that the benefit of a practice is not concerned with health, the comparison of the benefit with the risk is difficult and needs for the social scientific considerations.

OCCUPATIONAL RADIATION WORK: Many people work under radiation environments and are exposed some amounts of dose. The benefit of the practice is the wages of working, the pay of the practice, the joy of working, and/or others. The occupational exposure is accepted naturally under the individual dose limits without the discussion on justification. The benefit changes depending on each individual, but the limit of the risk is same for all workers. If the individual dose limits were the maximum permissible doses, the occupational radiation work would be justified in respect of radiological protection.

THE DEFINITION OF THE JUSTIFICATION

The justification of a radiation practice is not only the comparison of the benefit with the risk, but also the comparison of the justification with the justification of other practices having the same purpose, as shown in above discussion. The benefit and the risk are concerned with health, economics, and others. In some cases, health and economics are trade-off. On the basis of the above observation, the justification of a radiation practice is defined as " The practice is the best in the options which are designed for the same purpose (including the option which does not execute the practice)".The capability approach developed by Amartya Sen(6) is adopted as the standard for what is the best . The capability approach can be discussed on many categories relating to a practice as the capabilities of a person to function.

THE CAPABILITY APPROACH

Sen proposed that the capabilities of functionings can be used for the standard of living(7), instead of opulence or utility. The relationship among capability, function, opulence and utility are schematically shown in Fig. 1. Opulence(goods) has been used for the standard for the evaluation of benefit and risk. Obtaining the goods gives the person command over the corresponding characteristics of the goods and the monetary cost for risk decreases the ability to buy the goods. Opulence, however, is not always proposed to the well-being of a person, because opulence can be used depending on physical and social environment. Utility is widely used in social sciences, especially in economics, and utilitialism is the most available ethical principle. The utility for a person, however, is affected for the person's subjective point of view and surrounding environment.

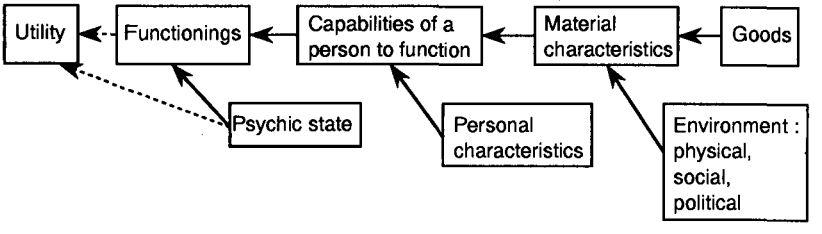

Fig.1. Utility, functionings, capabilities and other sources(7)

FUNCTIONING

The functionings of a person are included in many categories, for example, health, safety, community life etc. on the quality of life. In order to arrange many functionings, "the social indicator" or "the list of social concerns" can be used. For example the social indicators edited by Committee of Quality of Life, Council of Living Life in Japan are divided into ten main areas, 1.health, 2.education, study, culture, 3.quality of employment and working life, 4.leisure, 5.income, consumption, 6.physical environment, 7.crime and execution of the law, 8.family, 9.quality of community life, 10.social strata and movement. Under the main areas, they are further divided into 27, 79, and 160 concerns. With the reference of these indicators, the functionings relating to a radiation practice can be listed.

RANKING OF OPTIONS

If the capabilities of functionings concerned with the options can be ranked, the best option can be evaluated. But overall ranking is obviously impossible, because the comparison between functionings in different areas is very difficult. In practice, the partial ranking can contribute the evaluation of the options.

For the justification of X-ray mass screening for stomach cancer, three options, 1.no mass screening, 2. X-ray mass screening, 3. endoscopic mass screening are ranked in reference to the possibilities on mortality, economical effects, facilities of the screenings and the request from inspected people. The ranking on health has been mainly discussed and the second option is justified for the people above 40 years old.

For the justification of an artificial tooth containing uranium and a lens of eyeglass containing thorium, two options, 1. use of the goods containing uranium or thorium, 2.use of the goods not containing uranium and thorium, are ranked. The related functionings are personal adornment and health risk with radioactive materials(the public exposure with the waste is possible to be neglected). The ranking is not obvious. In this case, the author thinks that the use of an artificial goods containing uranium or thorium is justified, because low risk practices of the user only must be acceptable in society in considering with the importance of passive freedom.

From the viewpoint of the justification, the limits of occupational exposure dose should be changed with a type of occupation and should be made an agreement of the limits with the employee and the employer. The risk(an increase in the possibility of carcinogenesis) itself must be indemnified and furthermore the suffering of a cancer must be also indemnified. A new system for occupational radiation work is necessary, although difficult problems on the evaluation of the risk of low level exposures and the natural occurrence of cancer are existed.

CONCLUSION

The justification of a radiation practice are normatively defined as "The practice is the best in the options which are designed for the same purpose (including the option which does not execute the practice)". The options are ranked by the capabilities of a person to function.

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PAPER TITLE HSE's Central Index of Dose Information

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MAJOR SCIENTIFIC TOPIC NUMBER 8 (8.4?) (see page 7)

ABSTRACT (See instructions overleaf)

One of the key requirements in UK radiation protection legislation is that exposures should be kept as low as reasonably practicable. New, across the board, regulations came into force in 1986. From that date, the UK's Central Index of Dose Information (CIDI), run by the National Radiological Protection Board under contract to the Health and Safety Executive (HSE), has received from Approved Dosimetry Services annual summaries of the doses reported for 'classified workers' (category A workers in the Euratom basic safety standards Directive). HSE set up CIDI especially to handle the data, particularly with a view to receiving information on trends in worker exposure as an aid to developing policy and targeting inspection and enforcement priorities.

CIDI can analyse the data by dose and age range, sex, occupational grouping and, to some extent, by type of radiation; it also, of course, produces collective and mean dose. Dose data is published annually and the first 'trends' report, analysing occupational exposure 1986-1991, was published in 1993. It clearly demonstrated that whole body doses to workers, particularly in the nuclear industry, had fallen dramatically since 1986; by the end of the period they were in most cases well below 20 mSv yr^{-1}, the average annual dose limit recommended in ICRP 60. However, it also showed that there were two categories of worker, industrial radiographers and non-coal miners, whose doses had not followed the downward trend and remained unreasonably high.

The trends data, updated to 1994, and HSE's actions in response to the information, will be described.

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PAPER TITLE THE USE OF THE WILLINGNESS TO PAY APPROACH FOR THE DETERMINATION OF MONETARY VALUES OF THE MAN-SIEVERT

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MAJOR SCIENTIFIC TOPIC NUMBER 8.4... (see page 7)

ABSTRACT (See instructions overleaf)

The monetary valuation of the man-sievert used for the optimisation of radiological protection is based on the loss of life expectancy associated with one unit of collective exposure and is estimated using the ICRP risk coefficients of radiation-induced cancer and genetic effects, and thus refers to the monetary value of statistical life. The review of the literature on the value of life derived from the willingness to pay approach clearly demonstrates a large range of values according to the various risk situations that can be considered in the society.

In order to apply the willingness to pay approach for the valuation of the man-sievert, a specific questionnaire has been established. Preliminary results have been obtained concerning occupational exposures in nuclear power plants. According to the framework developed for the determination of the monetary values of the man-sievert, including risk aversion considerations, a baseline value for the risk of cancer can be considered and compared to the human capital approach as well as the risk aversion coefficient driving the increasing function of the value of the man-sievert associated with the individual level of exposure.

IS THIS REMEDIATION NECESSARY? COMPARING COST VERSUS RISK ABATEMENT WITH OTHER RISK REDUCTION MEASURES

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INTRODUCTION

Radiologically-contaminated sites abound and include a wide variety of sites and sources. The responsible party is not always known . No estimate is available for potential remediation cost or volume. Area residents are concerned about the potential risk posed by these sites. This concern is transferred to their governments via their state and federal elected representatives, and to regulatory bodies. A large fraction of sites require site remediation, either at private or public expense. An assessment of the risk reduction to be accomplished is often lacking. Risks posed by site remediation may exceed those of the "no-action option."

CASE STUDIES

I assumed that all exposure takes place over 50 years, during which the workers spent 40% of their time on-site. Residents were assumed to live in the area for fifty years, spending 100% of their time in their residences. I assumed that the average person lost 35 years of life if they contracted a fatal cancer as a result of their radiation exposure. The risk associated with chronic radiation exposure is assumed to be 2.6×10^{-2} per person-Sv (1).

Hypothetical Facility (Facility X)

Facility X processed ^{238}Pu ($t_{½}$=87.7 a) for radioisotope thermal generators. As a result of normal work practices, some of the grounds have become contaminated with ^{238}Pu. The owners of Facility X now wish to convert their facility to the production of three-hole punches. The facility is an industrial area in a medium-sized city, several miles from the shores of a large lake. They are attempting to develop a decommissioning plan which will provide the maximum safety to their workers and the public while permitting them to successfully convert their facility. If they cannot do so, they may be forced to close the facility.

The plutonium contamination is greatest directly beneath the facility's main building. At this point it is 37 Bq/gm in a hemispherical volume that is 1 meter in radius. Concentration drops off linearly with distance. Options currently under consideration are remediation to a level of 0.2 Bq/gm and to 20 Bq/gm with clean backfill. If the company cannot reach an agreement with the regulatory bodies then it may file for bankruptcy.

Southerly Waste Water Treatment Plant (SWWTP)

The SWWTP was identified as being contaminated with ^{60}Co ($t_{½}$=5.27 a) in early 1991. The cobalt was apparently disposed of into the sanitary sewer system by a licensee of the Nuclear Regulatory Commission (NRC) during the mid- to late-1970s. It is not known if this material was disposed of illegally or as a part of licensed activities. The radioactive material is relatively immobile and is heterogeneously deposited around the site at the SWWTP.

Site characterization showed the highest levels of contamination to be on the order of tens of Becquerels per gram of soil (4). The estimated maximum possible exposure to a worker on-site was determined to be 0.14 Sv with a maximum credible worker dose of less than 0.01 Sv (5). The calculated dose to the public was less than 0.01 mSv over 50 years. The risk associated with the maximum credible exposure is 2.6×10^{-4}. The risk to the public is approximately 2.6×10^{-7} per person.

Chemetron Bert Avenue Dump Site

The Chemetron Corporation manufactured chemicals utilizing a depleted uranium catalyst during the 1960s, contaminating a portion of the facility. Much of this material was disposed of into an industrial dump owned by Chemetron. Approximately 1.5 million cubic feet of contaminated soil is contaminated in excess of the NRC-allowed limit of 1.3 Bq/gm. Contamination levels varied from background to over 330 Bq/gm of uranium, in addition to metals and chemicals (7).

The owner-preferred remediation option is to construct an on-site clay closure cell which would be covered with 5 meters of clean fill. The state and citizens prefer that all soil which is contaminated in excess of 31.3 Bq/gm be removed.

LEVEL OF REMEDIATION, RISK, AND COST

Facility X

Scenarios investigated for this facility included no action, excavation to a level of 20 Bq/gm with one meter of clean fill, excavation to a level of 0.2 Bq/gm with eight meters of clean fill, and placing 10 cm of asphalt cover over the contaminated areas. An "all pathways" analysis was performed for each of these options, assuming industrial use. In all cases except for the

"no action" scenario, the primary exposure source was that of direct gamma radiation exposure. This is chiefly due to plutonium's geochemical immobility. Under the "no action" scenario, the major source of exposure was dust inhalation. The "no action" option resulted in an exposure of approximately 0.13 Sv over 50 years to the individual for a risk of $3.3x10^{-3}$. Assuming a worker population of 1000, this will give a total of 3.3 lives and 114 years of life lost over 50 years.

The most cost-effective method of site remediation is to cover the site with 10 cm of asphalt. This would reduce dose to a maximum of 0.70 mSv/yr, giving a dose of 0.03 Sv over a fifty-year time span for a risk of $7.8x10^{-4}$. A worker population of 1000 will have an overall risk of approximately 0.78. The cost of this option would be approximately $1000 for materials and one afternoon of laborer time, giving a total cost of $303 per life saved, or $9 per year of life saved.

Remediation to levels of 20 Bq/gm and to 0.02 Bq/gm, result in doses of $1.5x10^{-5}$ mSv/yr and $1.6x10^{75}$ mSv/yr respectively. These maximum doses occur at a time of approximately 1000 years after deposition. If the cover layer is maintained so that erosion does not occur, then the doses are zero.

The cost associated with these options is difficult to estimate due to the difficulty of remediating a plutonium-contaminated site. The volume of material to be removed to remediate to 20 Bq/gm is that of a hemisphere one meter in radius, or 2.1 cubic meters (74 cu. ft.). Assuming a cost of approximately $1500/cu. ft. for remediation and disposal yields a total cost of $110,000 for this remediation and $960 per year of life saved. A remedial level of 0.02 Bq/gm requires excavating 1072 cubic meters (38,000 cu. ft.) at a cost of $57 million or $500,000 per year of life saved.

Southerly WWTP

Under the "no-action' remediation option the risk presented to the workers is approximately $2.6x10^{-4}$ per person. The projected dose to the public from this contamination is negligible due to the relative immobility of cobalt in the environment. The risk to workers at the SWWTP is approximately 0.26 excess deaths, assuming a population of 1,000 workers at this site.

The likely remediation option at the SWWTP will be excavation and disposal of the most highly-contaminated material. Due to the short half-life of ^{60}Co, nearly 90% of the original radioactivity has decayed away since the deposition of the material sixteen years (three half-lives) ago. Remediation at this time will reduce overall radiation exposure and its associated risk by only 10%. At the estimated remediation cost of $40 million (6), the cost per year of life saved is approximately of $44 million.

Chemetron

Modelling (for this paper) indicates that, under the "no action" remediation option, the lifetime dose to an occupant of this site would be 4.0 REM with a risk of $1.5x10^{-2}$ at zero cost. Burial in an on-site closure cell would give a dose of less than 0.01 mSv with a risk of $<2.6x10^{-7}$ per person. The projected cost of this option was $7 million (9). Remediation and off-site disposal of all material with a contamination level of greater than 1.3 Bq/gm of uranium also gives a dose of less than 0.01 mSv. The projected cost of this option was in excess of $20 million (9). The dose resulting from the removal activity may exceed that of on-site disposal due to construction activities contaminated soil, dust generation, and transportation accidents.

The cost per year of life saved at this site would be $19 million for the construction of an on-site closure cell and $55 million if the material is removed if the site is not used for residential or industrial/business purposes after remediation.

Facility and Remediation Option		Cost	Years of Life Saved	Cost per Year of Life Saved	Years of Life Saved for Same Investment (10)		
					Medical (Ave = $5000)	Fatal Injury Reduction (Ave = $48,000)	Toxin Control (Ave = $2,782,000)
Facility X	asphalt cover	$1000	113	$8.70	0.2	0.021	$3.6x10^{-4}$
	20 Bq/gm	$11,000	114	$96	2.2	0.23	$4.0x10^{-3}$
	0.02 Bq/gm	$5.7 million	114	$50,000	1140	119	2
SWWTP	current plans	$40 million	0.35	$4.4 million	8000	833	14
Chemetron	closure cell	$7 million	0.35	$20 million	1400	146	2.5
	material removal	$20 million	0.35	$57 million	4000	417	7.2

Table 1: Summary of Costs and Years of Life Saved for Remedial Options

DISCUSSION

If the level of contamination decreases linearly with distance from the source, reducing the level of contamination by a factor of two increases the volume of soil to be removed increases by a factor of eight (2^3). If risk varies directly with contamination level, in order to reduce the risk by a factor of two it is necessary to increase the volume of soil to be removed and disposed of by a factor of eight. *Any agency or organization deciding to spend money to reduce a very low risk has also decided to remove that money from other, possibly more effective risk reduction measures. If the risk being mitigated is very low in comparison to those risks to which we are exposed on a routine basis, then this money may have been wasted.*

Table 1 compares the cost of the remediation options mentioned above and general intervention categories mentioned by Tengs, et al (10). Tables 2 and 3 show the median cost per year of life saved for several US regulatory agencies and interventions.

CONCLUSIONS

Remediation of contaminated sites is expensive. The level of risk reduction achieved is not always commensurate with the investment of time, money, and materials that must be made.

In addition to fiscal considerations, there is risk inherent in any remedial activity. While a remediation will result in a lower level of risk to plant workers or area residents, all of the activities necessary for this remedial action involve some degree of risk. *If the unavoidable risks inherent in a remediation effort equal or exceed those that the site poses "as is" then this remedial action is not reducing risk, it is exporting risk to another group of people or to another locality.*

Risk is not the only consideration in an environmental restoration project. Risk reduction per dollar spent should not be used to justify either pollution or abandoning a heavily contaminated site. Rather, using risk reduction as the sole argument for site remediation is fallacious. We must consider not only risk reduction contaminated site, but the reduction of risk to the population as a whole. We cannot indiscriminately pollute the environment, but neither can we indiscriminately remediate our past mistakes. We require a balanced approach whereby we invest our money wisely by remediating those sites which pose a clear threat to either human safety, the environment, or both while devoting more attention (and money) to those activities which are more cost-effective but less glamorous, such as vaccination programs, home smoke detectors, traffic safety, and so forth.

Regulatory Agency	Number of Examples	Median Cost per Year of Life Saved
FAA	4	$23,000
Consumer Product Safety Commission	11	$68,000
National Highway Traffic Safety Administration	31	$78,000
OSHA	16	$88,000
EPA	89	$7,629,000

Table 2: Median Cost per Year of Life Saved for Federal Regulatory Agencies' Interventions

Intervention	Median Cost per Year of Life Saved	Intervention	Median Cost per Year of Life Saved
Smoke detectors	$5000	Airplane Safety	$23,000
Emerg. Vehicles	$2000	Speed Limit	$45,000
Smoking Cessation Advice	$6000	Flammability Standards	$68,000
Health Services	$14,000	Asbestos Control	$1,865,000
Traffic Safety Edu.	$20,000	Radiation Control	$27,386,000

Table 3: Median Cost per Year of Life Saved for Several Interventions

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THE RECORDING LEVEL IN THE 1990s

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ABSTRACT

This paper addresses the subject of what is the lowest occupational dose one should try to measure in personal dosimetry for external radiation. This question is raised because modern personal dosemeters have improved measurement capabilities, allowing very small doses to be detected and measured. These are similar in size to natural background doses. It is not intended to provide definitive advice here, but to show what the considerations might be; the discussion is limited to measurements of photon and beta doses, and does not deal with specific systematic sources of uncertainty, such as energy dependence of response.

THE RECORDING LEVEL

The recording level is defined by ICRP (1,2) as "a formally defined value for dose equivalent or intake above which a result from a monitoring program is of sufficient interest to be worth keeping". The Commission recommends (1) that the recording level be based on one-tenth of the dose limit, pro rata for the monitoring period concerned; for personal dosemeters, this effectively sets a minimum standard. However, the Commission also recognises that lower recording levels may be justified for purposes such as analysis of collective dose. In revising the earlier recommendations of Publication 35 (1), ICRP may consider changing the fraction of the dose limit upon which the recording level is based; however, for the present purposes, the value of one-tenth will be assumed. It is to be noted that a "result from a monitoring programme" will necessarily refer to a single component, for example the beta/photon component, of effective dose or effective dose equivalent.

During the last decade, personal monitoring for photon and beta doses was achieved almost entirely by the use of film badges and thermoluminescent dosemeters (TLDs), which, in the main, could detect doses as low as 50-100 μSv. The annual effective dose equivalent limit was 50 mSv, which implied a maximum recording level of about 400 μSv for a typical four-week or one-month monitoring period. For analysis of collective dose, there was a desire to measure doses rather lower than the dosemeters could allow, with a result that the "decision limit" (see below) for the personal dosemeter was usually taken as a recording level.

However, as the 1990 recommendations of ICRP are adopted, there is naturally a pressure to reduce recording levels. Taking an average annual dose limit of 20 mSv (2), the maximum recording level for a four-week period will be reduced to about 150 μSv. This would be achievable by most older systems for photon and beta doses; however recent technological developments now allow still lower doses to be measured comfortably, as discussed below. For demonstrating that average doses are normally well below the maximum recording level, even the improved performance of modern dosemeters might be inadequate.

FIGURES OF MERIT

In describing the limiting behaviour of dosimetric methods at low doses, clear terms must be used. Here, the terminology of Christensen and Griffith (3) is used; their choice essentially followed the work of Currie (4) and of others (5,6). There are three figures of merit for any system, which are as follows. The basic figure, and the one most often quoted, is the *decision limit.* This is also known as the "critical level" or the "detection threshold", and is defined as the level of signal, expressed in terms of dose, at which there is a given confidence (normally 97.5%) that the signal is not due to variations in intrinsic background. Results below this level are not normally reported, or included in dose summations. However, this practice of artificial truncation can be criticised on the grounds that it gives a negative bias to cumulative dose assessments.

The second figure is the level of dose which will, with a certain level of confidence (often 95%), give rise to the correct detection of a dose. This is known as the *detection limit*, and its definition entails not only confidence that the "true" dose has a value at the detection limit, but also that the signal is not due to intrinsic background. See reference (6) for a full discussion. The third figure of merit is the level of dose at which the precision of measurement reaches a certain value, and is known as the *determination limit*. Its use requires the input of the required precision: Christensen and Griffith (3) suggest the use of a relative standard deviation of 45%, derived from consideration of the recommended recording level. In published performance data on dosimetry systems, however, the latter two figures of merit are not often quoted, although it is really one of these which should be compared with the recording level. Which of the detection and determination limits should be used for this comparison is a question which needs resolving.

NEW TECHNOLOGY

Recent developments in personal dosimetry have seen both an improvement in decision limits for passive dosemeters, and the advent of electronic dosemeters. Typical standard TLD systems now have decision limits of less than 25 μSv (see for example references 7 and 8), whilst better performance is claimed to be possible when laser heating methods are used (9). The availability of systems based on LiF: Mg,Cu,P (10) is expected to reinforce this capability, giving routine decision limits of 10 μSv and lower. Most recently, a system has been described which uses a special plastic matrix in a technique known as cooled optical luminescence dosimetry (11), and which may have a decision limit much lower than 1 μSv.

Active personal dosemeters have been widely used for some years, but recent advances have allowed doses to be measured over a wide range of photon and beta energies, to the extent that electronic dosemeters can now form the basis of "legal" dosimetry services. NRPB has been running a legal dosimetry service based on the Siemens/NRPB Electronic Personal Dosemeter (EPD) (12) for over a year; the dosemeter uses PIN diodes and possesses a decision limit of 1 μSv. In a separate development, the Direct Ion Storage (DIS) dosemeter, using a modified EEPROM chip, has been shown to have a similar capability (13). One of the ways in which the mode of use of active dosemeters differs from that of passive dosemeters is in the monitoring period. Users of electronic dosemeters are likely to be nuclear site operators, where dose information may be stored on a daily basis. In this case the monitoring period can be said to be one day, and the recording level may need to be viewed accordingly.

Using the ICRP recommended average annual effective dose limit of 20 mSv, the recording level for a four week or one month monitoring period can be derived as about 150 μSv. For one day, the figure will be about 5 μSv. As suggested above, this figure ought to be compared with either the detection or the determination limit for the dosemeter concerned; however the foregoing paragraphs have quoted values of the decision limit. The relationship between the decision limit and the higher figures of merit will vary according to the specific dosemeter type, and it is therefore impossible to infer the values for the latter in a general way. For the present purposes, however, it may be appropriate to compare the recording level with a value equal to twice the decision limit. Using this simplistic approach, it can be seen that most passive systems will comfortably meet the requirement to measure doses as low as the recording level for four weekly or monthly wearing; whilst active devices can similarly meet the recording level for daily monitoring. However, for analysis of collective dose, the limitations of the dosimetry systems still need to be borne in mind.

NATURAL BACKGROUND

There remains one further factor to be considered when dealing with recording levels: the magnitude and variability of natural radiation background dose. The components of natural background which are detected by beta/photon dosemeters are solely those of cosmic rays and terrestrial gamma radiations. In the UK, for example, these amount to an annual average effective dose of 600 μSv, with a range of about 300 - 1000 μSv (14) determined by geological and geographical factors.

Whilst mean occupational doses have been higher in the past, these are now being reduced to levels where natural background doses are significant by comparison. The operators of single sites can take reasonable account of local natural background by measurement, provided that the distinction between natural and occupational dose is clear. Here the uncertainty in natural background will depend chiefly upon the measurement method. By contrast, commercial personal monitoring services who supply many sites may adopt one of two approaches: a global average figure can be used, or extra dosemeters can be supplied, as controls. Both approaches have drawbacks. In the first case, local variations in background are not taken into account, whilst in the second, extra cost is involved and several dosemeters must be used to obtain a useful value.

CONCLUSION

At very low dose levels, such as are measurable by modern dosemeters, it may be the case that natural background represents a large fraction of the total dose received by an individual. It may at least be true that it is no longer the technical capability of the dosemeter which limits the smallest dose to be recorded (the local recording level), but rather the complexity and cost of eliminating natural background doses from measurements of personal dose. At present it appears that passive dosemeters with decision limits of 10 - 20 μSv, and electronic dosemeters with decision limits of 1 μSv, are sufficient to meet all low dose measurement requirements. It is probably not worth investing resources in improving decision limits further, when dosimetric performance is no longer a limiting factor in determining practical recording levels.

The views expressed herein are those of the authors and not necessarily those of NRPB.

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THE MANAGEMENT OF CHANGE IN BNFL'S UK GROUP WITH PARTICULAR REFERENCE TO THE RESTRUCTURING OF THE HEALTH AND SAFETY FUNCTIONS

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INTRODUCTION

BNFL's UK Group business involves reprocessing of uranium metal (Magnox) fuel, storage and dismantling of Advanced Gas Reactor (Ceramic UO_2) fuel, waste treatment and storage, decommissioning and electricity generation. These activities are carried out at three sites in the North West corner of England and Southern Scotland:-

Sellafield	-	all activities
Chapelcross	-	electricity generation
Drigg	-	waste storage

THE NEED FOR CHANGE

The privatisation of the UK Electricity Supply Market in the early 1990's created a very competitive marketplace. The nuclear component of the supply market was specifically excluded from the privatisation due to economic factors. By 1993 the threat to the UK Nuclear Industry was severe. Some of the key issues were:-

- Over-capacity in the electricity market of 25%.
- Planned gas stations would increase over-capacity to about 45% by the mid 1990's
- A review of the Nuclear Industry was pending which would critically review the economics of the industry.
- Nuclear generation was not competitive with gas generation.
- Nuclear Electric and Scottish Nuclear, the companies owning the UK Nuclear Power Stations, had made significant advances in reducing their costs and were looking to BNFL, a major supply, to reduce its fuel cycle charges which accounted for two thirds of their generating costs.
- The Magnox stations were nearing the end of their original design lifetimes.
- Three Magnox stations had closed on economic grounds.
- BNFL had evolved, over the years, from a public sector environment bringing with it the attributes of such an organisation eg, overmanning, bureaucracy, hierarchical management structures and a need to improve employees business awareness.

Against this background the UK Group Board concluded that fundamental changes were necessary.

PROCESS OF CHANGE

The UK Group Board, in discussions with its main customers, concluded that savings of the order of £100M in the operating cost base had to be achieved within five years if Nuclear Generation was to stand any chance of being competitive in the electricity supply market. It was estimated that the £100M saving would come from a manpower saving of 2,100 posts (from a starting point of 7,000 posts) and £40M from other variable costs. To achieve this saving the Board identified two main areas for change:-

- The attitudes, behaviours and beliefs of its employees (culture).
- The alignment of the business activities with the key business drivers to clearly identify the groups core activities and to sharpen business awareness and customer focus.

An absolutely critical success factor during the change process was to at least maintain, if not improve, the safety performance of the Group.

In addition these changes had to be conducted under the critical eye of the UK's regulator, the Nuclear Installations Inspectorate.

The starting point of the change process was for the Board to achieve a clear focus and ownership of the changes required. This involved the Board working on and agreeing its role, ensuring the right individuals were on the Board, forming itself into an effective team and, most importantly, agreeing a set of "values" which would underpin the whole approach of the group.

These values included such key behaviours and beliefs as (i) leading by example, (ii) safety will not be compromised, (iii) we will trust people, (iv) we will allow individuals to work to their full potential, (v) customers are important.

In line with these underpinning values the UK Group Directors, together with a team of facilitators, reviewed the "Divisional" structure of the group and re-organised it around the "key drivers" of the business into "business" units. The old structure of six divisions transformed into the new structure of seven Business Units.

Key points of the restructuring were:-

- Each 'operational' business unit had its own Head of Safety and associated staff reporting directly to the Director.
- Management layers were removed.
- Management 'spans of control' were increased.
- Each business unit had a clear purpose linked to a key business driver.

Under the old divisional structure many functions and services were centralised, including Health and Safety. The new structure devolved those services essential to the hour to hour operation of the plant to the business units whilst centralising the other services into a Services Business Unit. Hence Health and Safety advice and monitoring rest in each operating area whilst services such as dosimetry and occupational health rest in the Services Business Unit.

The need to provide focus and co-ordination across all the Business Units on such items as policy and strategy was identified and a separate Business Unit was set up for this purpose. In this unit is a small team of Health and Safety specialists lead by the Head of Safety for the Group who reports directly to the Group Director. In addition to the co-ordination role this team has a responsibility to advise the Group Director on the group performance and to provide a monitoring role.

Such a structure can only work if every individual believes in and works to the 'values'. To this end a great deal of team building, at all levels, has been undertaken, both vertically and horizontally through the organisation. In addition individual managers were selected for their management style ie, those who sought consensus, empowered individuals, were people orientated and were good communicators.

At the time of writing UK Group is 20 months into its five year programme. The next stages of the change programme are (i) to develop its business processes to world-class standard via business process re-engineering and/or continuous improvement methods, as appropriate, (ii) to continue to develop the management style and achieve the reality of the 'values'.

RESULTS TO DATE

- 7000 employees down to 6,200.
- £30M saved on operating costs per annum.
- All production targets met
- Senior Manager posts reduced by 25%.
- No deterioration in safety performance.

To demonstrate the results achieved in a typical H&S Department a comparison of the Magnox Reprocessing H&S Dept is given: 7 management layers down to 4, spans of management control of up to 12, 138 employees down to 115, 10% saving on operating costs.

LESSONS LEARNT

The key lessons are listed below:-

- You must have a vision of the future and keep working towards it.
- Leadership and commitment from the top are essential. Senior Managers must be seen to lead the process.
- You are dealing with people and they will need support (eg stress counseling, redeployment support).
- Identify all the stakeholders and involve them (INCLUDING THE REGULATOR).
- Effective communication is all important.
- Keep your plans live and react to feedback.
- Use your best people in the change process.
- Keep checks and balances in place during the change process to ensure the business does not 'fall over'.
- Involve those people affected by the change in the process.
- Be radical, no turning back.
- Reassure and look after the people you need to keep in the organisation.
- Identify the 'barriers to change' and plan to overcome them.
- Keep your nerve.

SUMMARY

Achieving fundamental change in a well established nuclear business which is highly regulated is possible and can bring large benefits.

The keys to success are leadership and commitment from all Senior Managers, involvement of all stakeholders and communication. It is essential to involve the regulator as a stakeholder.

ENVIRONMENTAL RADIOACTIVE ANALYSIS INVOLVING PUBLIC IN THE NEIGHBORHOOD OF A ^{131} I PRODUCTION FACILITY

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ABSTRACT

The IPEN has a production plant,where elemental tellurium is processed to obtain iodine.The gaseous effluents produced in the three hot cells are filtered in a batch system before being discharged to the environment through a stack.This paper is mainly concerned with the definition of the critical group,the evaluation of the annual individual and collective effective dose due to the iodine release to the environment.This result obtained are discussed taking into account new concepts (Safety Series Standards 979 : "International Basic Safety Standard for Protection Ionization Radiation and Safety of Radiation Source ", 1995 such as exemption levels and the levels where is necessary to apply the quantitative decision aiding techniques to optimize the environmental delivery or the introduction of a new filter batch system.

INTRODUCTION

The IPEN has a swimming pool research reactor operating at a nominal power of 2 MW where elemental tellurium is irradiated. Radioactive tellurium and iodine radioisotopes are produced and ^{131} I is chemically separeted in a production plant. This chemical processing involves three hot cells and gaseous effluents are produced and filtered in a batch system before being discharged to the environment through a 25 m stack.

This paper is mainly concerned with the definition of the critical group,the evaluation of the annual individual and collective effective dose due to the iodine release to the environment in the last year (1994).

METHODOLOGY

The gaseous effluents are continuously monitored by using a air sampler coupled to a flux meter. This system is conected off line in the stack. The effluents are collected through a filter system arranged after the pump. The filters are routinely measured by using a hyperpure germanium detector with 15% efficiency coupled to a 4096 multi-channel analyser. The results are corrected for the air flux in the stack. The spectrum obtained is analised using the Omnigam software version 3.4 (EG&E ORTEC) .In order to evaluate the impact due to this effluents release the average meteorological data collected in a meteorological tower far 200m from the radioisotope processing plant have been used as input to one generic models for assessing the environmental transfer of radionuclides [1,2] .These data were collected during ten years. These were the more advantageous metereological data we have until the present moment to perform our calculation. Although we have installed since August 1995 some metereological instruments, we have not enough data. The average wind speed and frequency distribution were determined sharing the compass card in 8 sectors. The most time the wind blows in the southeast direction having an average frequency of 39 % and an average speed of 3.6 m/s.The major exposure pathway considered in the external radiation exposure were:beta and gamma submersion dose from the plume,gamma dose over contamined ground .In the internal radiation exposure we considered the intake by inhalation.There are not farming activities around 60 km of the IPEN site so doses due to the food ingestion pathway are assumed negligible.the annual effective dose have been carried out by using the AIEA transfer model [2] and by applying the proper dosimetric factors [3] . The critical group was obtained and is formed by the people living 3000m away from the discharged point in the southeast diffusion section and was obtained from the "Instituto Brasileiro de Geografia e Estatística" data and was determined the fraction of the number of persons living in the area delimited by a circular section with one kilometer of radius and the 8 compass card sectors. This Institute is encharged by the Federal Brazilian Government of the National Census. The data is a population estimation for 1991 from 1980 because we have not calculated the population distribution around IPEN from 1991 census. The main exposure pathway is the gamma external irradiation due to the deposition of radionuclides on the ground

INDIVIDUAL AND COLLECTIVE DOSE CALCULATION

The individual dose was calculated using the methodology above described. As was pointed, the critical group was formed by the people living 3000 meters away from the discharge point in the southeast diffusion section and received an effective dose of 1.9×10^{-6} Sv per year and so is below the exemption level to be considered radioactive[4] and the collective dose calculation is not necessary. In this case we can increase the plant production five times, maintaining the present environmental discharge system and we continue in the exemption level. During the last four years [5] no considerable change was observed in the operation condition of the IPEN discharge. Nevertheless the IEA-R1 reactor is being upgraded to operate at higher power (5MW) on a full-time (24 h per day) in order to attend the demand for radioisotopes in Brazil and other countries of South America. In this case, even the production increase until fifty times the present production is not necessary to record the source term because the individual dose in the critical group remains below 1/10 of the pertinent annual limit (100 μSv/year). If we consider the estimation for the future demand approximately year 2000, we can suppose an increasing of production of 150 times the present production and we need record the source term but we do not need to accomplish the environment monitoring around the IPEN facilities. Above 50 times the present production becomes necessary to evaluate the collective dose because if the value exceeds 1 person-Sv than we need to apply the ALARA principle. The collective efffective dose was calculated multiplying the population in each area by the dose calculated in the middle point of the same area. We obtained a value of 61.1 person-Sv. This value suppose that one increase of the production above the exemption level probably we need to introduce the ALARA principle and determine the different radiation protection options to reduce the collective dose. This result reflect the great number of person that live near IPEN facilities.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Comments on Interpretation of Intervention Principles and Levels for Protection of the Public in the Event of a Nuclear Accident

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MAJOR SCIENTIFIC TOPIC NUMBER 8.5 ..... (see page 7)

ABSTRACT (See instructions overleaf)

Whilst the basic principles for protection of the public in a radiological emergency are well discussed and widely accepted, application of these principles in real intervention practice meets serious problems. Radiological information is the most complicated one to understand and interpret for decision makers. The paper deals with some of those points in establishing intervention levels, which need careful attention to avoid misunderstanding in decision making process: formulation and wording of criteria for implementation of radiation protection and social protection measures, interrelation between primary (in terms of dose) and derived (operational) intervention levels, and balancing realistic and conservative approaches for estimating projected, averted and residual doses. The points listed are considered on the basis of analysis of international recommendations (ICRP, IAEA, OECD/NEA et al) and experience gained in post-Chernobyl intervention policy in Russia. The findings of this consideration may be useful in updating the interpretation of general intervention principles for both pre-planning of emergency response and justification of intervention criteria at various stages after an accident.

OPTIMIZATION GUIDANCE TO POST-ACCIDENT INTERVENTION: A SPECIFIC CASE.

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INTRODUCTION

ICRP recommends the application of the system of protection to intervention situations [1], i.e. those in which exposure pathways are already present; e.g. the public exposure following an accident. This implies that intervention must be justified and optimized, being the optimization the process of deciding the nature of protective action in order to obtain the maximum net benefit. This paper provides an example of one optimization model to guide a decision making process in a specific case of post-accident intervention. The involved scenario postulates the contamination of big quantities of reinforcing steel bars used in construction industry, and a lot of them present in the structure of several dwellings.

Inhabitants of these dwellings must be protected and the proposed action is to demolish those homes exceeding some intervention criterion. The objective of this study is to reach such intervention level trough an optimization process economically focused.

THE OPTIMIZATION MODEL

It is assumed that a gaussian distribution takes place in the radioactivity present among the contaminated reinforcing bars. The intervention level must be given in terms of some measurable quantity. In this study, we use the equivalent-dose rate, measured at the moment of the decision making, in some representative place inside the dwelling (i.e. relative to their inhabitants exposure).

Also in this paper, we use the cost-benefit analysis technique described elsewhere [2,3] to reach the optimization. The normalized cost (i.e. per contaminated home) of the protection is given by:

$$X = V \cdot F_d \tag{1}$$

Where:

- V is the cost of protection, which includes demolition, re-building and disposal of contaminated material per dwelling
- F_d is the fraction of demolished dwellings

On the other hand, the corresponding radiological detriment cost is given by:

$$Y = \frac{1}{\lambda} \cdot (1 - e^{-\lambda \cdot t}) \cdot \dot{H}_0 \cdot F_s \cdot T \cdot N \cdot \alpha \tag{2}$$

Where:

- λ is the involved isotope decay constant
- t is the considered period of time
- $\dot{H}_0$ is the sought intervention level
- F_s is the fraction of dwellings remaining stand after intervention
- N is the average number of people per home
- T is the average ocupancy factor per person
- α is the collective dose cost

The objective function U (to be minimized) is given by the sum of (1) and (2):

$$U = V \cdot F_d + \frac{1}{\lambda} \cdot (1 - e^{-\lambda \cdot t}) \cdot \dot{H}_0 \cdot F_s \cdot T \cdot N \cdot \alpha \tag{3}$$

$\dot{H}_0$ will be proportional to radioactivity present at the reinforcing bars, hence it will follow a gaussian distribution too, as depicted in fig. 1:

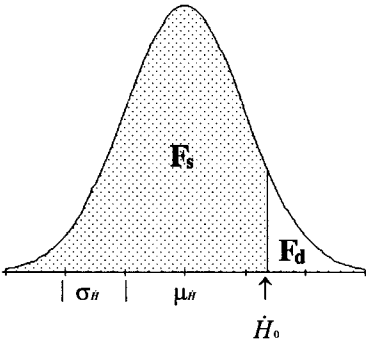

Figure 1. $\dot{H}$ distribution, and (F_d , F_s)-meaning.

In Fig. 1 $\mu_{\dot{H}}$ is the observed equivalent-dose rate averaged over the contaminated homes, and $\sigma_{\dot{H}}$ is the corresponding standard deviation. Fig. 1 illustrates the correlation existing between the chosen value of $\dot{H}_0$ and the fraction F_s (or F_d) of remaining (or demolished) dwellings through the selected distribution function. The strict shape would be the corresponding cumulative function. Table 1 shows values taken from real case ocurred in México's 1984 accident [4,5].

Table 1.- Values of parameters used as data of calculations

λ	1.50×10^{-5} h^{-1} (Co-60)
V	50,000 USD
$\mu_{\dot{H}}$	5×10^{-7} Sv h^{-1}
$\sigma_{\dot{H}}$	1×10^{-7} Sv h^{-1}
t	4.38×10^{5}
T	0.5
N	5 people per dwelling
α	100,000 USD

In Table 1, the value of V corresponds to the moment of the accident. The values of $\mu_{\dot{H}}$ and $\sigma_{\dot{H}}$ are estimated from the data of the same accident, however in a hypothetical case, they could be determined from a rapid and enough sampling. The value of t (50 years) is chosen consistently with a long term situation, whenever T, N and α values are country-specific values, being α the proper value for intervention situations.

Substituting table 1 parameter values, and giving values in ec. (3) to $\dot{H}_0$ (finding the corresponding values to F_s), we obtain the correlation between function U and $\dot{H}_0$, as depicted in Fig. 2.

Table 2, gives the results for the around-minimum values, showing that the sought value is about 7.1×10^{-7} Sv h^{-1} which would implicate that only 1.8 % of the contaminated dwellings would have to be demolished.

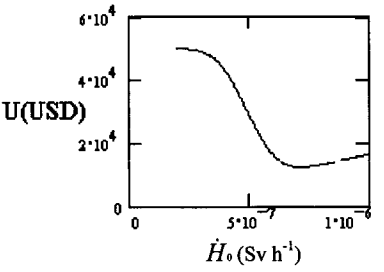

Figure 2. U function vs. $\dot{H}_0$, obtained through ecuation (3) and values of table 1.

Table 2. Some calculated values through ecuation (3) and values of table 1.

$\dot{H}_0$ (Sv h^{-1})	F_s	U (USD)
6 x 10^{-7}	0.841	16,326
6.5 x 10^{-7}	0.933	13,426
7 x 10^{-7}	0.977	12,511
7.1 x 10^{-7}	0.982	12,487
7.2 x 10^{-7}	0.986	12,500
8 x 10^{-7}	0.999	13,351

CONCLUSIONS

The assumption of considering a gaussian distribution of the radioactivity among contaminated reinforcing bars seems to be qualitatively and quantitatively right for a scenario like the contemplated in this study, and it permits that an optimization model could be applied to get a proper intervention level in order to guide a decision making process. In addition, average parameters, such as T and N in ec. (3) were introduced consistently with the fact that the overall vision must be statistically approached. Although the data from the Mexican accident were used only as an example, a retrospective view shows that, refering to adopted intervention levels, decisions took at that time can be considered right even with the fact that they were taken without an optimization analysis.

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Poster No: P85-3

THE METHODS, MODELS AND INFORMATION SUPPORT ON ADMINISTRATIVE DECISIONS ON MEDICOSANITARY MAINTENANACE OF POPULATION OF UKRAINE IN THE AFTER CHERNOBYL ACCIDENT PERIOD

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The liquidation of serious consequences of Chernobyl accident, taking into consideration their character and scale, requires the huge material and human resources. The complexity and plural of factors, affecting on planning and management of measures on medical-sanitarian maintenance of damaged population, requires the processing during acceptance of decisions, large files of informations, containing parameters of healths condition of damaged population, information about the level of complex medical-sanitarian maintenance, dynamics of investments and compensation and etc.

The instability of economic situation in Ukraine, connected with transition to new forms of managing and formation of political system, does not permit to increase the resources on decisions the problems of liquidation of Chernobyl accident, requires the increasing the effectiveness, efficiency and qualities of accepted decisions on control and management of forming situation.

The operative and qualitative decision of all problems of medicalsanitarian maintenance appears possible in conditions of introduction of new methods of tax, processing and analysis of information on the basis of modern information technologies, methods and algorithms of mathematical theory of acceptance of decisions.

With this purpose are developed the principles and structures of information-analytical system of profile purpose, intended for programme realization of following technologies of organizational management are developed: 1. technology of automated planning,

including tasks:

- planning and distribution of medical equipment, medicines, reagents, instruments and other medical, sanitary- epidemiological and other health-improvement establishments;
- planning and distribution of pases on making healthier to persons, victim of a Chernobyl failure;

2. technology of automated account, analysis and acceptance of administrative decisions, including tasks:
   - account and analysis of condition the complex medical - sanitarian maintenance;
   - account and analysis of condition of health-improvement system of population;
   - account and analysis of system of preventive maintenance of mass disseases of victims;
   - account and analysis the system of maintenance of population by medical-preventive products;
   - account and analysis the parameters of condition of health of victims;
   - account and analysis of condition (including the personnel and technical maintenance) of medical establishments;
   - account and analysis given about preventive maintenance and treatment thyroid gland lossed ones;
   - formation, management and analysis the register of persons, lossed owing to Chernobyl failure.
3. technology of automating formation of expert-analytical valuations, including decisions of tasks:
   - formation of expert-analytical valuations of condition of health damaged population and sanitary-epidemiological environment of his residing,
   - formation of integrated valuations and radiation situations in administrative territories and forecast its development;
   - formation of expert-statistical valuations for definitions of priorities in investment policy on liquidation of failure and protection of population.

The decision of listed tasks is based and on the traditional methods of theory of acceptance of decision, on using of effective expert systems, on developed methods of mathematical simulation and

forecasting of development of formed situation with using the methods of mathematical (linear, dynamic) programming, theory of distributions of resources and theory of games.

The optimization of created data bases, perfection of algorithms of acceptance of decisions permits not only operatively to execute the investments medical-sanitarian maintenance of regions, but also to execute the necessary check by their effective employment.

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PAPER TITLE ON THE PROBLEM OF EFFECTIVENESS ESTIMATION FOR COUNTERMEASURES, INTERVENTION LEVELS AND DECISION MAKING

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In estimation of the effectiveness of countermeasures (CMs) and choices of justified intervention levels use is made of the cost-effectiveness analysis (CEA), cost-benefit analysis (CBA), as well as corresponding economical, radiological parameters and different variants of the multi-attribute utility analysis. The authors works give the evidence that CM implementation under conditions of non-uniform contamination of lands and distinguishing of agricultural production obtained as "pure" (with contamination levels below the DILs) and "dirty" (accordingly, those above the DILs) can result in increase of (internal) collective doses for the population. The CEA is not applicable under these conditions, and standard algorithms of CBA require a generalization for the correct use in investigation of the scenarios stated. The CBA generalizations obtained give the possibility to estimate effectiveness of CMs (including the cases with consideration of various uses of "pure" and "dirty" production), and to define "optimum" DILs for various strategies of using the production. Estimations of countermeasures are considered with concrete examples based on a system of averted collective dose, costs of CM implementation, costs of aversion of unit collective doses (CEA), total detriment and complete CBA. Different criteria are shown to render preference to different CMs. In this connection, when it is decided to initiate the most efficient ("optimum") CMs,- along with consideration of the problem of decision making with uncertainty, also problems of multiple criteria (comparison/preference of different criteria) should be taken into account.

THE PROTECTION FROM POTENTIAL EXPOSURE IN THE BRAZILIAN LEGISLATION

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ABSTRACT

The Brazilian Legislation contemplates the work in potential danger places for practices involving originally inflammable materials, including explosive materials, and finally electricity. After the Goiânia accident, ionizing radiation was also included in the Brazilian Legislation, except for federal civil servants that were excluded because they work under special legislation. Immediately after the ICRP Publication-64 : "Protection from Potential Exposure: A Conceptual Framework" had become available , the potential exposure danger for federal civil servants was also acknowledged, and the federal government issued a new legislation based on three levels of danger, that favored them. Thus, we could include in the new legislation the main concepts of the framework suggested by the ICRP publication-64. This paper intends to discuss the manner in which the ICRP publication-64 was introduced in the Brazilian Legislation.

I INTRODUCTION

The Brazilian legislation contemplates those who work in potentially dangerous places, in practices involving inflammable material (Law nº 2573 of August 15, 1955) explosive material (Law nº 5880 of May 26, 1973) and electricity (Law nº 7369 of September, 1985) because of the possibility of electrocution.

This legislation is based on an old fashioned policy, according to which the employer must pay an extra amount in the salary if the practice does not present the same normal safety conditions.

After the Goiânia accident, the ionizing radiation was included in the Brazilian Legislation through Decree nº 97458 of January 01, 1989, and was included in the regulation norm nº 16 of the Ministry of Labor, except for the federal civil servants, who were excluded because they have on special legislation.

In December 17, 1991, Law nº 8270, concerning federal civil servants salary, as well as it's readjustments and restructuring and other provisions , came into effect, and article nº 12 states:.

Article 12 - Federal civil servants will receive additional salary concerning insalubrity and potential risk, in terms of the legal norms and regulations pertaining to workers in general and based on the following porcenteges:
I - five, ten, and twenty percent, concerning insalubrity in the minimum, medium and maximum degree respectively.
II - ten percent, in the case of potential risk,
§1 The additional amount for ionizing radiation will be received in five, ten and twenty percent, according to what is determined by the regulations.
§2 The gratuity for work involving X-rays or radioactive substances will be based on ten percent.

The author of this paper was invited by the Brazilian Nuclear Energy Commission to present scientific bases to support the federal decree mentioned in §1 above.

Next, the author was invited to participate in the study group responsible for the drafting of the decree that was submitted to the workers' associations, which eventually resulted in federal decree nº 877 of July 20, 1983. The decree contemplated the scientific bases established in the draft, which are the main objective of this paper.

II SCIENTIFIC BASES USED IN THE DECREE Nº 877 OF JULY 20, 1993

First we must mention that any individual has the right to receive the additional amount if he is working in a place where there is the possibility to receive the expected radiation doses that exceed the annual dose limits for workers, in case the potential risks become real.

The ICRP publicationS (1) were used to set the scientific bases. In these publications (1) we referred mainly to the following items:

(29) At levels of effective dose below about 0.1 Sv, only stochastic effects are expected to occur and the probability of their recurrence is assumed to be directly proportional to the effective dose. The relationship of probability of harm to dose is therefore linear without threshold in this range. A nominal proportionality coefficient of 5×10^{-2} Sv^{-1} for the probability of fatal cancer in the general population, given a dose and dose rate effectiveness factor (DDREF) of two for low dose or dose rate, is used by the Commission in ICRP Publication 60.

(30) For absorbed doses higher than approximately 0.5 Gy, delivered over a short period, some deterministic effects begin to occur in addition to stochastic effects. The dose response relationship for attributable death approximates a sigmoid curve, although the exact shape depends on a number of factors, such as the dose rate and the distribution of the exposure over time. For a dose to the whole body of approximately 3 Gy, the probability of death is about 0.5 in the absence of medical attention. For acute doses higher than about 6 Gy, delivered over a short period, practically all irradiated individuals sill suffer an acute syndrome and are likely to eventually die as a consequence of the irradiation.

(61) Limits are used in radiation safety to control the risk to individuals from all stipulated sources of exposure. However, in order to establish requirements to constrain exposure to individuals from a particular source, the Commission has recommended the use of constraints in the process of optimization, which are source related and should be established in a manner such that the sum of the risks from all relevant sources does not exceed the individual limit. For the treatment of potential exposure, the Commission recommends that the limits of risk be of the same order of magnitude as the health risk implied by the dose limits for normal exposures. However, the dose limits themselves are not applicable to potential exposure situations.

(62) Constraint for individual risks from potential exposure situations can be obtained by constraining the probability from specific potential exposure scenarios or event sequences leading to exposure, or by constraining the magnitude of the exposure or both, i. e., by limiting the probabilities of doses being incurred and/or the doses themselves. This may be achieved through measures... . .

From items 29 and 30, we see that the dose response relationship is linear up to 0.1 Sv, quadratic up to 0.5 Gy and a sigmoid curve over up to 3 Gy, where the probability of death is about 0.5 in the absence of medical attention.

In this case, we can associate approximately the three reference levels of §1 Law n° 8270 mentioned above, with these three different dose ranges using what was stated in item 61 of the ICRP publication (1), i. e., the limit of risk must be of the same order of magnitude as the health risk implied by the dose limits for normal exposure.

Following this reasoning, if the individual has an attributed dose for the potential risk above 1/10 up to 1 of the worker's annual limit dose for normal situations, he will receive the maximum additional value. If he receives an attributed dose for potential risks between the annual dose limit for the public and 1/10 of the worker's annual dose limit, he will receive the intermediate additional value. For smaller attributed doses for the potential risks, he will receive the minimum additional value.

There is still one more factor to be considered: the area occupancy time. This factor offers the probability to meet the person in the area at the moment of the accident. In this case, the area occupancy factor, T, used by the National Council on Radiation Protection (2) of the United States of America.

If the individual has an attributed potential dose based on potential risk above 1/10 of the annual dose limit for workers and remains in the workplace for more than 1/16 of the normal weekly working time (40 hours /week), he has a right to the maximum additional value. If the individual remains in the workplace for less than 1/16, but more than 1/80 of the weekly working time, he has a right to the intermediate additional value. If the fellow remains less than 1/80 of the weekly working time, he has a right to the minimum additional value.

The same reasoning is applied for the person that, according to the attributed potential dose has a right to receive the intermediate additional.

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HOW TO DEMONSTRATE ADEQUACY OF PROTECTION AGAINST A CORE MELT

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INTRODUCTION

After the Chernobyl accident the public - and consequently the politicians - in western countries requested improvements in safety for future reactors even in those designs where the type of accident which had destroyed the Chernobyl plant is excluded by fundamental physics.

When the major German and French suppliers of nuclear power stations, Siemens and Framatome, decided to develop jointly a next generation reactor type, this political "request" had to be taken into account.

It was decided to include safety features to mitigate the consequences of a core melt - the severest type of accident in a western light water reactor - should it occur despite the many other safety features which are included in this design in order to reduce the probability of occurrence of this type of event to extremely low values.

The question arose: How to demonstrate the adequacy of this additional protection? It was evident that the methodology proposed by ICRP namely to demonstrate that the risk of individual members of the public caused by "probabilistic events" is sufficiently low, could not be used: Due to the low probability of occurrence the contribution of this kind of accident to the risk of any average member of a critical group would already be sufficiently low even if there were no additional countermeasures. In addition, this approach would not cover severe societal effects potentially caused by such an accident.

We therefore introduced a different methodology in order to demonstrate the adequacy of additional design features which are only required "just in case": The consequences of such an unlikely but nevertheless very severe event shall be restricted to the plant itself. Severe consequences outside the immediate vicinity of the plant shall be excluded by the design.

METHODOLOGY

In order to cover individual as well as societal consequences we used the following methodology: Despite its low probability a severe accident - a core melt - was postulated to occur. In such a "postulated de facto situation" countermeasures outside the plant may become necessary depending upon the release of radioactive substances to the environment. We intended to show that these releases are so low that there is "no need for stringent countermeasures " outside the immediate vicinity of the plant .

The need for stringent countermeasures could then be evaluated using the advice given by ICRP for "de facto situations" /1/. Countermeasures which could have a severe impact on society were considered "stringent": This includes the evacuation of the inhabitants of whole communities (i.e. the next town) in order to restrict radiation exposure during the emission of radionuclides and the necessity to relocate populations because of radioactive substances deposited on the ground.

Considering the fact, that these provisions are additional to those which shall in the practice exclude this type of accidents, temporary restrictions in the commercialization of foodstuff were not considered to be "a stringent countermeasure": This would mainly be an economic issue (small compared to the value of the plant which is also at stake in this postulated event). In addition, this economic loss would only arise should the accident occur in the growing season. In order to quantify the term "temporary", an unrestricted commercialization of foodstuff shall be possible for a "second harvest" (in Europe generally the one in the following year). In analogy to the use of the values given in ICRP publication 63 for evacuation and relocation, we proposed to use the criteria given by the European Community for the commercialization of foodstuff in a de facto situation for a postulated situation also /2/.

ADDITIONAL ASSUMPTIONS

In order to evaluate consequences in the environment in such a postulated case it is not sufficient to postulate the event itself (a core melt). The environmental conditions have to be defined also. If one would use extreme assumptions (with a corresponding low conditional probability of occurrence) this would further reduce the overall probability of having this result in reality. We concluded that "reasonable" assumptions would be those which can be used in German regulation in the evaluation of the adequacy of design features controlling design basis accidents. The main ones are:

1. The event is postulated to occur in the growing season.

2. For each countermeasure (e.g. evacuation, relocation, commercialization of foodstuff produced in a second harvest) we evaluated a corresponding pathway potentially leading to the exposure of members of the public.

3. We concluded that the safety features are adequate if the considered countermeasure was not required at any location outside a defined exclusion area in 95 % of potential atmospheric conditions (given the postulated release).

4. In order to be independent from the characteristics of a specific location (e.g. distribution of the population) it was assumed that persons could be at any location for a unlimited time and that they would be unprotected (no sheltering).

5. Relevant biological and ecological factors were taken identical to those prescribed in German regulation for the evaluation of design basis accidents.

SOURCE TERM

Two different approaches have been used:

1. A "design release limit" was defined and shown that the safety goals stated above are met in the environment if there should be a release at or below this value. As "design release limit" we specified the following values for the total release and (given in parenthesis) maximum values for any 24 hours:

Xe-133 : 1 million TBq (0.1 million TBq); I-31: 2000 TBq (300 TBq); Cs-137: 100 TBq.

Each of these three nuclides represents the behavior of a specific group. The release of other nuclides was then calculated from these values by the use of characteristic data (e.g. the ratio of the core inventory of the various radionuclides after 1 year of full power operation).

This approach allows the safety analysis to performed in two independent steps:

a. Demonstration that safety goals in the environment are met given a release at the design release limit..

b. Demonstration that the safety features of the plant keep the release below the release limit.

Because atmospheric conditions will quite likely change during the release and because consequences in the environment also depend upon the release height additional assumptions concerning these factors were necessary. With respect to a release at the "design release limit" we evaluated consequences for the two extremes: release only at ground level and release only via a 150 m stack.

2. The method described before automatically introduces unnecessary conservatism: consequences in the environment are only evaluated for the "design release limit" and not for the (much lower) "potential release". Because the contribution of different radionuclides to exposure is different, the ratio between design release and potential release is not a direct measure for the reduction in dose.
In an alternative method we therefore calculated release rates as a function of time and the location of the releases for all radionuclides taking into account the major safety features of the new design. For this evaluation a core melt was postulated releasing 100% of the core inventory of noble gases, iodine and cesium into the containment atmosphere. A conservative function describing the consecutive deposition of aerosols was used. A conservative value was also selected for the leak rate from the primary containment to the secondary containment (a so called annulus) taking into account the intended specification for the containment. The evaluation also takes into account the fact that the secondary containment will only be ventilated via filters and the stack. A sub-atmospheric pressure will be provided in this secondary containment for a sufficient period of time in case the operation of the ventilation system would be delayed (e.g. in case of lack of power). This would provide decay time for the radionuclides and would thus reduce the release. It is therefore conservative to assume that the ventilation system operates during and after the postulated event.

TRANSPORT OF RELEASED RADIONUCLIDES IN THE ATMOSPHERE

The potential exposures resulting from given release functions for all relevant radionuclides as described before were calculated for specific immission points. These points were selected in such a way that each could be considered representative for a given area. They were defined by a grid of 72 sector lines and 20 distances. The distances were optimized in order to guarantee that the actual maximum value would be less than 10 % higher than the highest value found in the grid. Because limited countermeasures i.e. the evacuation of a few persons living very close to the plant is considered acceptable under these extreme circumstances an exclusion radius was also defined. We then calculated the potential exposure at each immission point outside the exclusion distance assuming a specific weather sequence during the release. The highest

of these values was then identified. If this highest value is below the corresponding intervention value, no intervention is required at any location. The assumed weather sequence was taken from real data. This calculation could in principle be repeated for a sufficient number of weather sequences all taken from reality. If the considered countermeasure is not required anywhere in more than 95 % of the cases the safety goal is met.

The simple yes/no evaluation with respect to countermeasures, however, does not identify existing safety margins. We therefore evaluated the detailed statistical distribution by answering the question: what is the probability that value x is not exceeded anywhere outside the exclusion area ? From this probability distribution function we interpolated the 95% value. The accuracy of this interpolation could then be improved: In a second run the probability distribution function was evaluated using a more detailed structure in the neighborhood of the approximate 95% value obtained in the preceding evaluation.

Another issue is the number of weather sequences to be evaluated. Computer programs which have been generated in order to evaluate "probabilistic events" tried to overcome this problem by selecting a limited number of weather sequences (in the order of 100) and giving weighting factors to each of them. However, consequences are influenced by a variety of factors: different wind speeds, different wind directions, different precipitation rates and (for long releases) different dispersion conditions. All these factors are more or less likely to change during the release. Their impact on exposure is quite different for the different exposure pathways (e. g. radiation from the cloud or ground contamination) and is also different at different distances from the plant. This becomes even more complicated if simultaneous releases at ground level and at an elevated level occur. The determination of one weather sequence to be a sufficiently good representative with respect to all these aspects for many more situations is therefore certainly a source of inaccuracy and possibly even bias. In order to avoid any discussion on that issue we therefore calculated the consequences of all weather sequences within a given period of time. We found that a period of one year (vegetation period only, i.e. some 4000 weather sequences with hourly weather data) results in a reliable statistical evaluation.

EXCLUSION AREAS

Countermeasures could easily be performed on small areas. Such an area could therefore be excluded from the requirement. The identification of such an area, however, may be different for different types of countermeasures. Evacuation of a few persons could be performed in order to limit exposure by the cloud and by inhalation. Therefore it should be performed before or at the beginning of the release. Because variations in wind direction are difficult to predict with certainty in a de facto situation, it would be necessary to evacuate all persons within a circle around the plant. In order to find the necessary size of such an exclusion radius we performed the calculations for a large number of different exclusion radii..

In contrast to evacuation, relocation would be performed after the release and would be based on measured dose levels. Persons living on an area on which a specified dose would be exceeded would then be relocated independent from the distance to the plant. If this area is sufficiently small it could then also be excluded from the requirement. We therefore evaluated the correlation between doses caused by deposited radioactive substances and the corresponding areas independent from its location and determined the statistical distribution in order to identify the values which are not exceeded with a probability of 95 %.

CONCLUSIONS

Criteria which were developed for intervention situations may also be used for demonstrating the adequacy of a design with respect to probabilistic events. In this case an extreme event i.e. a core melt is postulated to occur irrespective of its probability of occurrence. We were able to demonstrate that the European Pressurized Water Reactor jointly designed by Electricité de France, Framatome and Siemens would not require stringent countermeasures outside an exclusion area in this postulated case. Furthermore, if all safety features of the design are taken into account, no area outside the plant fence has to be excluded from the requirement.

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Amtsblatt der Europäischen Gemeinschaften Nr. L371/11 and Nr L 211/1

PROGRESS TOWARDS A CONVENTION ON THE SAFE MANAGEMENT OF RADIOACTIVE WASTE

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INTRODUCTION

The Convention on Nuclear Safety (1) was prepared during the period 1992 to 1994 and after consideration by a Diplomatic Conference in June 1994 was opened for signature at the General Conference of the IAEA in September 1994. The matter of the safety of radioactive waste was discussed many times during the development of the convention but it was eventually decided to restrict the coverage to matters concerned with nuclear safety of land-based civil nuclear power plants and those aspects of radioactive waste management directly connected with and carried out on the same site as the power plant. In the preamble to the convention, however, item (ix) affirms "the need to begin promptly the development of an international convention on the safety of radioactive waste management as soon as the ongoing process to develop waste management safety fundamentals has resulted in broad international agreement".

In September 1994, the General Conference of the IAEA also passed a resolution inviting the Board of Governors and the Director General to commence preparations for a convention on the safety of radioactive waste management. The Director General therefore organized a preparatory meeting of experts from member states to discuss the basic concepts and the possible scope of such a convention and to examine working methods and the procedures for its preparation. This meeting which took place in February 1995 prepared a paper entitled "Inventory of Issues Raised" and proposed that the appropriate mechanism would be the setting up of an open-ended group of legal and technical experts to prepare the convention.

The Safety Series document at the fundamentals level on the principles of radioactive waste management was approved by the Board of Governors in March 1995 (2) and all the initial preconditions for starting work on the convention were then fulfilled.

MEETINGS OF THE GROUP OF EXPERTS

First Meeting

The first meeting of the group of legal and technical experts was held in July 1995. There were 128 participants from 53 countries and observers from 4 international organizations. The meeting elected Professor Alec Jean Baer from Switzerland as its Chairman.

In an introductory general discussion, experts agreed that the Convention on Nuclear Safety was to be considered as a model. The intention was to develop a "sister" convention on radioactive waste safety: notably it should also be an "incentive" convention and should follow a similar structure; it should contain a reporting requirement to a meeting of Contracting Parties and rely for its implementation on a peer review process thus acknowledging the sole national responsibility for radioactive waste management activities. It should so far as possible take over where the Convention on Nuclear Safety ceases to apply so as to avoid any gaps in coverage. It was also noted that consideration should be given to including the substance of the Code of Practice on the International Transboundary Movement of Radioactive Waste (3).

In the preliminary discussions there was agreement that the convention should apply to the full range of radioactive wastes as described in the Waste Management Fundamentals, namely "liquid", "gaseous" and "solid form" wastes but it was recognized that the scope would need careful definition, especially in determining what materials would be excluded or otherwise be deemed to not be covered by the convention. Materials presenting difficulties in this respect included those wastes containing low activity concentrations of radioactive materials, especially if these were only naturally occurring radioactive materials. Another difficulty of a different kind was whether the convention would apply to the storage of spent fuel not intended for disposal as waste or for which the intention was not yet firm. Although it was clear that such spent fuel did not fall under the definition of "waste" i.e. material for which no further use is foreseen, it was also clear that the safety provisions during interim storage should be the same whatever the destination of the spent fuel when retrieved from storage. A further item needing resolution was the point at which materials originating from military applications and becoming wastes would come under the convention coverage.

During the discussions the technical articles of the Convention on Nuclear Safety were reviewed in some detail. A number of provisions were found that could readily be transferred to the new convention by appropriate adjustments and changes of wording. For some, however this was not the case and there were some aspects of waste safety that had no counterpart in the Convention on Nuclear Safety.

The meeting also considered the suitability for incorporation in the convention of each of the principles in the Safety Fundamentals document (2) and the Safety Standard on the establishment of National Systems (3). There was no substantial dissention from the general message embodied in the major statements of principle but it was clear that care would be needed over the wording to be used to express them in a convention.

The meeting concluded by requesting the Chairman to produce a first draft for consideration at the next meeting.

Second Meeting

The second meeting of the open-ended group of technical and legal experts on the Convention on the Safety of Radioactive Waste Management took place from 4 to 8 December 1995. The meeting was attended by participants from 52 countries and observers from 3 intergovernmental organizations.

The meeting carried out a detailed review of a first draft of the convention, which had been prepared by the Chairman taking into account comments from "Friends of the Chair"- a small group based on geographical distribution. General agreement was obtained that this draft provided a very good basis for discussion, and the meeting proceeded to review the document on an article-by-article basis. The group agreed that the title of the draft convention should be identical to that used in the Preamble to the Convention on Nuclear Safety, namely "Convention on the Safety of Radioactive Waste Management". It was also agreed that the title should not specifically refer to spent fuel.

During the course of the meeting, various delegations submitted general position papers in addition to preparing detailed drafting comments for improvement to particular articles. These were considered to be very constructive in tone, and the meeting was characterized by a willingness to appreciate differing points of view and to suggest compromise wording where necessary to overcome particular areas of concern. A good example was the proposal to simplify and combine various definitions.

The meeting agreed that the Chairman should prepare a revised second draft , based on the numerous proposals and comments made during the course of the meeting, for distribution to Member States by mid February 1996. This would then form the basis for the deliberations of the next meeting of the open-ended expert group, which has been scheduled for 25-29 March 1996 in Vienna. Once the Group has an agreed text this will be considered by a Diplomatic Conference before being endorsed by the IAEA Board of Governors and opened for signature.

CONCLUSIONS

Based on the adoption of the Convention on Nuclear Safety as a model, the general style, format, content and procedures of the Convention of the Safety of Radioactive Waste Management have been agreed by the Group of legal and technical experts responsible for preparing a draft. There remain many specific questions to be resolved but the meetings so far have been characterized by a willingness to seek solutions and expedite progress so the prospects remain encouraging.

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RADIATION MONITORING SYSTEM FOR ASTRONAUTS

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ABSTRACT

Astronauts in space are constantly under the bombardment of radiation particles from trapped electrons, and trapped proton. In addition, cosmic rays, while penetrating the spacecraft shell, generate secondary radiation of neutrons. As astronauts' stay in space is getting longer, the need for a real-time radiation monitoring device has become critical. ThermoLuminescent Dosimeter (TLD), used onboard both the MIR and the Space Transportation System (STS), cannot provide real-time dose reading. This paper describes a real-time direct read-out device, currently under development, which can measure skin, eye, and Blood Forming Organ (BFO) doses separately.

INTRODUCTION

The radiation environment found in space, particularly Low Earth Orbit (LEO), can be hazardous to the well being of the mission crew. Figure 1 shows the simplified space environment both external and internal to a manned spacecraft. There is a complex mixed radiation field both inside and outside the spacecraft. The types of radiation inside a spacecraft ranges from primary particles, e.g. trapped high energy electrons and protons, cosmic rays (light and heavy ions) to secondary particles, e.g. X-rays, gamma rays, neutrons. The majority of the radiation dose arises from high energy electrons and protons, with protons dominating for shields of greater than 1g/cm^2 (i.e. 4 mm Al). A study of relative doses for different missions has been presented in detail in ESA's *PARIS* study (1). Measured doses with TLDs are

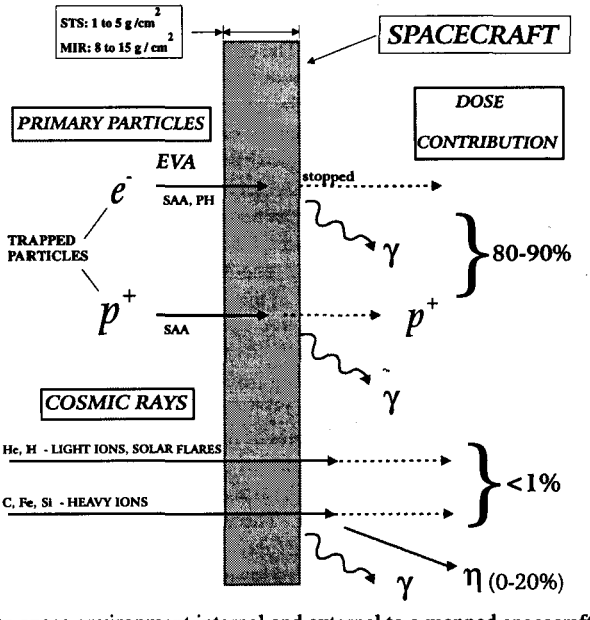

Figure 1 The space environment internal and external to a manned spacecraft

in the region of 1 mSv/day for STS, 0.3 mSv/day for MIR and 0.2 mSv/hour for Extra Vehicular Activity (EVA). These number show that the lifetime limit can be achieved much sooner for those astronauts who undertake EVA. Fry has found that (2), depending on the length and time of the mission as well as the orbit inclination and altitude, it is quite possible that an astronaut lifetime dose can be achieved in as little as 6 missions each lasting 6 months. During EVA the limiting dose could well be that absorbed by the astronaut's skin. It is imperative, especially with the construction of Space Station Alpha within the next decade, where astronauts are expected to work many hours in EVA, that the dose not only to the BFO be monitored but also that experienced by the skin and to the eye.

In 1992 a series of space radiation experiments were conducted using Thomson & Nielsen (TN) MOSFET dosimeters. Three TN MOSFET dosimeters were placed on the MIR space station to measure the absorbed dose within the interior of the spacecraft. The orbit apogee/perigee was 419/399.8 km at an inclination of 51.64°. The units were returned to earth after 96 days, 268 days and 430 days measuring 0.025 Sv, 0.075 Sv and 0.135 Sv respectively. The MOSFET's agreed with Russian TLD measurements to within 10% (3).

SPACE RADIATION ENVIRONMENT

The most important characteristic of the environment encountered by spacecraft in LEO is that several times each day it must pass through the particles trapped in the Van Allen belts. These particles are primarily electrons and protons which are trapped in the Earth's magnetosphere. For LEO there are three areas where particle fluence is of concern: the two polar regions and the South Atlantic Anomaly (SAA) The SAA occurs in a region over the South Atlantic and is due to the tilt of the Earth's magnetic axis which causes the inner edge of the trapped belts to be encountered at relatively low altitudes. Trapped electrons range in energy from a few keV to 5 MeV while trapped protons range in energy from 30 MeV to several hundred MeV.

The level of trapped particle flux varies greatly with orbit inclination and altitude. The amount of protection the geomagnetic field provides a spacecraft from cosmic rays and solar flares is also dependent on the inclination and to a smaller degree the altitude of orbit. As the altitude increases, the exposure to cosmic rays and solar protons gradually increases. The effect of orbit inclination is much more dramatic. As the inclination increases the spacecraft spends more time in regions accessible to these particles. In a polar orbit the spacecraft is no longer protected by the geomagnetic field lines and is fully exposed to cosmic ray and solar flare particles in addition to trapped electrons and protons.

Space Station Alpha's proposed orbit is approximately 400 km at an inclination of 51°. The average shielding afforded by an U.S. astronaut EVA suit is (1): arms, 0.2 g/cm^2, body 0.5 g/cm^2 and head 0.9 g/cm^2. By definition the shielding from the human body to the skin, eye, and BFO is 7 mg/cm^2, 0.3 g/cm^2 and 3.0 g/cm^2, respectively. Using these figures, the total shielding for arms, skin, eye, and BFO are approximately 0.207 g/cm^2, 0.507 g/cm^2, 1.2 g/cm^2, and 3.5 g/cm^2 , respectively. Figure 2 shows the resultant dose as simulated for an astronaut in EVA at the Space Station orbit.

ASTRONAUT RADIATION MONITOR

Thomson & Nielsen Electronics, with funding from the Canadian Space Agency, is developing a prototype Astronaut Radiation monitor. The TN MOSFET dosimeter technology has already found applications in nuclear and medical fields. Uniqueness of the sensor is its thin active region (< 0.001 mm) and its sensitivity. By using shields of various thicknesses over the sensor, the dosimeter can be simulated to measure the dosage to the skin, eye, and BFO. The size of the monitor is expected to be less than 15 cm × 10 cm × 5 cm with a mass of not more than 800 g. The monitor is designed to run on a 10 V power source (battery) and can operate continuously over 21 days. The monitor consists of three MOSFET dosimeters to measure BFO, eye, and skin dose with a lower limit of detection of 0.4 mSv. The monitor also has a charged particle detector to monitor dose rates as low as 0.01 mSv/h.

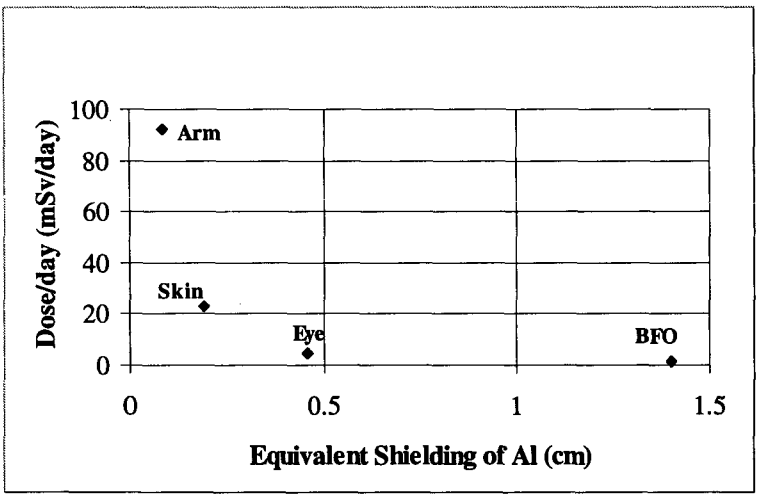

Figure 2 Simulated dose level for an Astronaut in EVA for the proposed Alpha Space Station orbit.

The monitor takes dosimeter readings once per hour, storing the measurement in memory along with the time and temperature. The dose rate detector integrates over 1 minute interval storing data in memory along with the time of measurement. In this manner the radiation environment encountered by the astronaut will be monitored and mapped. Data is obtained by interfacing the monitor to a personal computer and downloading into a computer program. It is expected that the first flight unit will be built, tested and be ready to fly in the STS in 1997.

ACKNOWLEDGEMENT

The authors wish to thank the input to the project by Dr. Steve MacLean and Dr. Alan Mortimer of Canadian Space Agency, and Dr. Ken Shortt of National Research Council. The work is supported by the Space Systems Directorate and Canadian Astronaut Program Office of the Canadian Space Agency, Contract # 9F028-5-5107/01-XSD.

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POSSIBLE IMPACT OF ICRP 60 RECOMMENDATIONS ON MEDICAL PRACTICE

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Introduction

Three changes recommended by the I.C.R.P. are likely to have a significant impact on medical procedures, the revised occupational dose limits, which include a new five year cumulative effective dose limit of 100 mSv and also a lower limit for exposure of the skin, the lower annual dose limit of 1 mSv recommended for members of the public and the recommendation that the dose to the foetus during pregnancy should be limited to 1 mSv. The proposed new 100 mSv in 5 years occupational dose limit has been generally presented as the most important proposal which was made by I.C.R.P. Many hospital administrators have reviewed the implications of such a dose limit for their own organization, found from past dose records that very few staff appear to be at risk of exceeding the proposed five year limit, and filed the report, or passed it on to the hospital Radiation Safety Officer, assuming that they can safely ignore it. In practice however the other major changes summarised above may have more impact on the operation of hospitals than the proposed new occupational dose limit. Some changes in working practices which hospitals may need to make to meet ICRP recommendations are reviewed below.

The Proposed New Occupational Dose Limits

Most occupationally exposed radiation workers in a hospital would regard it as very exceptional if they received a radiation dose greater than 20 mSv per year. Such exceptions may occur after there has been a major change in the workload which has not been reflected in corresponding improvements in staffing and facilities, procedures for reducing doses to normal levels are then self evident. However a few categories of workers receive consistently higher doses on an ongoing basis in some large hospitals. In diagnostic radiology these categories include staff extensively involved in cardiac catheterisation procedures and nurses in paediatric hospitals who frequently hold children during X-ray procedures. In nuclear medicine radiopharmacists producing large quantities of labelled drugs will need to handle very large activities and will receive significant whole body doses if the radiopharmacy has not been adequately designed to minimise their exposures. Nuclear medicine technicians in busy departments also sometimes accumulate relatively large doses when dispensing is carried out in house and a properly equipped dispensary is not available. Both these groups are also liable to receive significant finger doses, and meeting the proposed new limit for skin dose averaged over 1 cm^2 (ICRP 60 paragraph 173) is likely to require some changes in working procedures. In radiotherapy departments staff involved in brachytherapy also frequently receive both effective doses, and finger doses, which are not inconsiderable.

Hospital procedures with respect to the classification of staff as radiation workers vary widely; but in my experience nurses and orderlies have not generally been classified as radiation workers and they are therefore, with a few exceptions, subject to the same dose limits as members of the public. Nevertheless nurses caring for iodine therapy patients may receive significant whole body doses, and attention may also need to be paid to the dose received by orderlies responsible for transporting patients who have just received radioactive injections. In some children's hospitals which are not equipped with adequate child restrainers, or where clinicians oppose the use of restraining devices, nurses are frequently called upon to hold children during radiological examinations and this may lead to significant radiation doses being incurred. Doses to nurses are also possible when working in wards where there is a considerable use of mobile X-ray equipment, or where patients with implanted radioactive sources have been placed. In all these cases the new I.C.R.P. recommendations may necessitate such members of staff being formally designated as radiation workers subject to the new occupational dose limits rather than the existing limit for members of the public, or at least being provided with monitoring devices to demonstrate that this is not necessary.

Radiation Exposure of Members of the Public

Radiation facilities in hospitals have generally been designed so that the maximum dose which might be received in an area accessible to the public does not exceed the present public dose limit of 5 mSv/y. In principle, if this limit is reduced by a factor of five the relevant areas in every hospital should be provided with additional shielding. Equally there are likely to be other areas of the hospital occupied by general staff members where there is a possibility of exposing the workers concerned to annual doses in the range 1-5 mSv/y and these will therefore also need additional shielding. Such areas can include canteens, physicians rest rooms, teaching rooms and office space. Paragraph 168 of ICRP 60 indicates clearly that the Commission expects such modifications to be completed as early as possible. Shielding requirements for diagnostic radiology suites are calculated using a formula which includes relevant data on all the x-ray sources present, the work loads for each, the occupancy factors for adjacent areas and the existing structural shielding. Although many rooms with low work loads have been more than adequately shielded to allow for increasing future work loads, this has not been a universal practice and, when the recognised formula has been rigidly applied, existing shielding may be quite inadequate to meet the recommended new limits. This is not a technical problem as additional thin sheets of lead can quickly cure any deficiencies, but the economic impact of introducing such requirements for a significant proportion of the existing radiology departments would be enormous. Some calculations made by Health Canada suggest that this problem may be most severe for dental operatories currently shielded only by gypsum plasterboard, but that standard radiographic facilities with workloads of around 200 mA-min/week may require more than an additional 0.5 mm of lead shielding.

Similar concerns apply to most existing nuclear medicine and radiotherapy departments, but with two important differences. The number of departments affected is very much smaller, but the problem of adding additional shielding within the constraints of the existing building may be very much greater because of the higher energy of the radiation of concern. For example there are relatively few 20 MeV linacs, but retrofitting the shielding of the bunkers in which these are installed to increase the beam attenuation by a factor of five would often be almost impossible. Radiotherapy departments may also be faced with a need for the installation of additional shielding where their brachytherapy sources are stored, and around the beds of patients with implanted sources.

Additional problems will affect nuclear medicine departments. These have all developed rules for allowing the discharge of patients after the administration of significant quantities of radioactivity. Such rules are generally based on limiting the dose which could be received by any member of the public outside the hospital to less than 5 mSv. These rules will have to be rewritten to reflect the new I.C.R.P. limit once this has been introduced. The Commission has introduced one concession which may help in this regard, the greatest problems will arise in the patient's home where it is likely that the patient will spend the longest periods of time in the vicinity of other people. In paragraph 139 of ICRP 60, it is stated that any exposure knowingly and willingly received in the course of supporting and comforting patients, may be classified as a medical exposure that is not subject to the normal dose limits. Other rules that limit the activities that can be discharged from busy departments, or from the homes of outpatients may also have to be modified. An indirect effect of the new public dose limit in ICRP 60 is likely to be a reduction in the permissible concentration of radionuclides in effluent, and this would have very important consequences for the operation of nuclear medicine departments. Calculations made by one Canadian hospital suggest that such changes would necessitate delaying the discharge of all iodine therapy patients, at a cost to the hospital of several million dollars per year.

Radiation Doses Received by Pregnant Workers in Hospitals

The I.C.R.P. recommendations have raised considerable concerns among female nuclear medicine technologists in hospitals who fear that it may not be possible for them to continue normal work during a pregnancy if these new recommendations are implemented. They believe that should this prove to be the case it would lead to the preferential employment and promotion of male staff, and would therefore have a very adverse effect on their career prospects. Similar concerns are felt by some X-ray technologists who are routinely involved with high dose procedures such as cardiac catheterisation. Other pregnant workers, such as nurses, orderlies or even cleaning staff, have become aware that a large reduction in the permitted dose during pregnancy is being

recommended, and this has led to concerns that the doses which they are currently experiencing may be too high and may be harmful to their expected child. Explaining to all these workers that such fears are generally unwarranted has placed a considerable additional burden on many hospital Radiation Safety Officers.

Optimisation in Hospitals

One of the most important points stressed in this I.C.R.P. report is the need to improve optimisation procedures in hospitals (ICRP 60 paragraph 180). Any attempt by a hospital to comply with the various recommendations necessitates a complete review of its ALARA program, with particular attention being paid to the available dose data and to ways in which existing dose levels for both staff and members of the public could be reduced. This could not be restricted to consideration of occupational exposures only as the I.C.R.P. report places great emphasis on the need to reduce the dose delivered to the patient during commonly used diagnostic procedures. Some of the principal areas in which it appears possible to improve the optimisation of the uses of ionizing radiations in hospitals as recommended in this ICRP report, are reviewed below:

(a) Radiotherapy

External beam treatment bunkers have all been designed to ensure that the dose which could be received by members of the public (including ancillary hospital staff) does not exceed 5 mSv/y. A requirement to reduce this by a factor of five would involve additional shielding of the bunkers that may be felt to be impracticable. An alternative in some circumstances may be to address specific concerns individually, for example where rooms immediately adjacent to a treatment bunker are continually occupied by ancillary hospital staff, problems could be alleviated by relocating accommodation so that these rooms are restricted to uses involving low occupancy factors such as storage. Although no major problems should be encountered in controlling the doses received by occupationally exposed staff associated with external beam treatment procedures, this does not apply to brachytherapy where there is often a long history of relatively high radiation doses to the staff involved. Optimisation may require more hospitals to adopt remote afterloading techniques to reduce these doses.
The doses received by other staff; such as porters, orderlies, nurses and nursing auxillaries who are not classified as radiation workers but who may be exposed to radiation from brachytherapy sources; will also have to be reviewed very carefully. In most instances additional shielding would probably help to meet optimisation requirements, but it may also be important to review whether some of these workers should be reclassified as occupationally exposed radiation workers. If this becomes necessary additional problems arise as soon as one of the staff concerned became pregnant and has to comply with the ten times lower dose limit to the foetus which the I.C.R.P. are now recommending. Serious problems would result if these considerations lead hospital administrators to preferentially appoint male staff to some of these positions.

(b) Diagnostic Radiology

The most important aspect of optimisation in diagnostic radiology is the reduction of patient doses associated with different procedures by improved quality assurance procedures. This is too large a subject to discuss further here, but there is increasing recognition that patients have the right to be given dose measurement data relating to any procedures that they undergo, and in many instances this should also include comparative data generated in other centres. In the case of staff, optimisation generally involves a careful review of the maintenance procedures for equipment and of the use of personal shielding such as aprons, gloves and lead spectacles.

(c) Nuclear Medicine

As with diagnostic radiology, optimisation will involve attention both to doses delivered to patients and those received by staff. An effective quality assurance program for imaging equipment is clearly essential. It is also necessary to assess whether the availability of imaging equipment is adequate. In an expanding department it may become necessary to reduce scanning times by increasing the activity administered. It is easy for such temporary expedients to become permanent practice. Attention also need to be paid to the optimisation of doses outside the hospital which result from the discharge of treated patients; at the minimum such patients must be given adequate instructions on elementary radiation hygiene and on the precautions that they should follow to protect family members. In the case of doses to staff members the most promising (and expensive!) approach to optimisation is usually a careful review of both equipment and working practices in the radiopharmacy and the dispensing area.

CRITERIA FOR THE RELEASE OF PATIENTS ADMINISTERED RADIOACTIVE MATERIAL

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INTRODUCTION

In June, 1994, the U. S. Nuclear Regulatory Commission proposed amendments to its regulations concerning criteria for the release of medical patients administered radioactive material. The existing criteria are that the release of patients from hospitalization may be authorized if the activity in the patient is less than 1,110 megabecquerels (30 millicuries) or if the dose rate at a distance of 1 meter from the patient is less than 0.05 millisievert (5 millirems) per hour. The proposed new criteria for patient release would be based on the potential dose to individuals exposed to the patient. Under the proposed new criteria, the release of a patient could be authorized if the total effective dose equivalent to any other individual from exposure to the patient is not likely to exceed 5 millisieverts.

RECOMMENDATIONS OF ADVISORY BODIES

The 5-millisievert value is consistent with the recommendations of national and international advisory bodies. In the U. S., the National Council or Radiation Protection and Measurements (NCRP) recommends public dose limits of 1 millisievert for continuous or frequent exposure and 5 millisieverts for infrequent exposure. The International Commission on Radiological Protection (ICRP) recommends that the limit for public exposure should be expressed as an effective dose of 1 millisievert in a year, except that, in special circumstances, the dose could be higher in a single year provided the average over 5 years does not exceed 1 millisievert per year. The NRC's radiation protection regulations incorporated the 1-millisievert long-term objective as a dose limit and included a provision to allow for a 5-millisievert limit on an occasional basis. Section 20.1301(c) provides that an annual dose of up to 5 millisieverts is acceptable if there is a need for it and if steps are taken to reduce the dose to as low as is reasonably achievable. In the case of released patients, it would be unlikely for a single individual exposed to a patient to receive a significant dose from more than one therapeutic administration because large therapeutic doses are rarely administered more than once within several years.

METHODOLOGY FOR CALCULATING EXTERNAL GAMMA DOSE

The dose to total decay $D(\infty)$, is calculated using the following equations. For radionuclides with a half-life greater than 1 day

$$D(\infty) = \frac{34.6\Gamma Q_o T_p (0.25)}{(100\ cm)^2}, \quad (1)$$

and for radionuclides with a half-life less than 1 day

$$D(\infty) = \frac{34.6\Gamma Q_o T_p}{(100\ cm)^2}, \quad (2)$$

where Γ = exposure rate constant for a point source, R/mCi-h at 1 cm,

Q_o = initial activity of the point source in millicuries, at the time of release,

T_p = physical half-life in days.

Equation 1 assumes, for radionuclides with half-lives greater than 1 day, that the individual likely to receive the highest dose from exposure to the patient would receive a dose of 25 percent of the dose to total decay (0.25 in Equation 1) at a distance of 100 centimeters (1 meter). For radionuclides with half-lives less than 1 day, the factor 1.0 is used in Equation 2 because the assumption that the time that individuals will spend near the patient will be limited is not valid when most of the dose is delivered in a relatively short time.

Doses among individuals who may come in contact with a released patient are highly variable and reflect the crucial, but difficult to define, parameters of time, distance, and shielding. Based on time and distance considerations, it is reasonable to conclude that for the overwhelming majority of released patients, the maximally exposed individual is likely to be the primary care-provider, a family member, or any other individual who spends significant time close to the patient.

Based on time, distance, and shielding factors, which describe normal lifestyles of the U.S. population, it is highly unlikely that doses equal to spending 100 percent of time at a distance of 1 meter from a patient would result to any individual including a patient's spouse. As a standard medical practice, patients undergoing therapeutic treatments with radiopharmaceuticals are given firm instructions, both verbally and in writing, regarding basic principles on how to minimize doses to other individuals.

Given all considerations, a reasonable estimate of the maximal likely dose to an individual exposed to a patient is 25 percent of the dose to total decay at a distance of 1 meter. The selection of an occupancy factor of 25 percent at 1 meter for estimating maximal likely exposure is based on judgment of time-distance combinations that are believed likely to occur when instructions to minimize time spent close to the patient are given.

The occupancy factor of 0.25 at 1 meter is also supported by empirical data. Harbert and Wells (1) monitored the external dose of 8 family members of 3 patients treated for thyroid carcinoma using iodine-131. The actual doses were far below the calculated doses for an occupancy factor of 25 percent, indicating that the model generally provides a conservative estimate of the dose.

Harbert and Wells (1) also measured the external doses to 11 family members of seven hyperthyroid patients. In each case, the measured doses were at least a factor of 10 below the doses predictedd by Equation 1 using an occupancy factor of 0.25 at 1 meter.

Jacobson et al. (2) measured the external doses to 10 family members of 7 iodine therapy patients. In each case except one, the external dose to the family member was below that predicted by Equation 1 using an occupancy factor of 0.25 at 1 meter and well below 5 millisieverts. In the case of the exception, the family went on a extended vacation spending much of the time together in an automobile. This demonstrates that if reasonable efforts to maintain distance are not made doses can be higher than predicted by Equation 1.

Buchan and Brindle (3) monitored the doses of 54 family members of patients who underwent iodine therapy for hyperthyroidism. This study is interesting because no instructions on minimizing dose were given. Thus, the results can be taken to represent the doses that would be received if no instructions were given or if instructions were totally disregarded. The effective occupancy factor at 1 meter was less than or equal to 0.25 in 45 of the 54 cases (83 percent). Thus, even in the complete absence of instructions, the occupancy factor at 1 meter was usually less than 0.25.

In conclusion, both empirical measurements and professional judgement support an occupancy factor of 0.25 at 1 meter as a generally conservative value. Using this value in Equation 1 should generally overpredict the dose even if instructions are not given or are not strictly followed.

PERIOD FOR INTERRUPTION OF BREAST-FEEDING

The proposed new criteria would apply to breast-feeding infants. The regulation would require that breast-feeding women receive instructions on the interruption of breast feeding if the dose to a nursing infant could exceed 1 millisievert if there were no interruption of breast feeding. The doses were calculated for newborn infants; doses to one-year-old infants would be less than half the doses to newborns. The external dose, typically small relative to the internal dose, is considered separately.

External doses were calculated using an occupancy factor of 0.16 and an effective distance from source to receptor tissue of 0.2 meter.

The decision on whether instructions would be required is based on the sum of the maximum value of the internal dose range for the newborn infant plus the external dose assuming no interruption of breast-feeding. The duration of interruption is selected to reduce the maximum dose to a newborn infant to less than 1 millisievert. The actual doses that would be received by most infants would be much less. The physician may use discretion in the recommendation, increasing or decreasing the duration of interruption somewhat depending on the mother's concerns about radioactivity or interruption of breast feeding.

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REGULATORY REQUIREMENTS FOR THE USE OF CONSUMER PRODUCTS CONTAINING RADIOACTIVE SUBSTANCES

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INTRODUCTION

In almost 100 years since the discovery of radioactivity, the properties of radioactive materials have been exploited in products such as clocks and watches incorporating luminous paint which are freely available to members of the public. Over time, regulatory authorities have felt it necessary to apply some degree of control to the supply and use of such products in order to protect public health. In many areas of radiation protection, national authorities take note of international recommendations when developing national standards, but the existing detailed guidance of the International Atomic Energy Agency (IAEA) for consumer products is incomplete and out of date [1]. Recently, a thorough revision of the International Basic Safety Standards (BSS) has occurred [2], which has prompted a review and revision of the related guidance published by the IAEA. A draft Guide on Regulatory Requirements for the Use of Consumer Products Containing Radioactive Substances has now been completed and is currently under review within the IAEA's system for development of documents in its Safety Series of publications.

OBJECTIVE AND SCOPE OF THE DRAFT GUIDE

The major objective of the draft Guide is to provide guidance on interpretation of the BSS in developing national regulatory requirements for the control of consumer products containing radioactive materials. In this context, a consumer product is taken to be anything supplied freely to members of the public and which, once acquired by the consumer, is essentially beyond further regulatory control. An additional aim is to provide guidance which promotes harmonization of policy between countries so that difficulties in international trade are avoided. The Guide applies to the distribution, sale, use and disposal of all consumer products in which radioactive materials have been deliberately incorporated, but not to their manufacture, which is covered by other Safety Series guidance.

PRINCIPLES OF PROTECTION

The Guide follows the fundamental radiation protection philosophy of the BSS and is based on the principles of justification of practices, optimization of protection and limitation of individual dose. It treats the supply of consumer products as a practice. If the incorporation of radioactive materials into a consumer product can be justified, the Guide requires a system of regulatory control to be applied to the practice. The system described includes the possibilities of exclusion, exemption and authorization of the practice, where authorization may take the form of licensing or registration, and is intended to provide an appropriate balance between measures that are necessary to protect the public and over-regulation.

Exposures to radiation from consumer products may be excluded from regulatory control if they arise from natural radioactivity in raw materials, for example, and public use of the product does not lead to pathways of exposure which are undesirable and amenable to control. The practice of supplying consumer products may be exempted from regulatory control if the consequent exposures are sufficiently small and if the product supplied is inherently safe. Specifically, if individual effective doses are expected to be less than 10mSv in a year and the collective dose is expected to be less than 1 man-Sv, the practice may be exempted without further consideration. It is anticipated that the majority of consumer products would fall into this category. If these conditions are not met, the practice might still be exempted if exemption can be shown to be the optimum regulatory option.

When supply of consumer products cannot be exempted from regulatory control, the supplier is required to meet specified conditions. The practice would normally be registered, as licensing is unlikely to be necessary for products made freely available to members of the public.

REQUIREMENTS FOR REGULATORY CONTROL

The Guide specifies requirements for the regulatory authority in establishing the regulatory status of the practice. In summary, the regulatory authority should require an initial notification from all suppliers of consumer products containing radioactive materials, make an assessment of the practice based on the decision tree illustrated in Figure 1, and then exclude, exempt or register accordingly. To avoid unnecessary notifications, the regulatory authority may issue a statement freeing suppliers from further notification relating to products similar to those already covered by exclusion or exemption. Different strategies need to be adopted for novel products, new models of existing products, products already on the market before regulations are established and those already in the public domain but no longer marketed.

The Guide specifies requirements for manufacturers and importers of consumer products and for quality testing and labelling of products and packaging. It also covers responsibility for surveillance, when necessary, of the supply and use of consumer products which have been exempted or authorized. The Guide recommends that, given the levels of activity usually encountered in consumer products, no restrictions should normally be placed on their disposal.

CONCLUSION

The draft Guide sets out what is regarded as being required for the regulation of consumer products in a manner consistent with the BSS. It also includes annexes providing information on some existing consumer products, on prototype tests for one of the most ubiquitous products - the smoke detector, and on the types of analysis employed in assessment of radiation risk arising from the use of consumer products. It is hoped that the Guide will be published before the end of 1996.

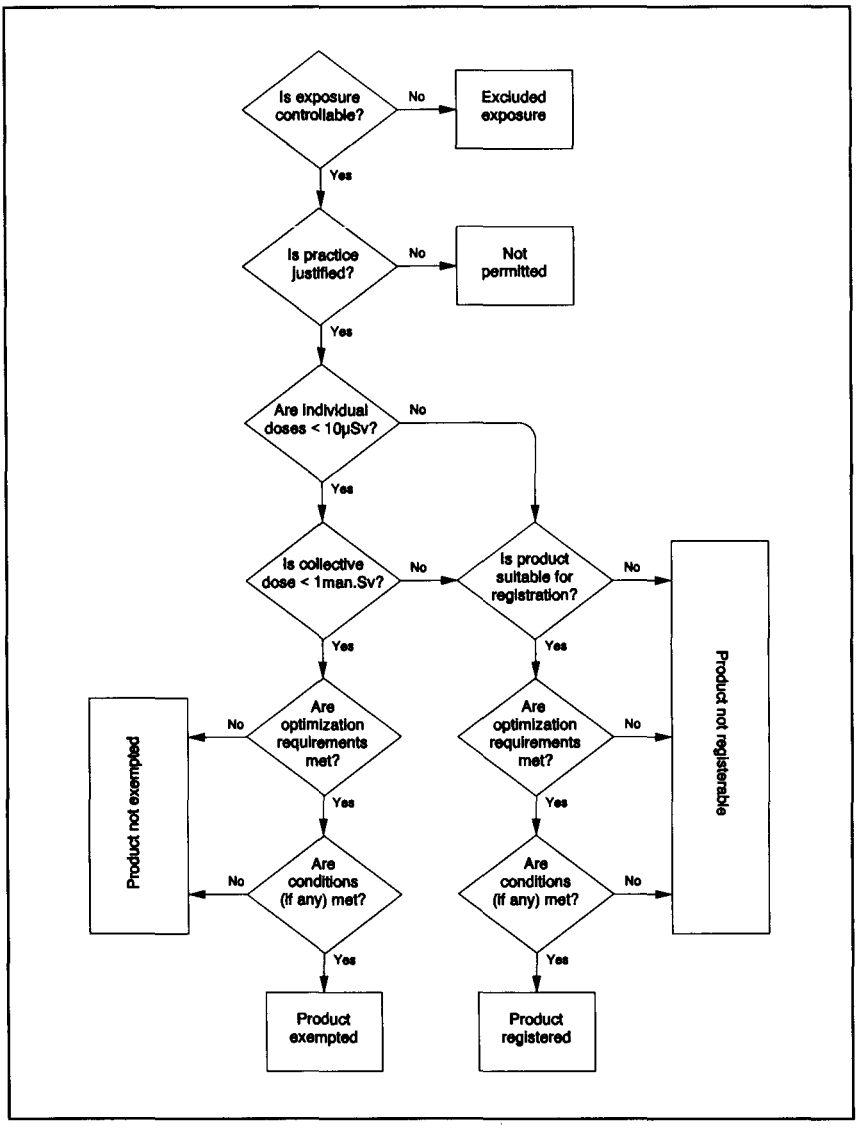

Fig.1 Exemption and registration of consumer products containing radioactive substances.

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RECOMMENDATIONS OF THE AIRP - SEPR - SFPR WORKING GROUP INDIVIDUAL RADIOLOGICAL MONITORING DOCUMENT - DOSIMETRIC DATA BASE

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1 - INTRODUCTION

On the initiative of the chairmen of the three Radiation Protection Societies (Italian, French and Spanish), it has been decided to extend the study carried out at TAORMINA (Sicily, Italy) joint conference on the harmonization of radiation protection practices.

In order to formulate proposals on this topic, a Working Group was created at the end of 1993 made up of representatives of each Society, with the objective of harmonizing the individual radiological monitoring documents and methods of recording and managing dosimetric information in data banks.

2 - THE PRESENT SITUATION

The present situation of the three Countries, in relation to data banks and individual radiological monitoring documents, is as follows:

•ITALY

•a lot of dosimetric services (80),
•dosimetric data validated by a qualified expert,
•some examples of individual radiological documents (i.e. INFN, ENEA),
•an example of national dosimetric data bank.

•SPAIN

•individual radiological document edited by the Regulatory Body for all the exposed workers,
•shared responsability between the contractors and the utilities,
•regulatory dosimetric data bank.

•FRANCE

•individual radiological document limited to "EDF" nuclear power stations,
•medical monitoring card (regulatory),
•re"operational" data bank (DOSINAT-DOSIMO),
•regulatory data bank.

3 - FOCUSING ON THE PROBLEMS

The analysis of the existing situations has showed the evidence of some problems:

3.1 - Individual radiological document

•some residual differences in rules,
•difficulty of changing documents in use,
•assigned responsabilities,
•lack of operational dosimetry at some plants.

3.2 - Data bank

- lack of guidance on data banks in the Directive,
- different software and hardware,
- problems in worker identification,
- different codification in use,
- different radiation protection quantities in use.

4 - WORKING GROUP PROPOSALS

The proposals of the Working Group, on the basis of the criteria established in the European Directive 90/641 "On the operational protection of outside workers exposed to the risk of ionizing radiation during their activity in controlled areas", have been the following:

4.1 - Individual radiological document

- Each country shall have an its own national individual radiological document: the existing ones will be recognized by other countries,
- the definition of the needed information for a communitary worker to be accepted (see the document "Recommendations of the AIRP-SEPR-SFRP Working Group").

4.2 - Data Bank

- The definition of the needed information for a communitary worker to be accepted in a communitary installation has been proposed (see the document "Recommendations of the AIRP-SEPR-SFRP Working Group),
- a common data codification on the basis of communitary criteria will be established,
- a standard data transmission methodology will be defined: periodicity, exchange format, type of transmission, information retrieval and access, ecc.

5 - CONCLUSIONS

The authors of this presentation have presented the document on the proposals to the DG-X1 of the European Commission and to some Commununitary Countries. Moreover they whish for the participation of other radiaction protection Societies in order to continue on working on the harmonization in the context of the European Union.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

Eingang-TAK
02. Aug. 1995

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Abstract No. 90636
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PAPER TITLE THE IMPLEMENTATION OF IAEA BASIC SAFETY STANDARDS AND 1991 ICRP RECOMMENDATIONS IN THE NEW YUGOSLAV RADIATION PROTECTION ACT

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MAJOR SCIENTIFIC TOPIC NUMBER 8.7 (see page 7)

ABSTRACT (See instructions overleaf)

The new scientific evidences on dose effect relationship and experiences and conclusions gained after Chernobyl accident have caused international scientific organizations to reconsider existing standards and codes and if necessary prepare the new ones. The Main Commission of ICRP finalised new recommendations at the and of 1990. Recommendations have appeared in the Annals of the ICRP in 1991 as Publication 60. Besides that, IAEA together with other sponsoring organizations have been working on preparation of new Basic Safety Standards for protection against ionizing radiation. Our expert teams actively have participated on reviewing and drafting the standards.

At the end of 1994. an expert group started a preparation of new radiation protection act in FR Yugoslavia. Working on this task, it was decided to prepare the whole legal system in radiation protection according to international recommendations.

Characteristics and experiences in implementing international standards in national specific legal system are presented in this paper. Subjects that should be regulated more precisely are suggested as well in this paper.

IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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Abstract No.	0685 Eingang - TAK
Receipt	09. Aug. 1995
Author	0052
Acceptance	
Mini-Presentation	

PAPER TITLE REGULATORY ACTIVITIES IN LATIN AMERICA

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ABSTRACT (See instructions overleaf)

Many years ago the countries of Latin America has started differents programs of nuclear development according to their own realities, but at same time they have made great efforts to strenghten the radiological protection in order to regulate properly the different applications of ionizing radiation sources.

In Latin America it notices that there is an increasing interest to work in this field which is assumed with responsability by different competent authorities. The control of ionizing radiation fall mainly on entities of nuclear energy but the State Health Offices, in the most of cases, has the responsability of controlling the X rays equipment of medical use.

The regulatory authorities are recognized by law in spite that the legal rules are not enough to achieve an adequate control in many countries. However, the most of them rely on a implemented system of register and installations license.

This paper shows the more important aspects concerning to the regulatory activities in Latin America.

PRINCIPLES FOR THE REGULATION OF RADIOACTIVE WASTE DISPOSAL FACILITIES IN THE UNITED KINGDOM

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INTRODUCTION

In the United Kingdom, the Radioactive Substances Act 1993 (RSA 93) provides the framework for controlling the creation and disposal of radioactive wastes so as to protect the public from hazards which may arise from their disposal to the environment. The Act imposes requirements for registration of the use of radioactive materials and for authorisation of accumulation or disposal of radioactive wastes. The responsibilities for determining applications for authorisation are different in each of the four countries of the United Kingdom.

Her Majesty's Inspectorate of Pollution (HMIP) has responsibilities in respect of England and Wales and has in preparation, a document setting out the principles and requirements against which any application for a specialised land disposal facility for radioactive waste will be assessed. This paper sets out the background to the document (referred to here as 'the principles document') and its main features and requirements.

On 1 April 1996, HMIP will become part of a new Environment Agency.

BACKGROUND AND SCOPE

The principles document was first issued for public consultation in 1994, in parallel with a consultation document issued by the UK Department of the Environment, setting out the preliminary conclusions of a review of radioactive waste management policy. The policy review has been completed and a White Paper (Cm 2919) was published in 1995 as a firm statement of Government policy (1). The policy is based on the same basic principles as apply more generally to environment policy in the UK and, in particular, on that of sustainable development. The White Paper gives a widely quoted definition of this concept as "development that meets the needs of the present without compromising the ability of future generations to meet their own needs". This requires that:

- decisions should be based on the best possible scientific information and analysis of risks;
- where there is uncertainty and potentially serious risks exist, precautionary action may be necessary;
- ecological impacts must be considered, particularly where resources are non-renewable or effects may be irreversible; and
- cost implications should be brought home directly to the people responsible - the polluter pays principle.

More specifically, and consistent with the above, the Government states in Cm 2919 that radioactive wastes should be managed and disposed of in ways which protect the public, workforce and the environment.

Following publication of the White Paper, the principles document was revised and issued in late 1995 for further consultation. The radiation protection principles and criteria are designed to ensure that there is no unacceptable risk associated with radioactive waste management. In defining these principles and criteria and in their application by the regulators, it is recognised that a point is reached where additional costs of further reductions in risk exceed the benefits arising from the improvements in safety achieved, and that the level of safety and the resources required to achieve it should be consistent with those accepted in other spheres of human activity.

The revised principles document describes the regulatory framework in the UK and the procedures for authorisation of radioactive waste disposal. There is a key chapter setting out four fundamental principles and two chapters containing more detailed requirements that will be taken into account by the regulators when assessing compliance with the fundamental principles. Another issue considered by the regulators to be central to their ability to determine an application for authorisation of a disposal facility is the timely supply of information by the developer. A chapter of the document provides guidance on the scope of the information HMIP considers it will need. The document emphasises the need for a range of technical assessments, that would include but not be confined to a risk assessment, to inform the judgements that the regulators will need to make.

PRINCIPLES AND REQUIREMENTS

Within the approach outlined above, HMIP has the duty to ensure that all disposals of radioactive waste to facilities on land are undertaken so as to safeguard the interests of present and future generations and the wider environment, in a manner that commands public confidence and takes due account of costs. For this purpose, four fundamental principles are defined, taking account of the principles of radioactive waste management set out in the IAEA Safety Fundamentals document issued under the RADWASS programme (2). A prospective developer will need to show that the proposals conform with the four fundamental principles.

Principle No. 1 - Independence of safety from controls

Following the disposal of radioactive waste, the closure of the disposal facility and the withdrawal of controls, the continued isolation of the waste from the accessible environment shall not depend on actions by future generations to maintain the integrity of the disposal system.

The objective of disposal of radioactive waste is to provide sufficient isolation of the waste from the accessible environment so as to ensure the long-term radiological protection of humans and of the environment. However, HMIP recognises that there are mechanisms and events that, at some future time or at some low probability, can result in some release of radionuclides into the environment. The practical implication of this principle is that authorisations for disposal will not be given until it is shown that the overall degree of containment provided by the disposal system after closure is such that the continued safety of future generations does not depend on monitoring, surveillance, preventative or remedial actions after control over the facility is withdrawn.

Principle No. 2 - Effects in the future

Radioactive wastes shall be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today.

Underground disposal facilities can provide a period of isolation which may be some thousands of years for near-surface disposal or some tens of thousands of years or even longer for deep disposal. The degree of isolation required, and therefore the type of facility chosen, depends on the concentrations and radioactive half-lives of the radionuclides present in the wastes, among other factors.. Eventually, some release of radioactivity to the accessible environment may occur, leading to a radiological risk to any human communities in the vicinity of the site. In specifying radiological requirements, the time over which assessments are to be undertaken is not defined since the view is taken that this is a matter for the developer to justify.

Principle No. 3 - Optimisation

The radiological detriment to members of the public that may result from the disposal of radioactive waste shall be as low as reasonably achievable, economic and social factors being taken into account.

In UK terms, this will involve demonstration that the best practicable means are being employed to restrict the radiological detriment to members of the public both before and after withdrawal of control over the facility. It will need to be shown that, among other things, the safety case has a sound scientific and technical basis and that good engineering principles are being applied in facility design, construction, operation and closure.

Principle No. 4 - Radiological protection standards

The assessed radiological impact of the disposal facility before withdrawal of control over the facility shall be consistent with source-related and site-related dose constraints and, after withdrawal of control, with a risk target.

The radiological protection standard for members of the public for the period up to withdrawal of control over the facility is expressed in terms of source-related and site-related dose constraints. During this period, the releases will be monitored and control measures applied to ensure that protection standards are achieved. The Government has accepted that the value of the source-related constraint should not exceed 0.3 mSv/y.

For the period after withdrawal of control, the safety case depends entirely on current assessments of the future performance of the disposal system. Because of the uncertainties inherent in such assessments, they are usually based on probabilistic techniques and the protection standard is more appropriately expressed in terms of radiological risk rather than dose. However, in this case, the standard is expressed as a target on risk to the individual, rather than a limit or constraint, in recognition of the more limited level of assurance of conformity that can be achieved. The risk target is set at 10^{-6} per year.

The release of radioactivity and its impact are dictated by the contents of the facility, its engineering design, the characteristics of the waste and the nature of the site. A number of technical requirements are included which are intended to provide guidance on the aspects that the regulators consider important to the safety case.

SUPPLY OF INFORMATION

Under UK legislation, the developer of a disposal facility is under no obligation to supply information to the regulators until an application for authorisation is made. However, it is envisaged that for a major disposal facility, a developer may choose to make early application and this will allow a programme and structure for the supply of information to be agreed. This aspect is considered to be fundamental to the process of authorisation of a specialised disposal facility for radioactive waste and a chapter of the document sets out the regulators' requirements. This covers a wide range of information related to the site, the facility and the waste, as well as results and supporting information from modelling studies and safety assessments.

The treatment of uncertainty is central to the establishment of the post-closure safety case for a radioactive waste disposal system and a requirement is placed on the developer to demonstrate that the safety case takes adequate account of all relevant uncertainties. This will entail:

- definition of the scope of the assessment;
- systematic identification of all relevant sources of uncertainty;
- quantification of significant uncertainties, where practicable;
- implementation of measures to reduce uncertainty; and
- maintenance of a detailed audit trail.

ADDITIONAL FACTORS

The UK takes the view that consideration of risk analysis alone cannot determine whether a disposal facility is safe. Whilst such calculations can inform a judgement about the safety of a facility, other technical factors, including some of a more qualitative nature, will also need to be considered in arriving at the decision. Sufficient assurance of safety over the very long timescales that need to be considered is likely to be achieved only through multiple and complementary lines of reasoning. Examples of performance indicators other than risk that might be used in technical assessments are given in the report of an IAEA Working Group(3) and include radiation dose, radionuclide flux, time, environmental concentration and radiotoxicity.

Each technical assessment such as a risk assessment is to be seen as a one of the inputs informing the judgements that the regulators must make. To support the judgements, the regulators may need to undertake their own performance assessments, based on information supplied by the developer. HMIP has developed a capability for such assessments, to ensure that it understands fully the issues to be addressed in any such assessment, and to ensure that it can perform such an analysis should the need arise.

CONCLUSIONS

A document is in preparation setting out guidance on the principles and requirements against which the environmental regulators will assess any application for authorisation of a specialised disposal facility on land for low- and/or intermediate-level solid radioactive waste. A draft of the document has been issued for public consultation and formal publication is expected in the near future.

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Abstract No.
Receipt
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Mini-Presentation

PAPER TITLE Bringing UK national legislation into line with ICRP 60 - the process

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ABSTRACT (See instructions overleaf)

ICRP recommendations form the basis for statutory radiological protection provisions worldwide, but the processes by which they acquire a statutory form can be both complex and various. For a Member State of the European Union, adoption of revised ICRP recommendations should first await their incorporation in a revised 'basic safety standards' Directive adopted under the Euratom Treaty. Once that Directive has been adopted, then the varied procedures within Member States will come into play as the Directive provisions are transposed into national legislation.

Adoption of a Directive is not of itself a quick or simple procedure and must take account of national customs and practice in the various Member States, which can vary quite widely. Even at national level, the opinions and wishes of employers, worker representatives, competent authorities and professional bodies seldom coincide exactly, even within their own category (for example, what is acceptable to a large employer may be seen as totally unreasonable by a small one). In the UK, the Health and Safety Commission and, on their behalf, the Health and Safety Executive consult widely before making worker protection regulations or publishing guidance. This slows down the process but does result in provisions that are generally accepted as workable by those affected. UK provisions include an approval system for dosimetry services, with agreed criteria to ensure consistency of standards and procedures.

THE IAEA RADIOACTIVE WASTE SAFETY STANDARDS

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INTRODUCTION

Radioactive waste is generated from the production of nuclear energy and the wide ranging application of radionuclides, for example in industry, research and medicine. The safe management of these wastes is important for the protection of human health and the environment.

The Radioactive Waste Safety Standards (RADWASS) programme is a contribution by the IAEA to the safe management of radioactive waste. Within the programme a set of internationally agreed documents is being developed to assist Member States in complementing or establishing a national framework for the safe management of radioactive waste.

INITIATION OF THE RADWASS PROGRAMME

More than thirty publications in the IAEA Safety Series had been issued on various aspects of radioactive waste management by the mid 1980's but the coverage of the subject areas was not consistent or comprehensive. Accordingly it was decided to establish an ordered structure to ensure a systematic approach to the subject. This was carried out through a review and reorganization of the IAEA publications in the waste management area and at the end led to the creation of the Radioactive Waste Safety Standards (RADWASS) programme.

The initial concept of the RADWASS programme was developed in 1988. The structure, content and scope of the programme was elaborated by international experts in 1990 and work on the programme started in 1991. The development of the programme included submissions to the IAEA Board of Governors at various stages, endorsement by the International Radioactive Waste Management Advisory Committee (INWAC) and in-house approval by the Director General.

The programme included a total of twenty four documents consisting of one Safety Fundamentals document, six Safety Standards (one in each of the six subject areas: Planning, Predisposal, Near Surface Disposal, Geological Disposal, U/Th Mining and Milling Waste, and Decommissioning) and seventeen Safety Guides. The programme was planned to be carried out in two phases, a Phase I from 1991 - 1994 and a Phase II in the post 1994 period.

PROGRESS OF THE RADWASS PROGRAMME

When the RADWASS programme was initiated it was already envisaged that a formal review of the programme would be undertaken in 1993 to define publication production rates and resources needed for the post 1994 period. INWAC carried out this planned review in March 1993 which resulted in a expansion of the programme from twenty four to fifty five documents. This expansion included the addition of eleven Safety Guides, e.g. on licensing, quality assurance, discharge limits, safety assessments and definitions, and twenty Safety Practices to the programme. Furthermore, "Environmental restoration" was added to the subject area "Decommissioning". The status of the programme at that time and the key results were presented at the SPECTRUM '94 in Atlanta [1].

Recently, the two highest ranking documents, the Safety Fundamentals "The Principles of Radioactive Waste Management" [2] and the Safety Standard on "Establishing a National System for Radioactive Waste Management System" [3], have been published in the IAEA Safety Series. That means that six out of twelve Phase I documents are now available. Three Safety Guides are finished and on their way to publication and fourteen documents are under preparation.

NEW DEVELOPMENTS

At the March 1995 session of the Board of Governors, the Director General of the IAEA announced a major reorganization of the Department for Nuclear Energy and Safety. He announced that the existing Department would be split into two new Departments for Nuclear and Radiation Safety and for Nuclear Energy in order to separate safety related and promotional activities of the IAEA. This reorganization became effective from 1st January 1996.

The new Safety Department will cover all radiation and nuclear safety activities performed by the IAEA including activities on the safety of radioactive waste management. At the same time new procedures for the preparation of all safety related documents are being established with the aim of further harmonization of their content, structure and format. For this purpose new advisory bodies are being established to assist the IAEA in the review of all safety related programmes and the preparation and review of the respective documents.

All these activities coincided with a review of the RADWASS programme that had already been planned for 1995. As a consequence, the 1995 review of the RADWASS programme is proving to be more intense and deeper than expected in 1993. This paper will focus on the results of the ongoing discussion, the decisions to be expected in the first half of 1996 and the effects of these activities on the course of the RADWASS programme. At present it seems to be most likely that the programme will be further expanded in the subject areas "Environmental Restoration" and "Discharge" and that the scope of the other subject areas will remain the same but the number of documents will be decreased.

NEW ADVISORY COMMITTEES

In the past, different processes have been applied for the preparation and review of IAEA's Safety Series publications. In order to achieve a better compatibility between the Safety Series publications a new, more uniform, preparation and review process is being introduced. The following set of advisory bodies with harmonized terms of reference is being created:

- the Advisory Commission for Safety Standards (ACSS),
- an Advisory Committee in each of the areas of Nuclear Safety (NUSSAC), Radiation Safety (RASSAC), Transport Safety (TRANSSAC) and Waste Safety (WASSAC).

The ACSS is a standing body of senior government officials holding national responsibilities for establishing standards and other regulatory documents relevant to nuclear, radiation, waste and transport safety. It provides advice to the Director General on the overall safety-standards-related programme and will, for example, resolve outstanding issues referred to it by any of the above mentioned advisory committees.

The advisory committees are standing bodies of recognized senior experts in the respective subject area. They provide advice to the Secretariat on the overall safety programme in the respective subject area.

The new Waste Safety Standards Advisory Committee (WASSAC) provides advice to the Secretariat on the overall radioactive waste safety programme and has the primary role in the development and revision of the Agency's radioactive waste safety standards. It will provide advice and guidance on the RADWASS programme which, in the past, has been the task of INWAC or extended INWAC (INWAC was extended by a regulator if the national representative was an operator). WASSAC, for example, will have to agree on the texts of Standards, Guides and Practices and will make recommendations to the ACSS in accordance with the IAEA's safety standards preparation and review process.

REVISED RADWASS PROGRAMME

The RADWASS programme, Safety Practices excluded, has been reviewed within the IAEA. In particular, this review was done in close co-operation with establishing the RASS (Radiation Safety Standards) programme in order to avoid any unnecessary overlaps or duplications between the two programmes. The resulting draft of the revised programme will be subject to the review by WASSAC in its February 1996 meeting. The following major changes of the RADWASS Programme will be proposed to WASSAC for approval:

1. The programme is organized below the Safety Fundamentals in the three areas "Infrastructure and General Aspects", "Waste Management (Practices)" and "Restoration (Interventions)".

2. "Infrastructure and General Aspects" is replacing the previous subject area "Planning" with only a few modifications.

3. "Waste Management (Practices)" will be subdivided into the three subject areas "Predisposal", "Discharges into the Environment" and "Disposal". The relevant parts of "Decommissioning" and "U/Th mining and milling waste" will be included into these subject areas.

4. "Discharges" are now organized as a new subject area, whereas it was covered previously by a Safety Guide only. Those aspects that are closely related to radiation protection have been moved from RASS into RADWASS.

5. "Restoration (Intervention)" is expanded in the programme because it is organized as a new area whereas it was previously in a common subject area with "Decommissioning". It is covering the subject areas "Environmental Restoration" and "Facility Restoration".

6. Numerous documents on the level of Safety Standards and Safety Guides were combined. For example, the Safety Standard on "Disposal" will include near surface disposal, geological disposal and disposal of U/Th mining and milling waste.

The proposals for a revised RADWASS programme may be changed during the review by WASSAC in February 1995.

CONCLUSIONS

The overall goal of the RADWASS programme still stands, however the in-depth review being carried out with the framework of a more unified approach to Safety Standards is likely to result in a substantial "mid-course" correction.

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abstract.wpw/EW/GW/js(22677)
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UPDATING RADIATION PROTECTION REGULATIONS IN EGYPT

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INTRODUCTION

The Aim of this treatise is to present the rational steps taken in the process of updating the Radiation Protecution Regulations in Egypt. The contents of the review will include a historical synopsis, and the current state of art regarding competent authorities. Furthermore, the various committees formed with responsibilities for specific issues are indicated, including the role of the Ministry of Health (MOH), and that of the Atomic Energy Authority (AEA). Finally, the efforts made towards updating the radiation Protection Regulations in Egypt are highlighted.

HISTORICAL

The Radiation Protection Activities in Egypt were and still remain among the duties and responsibilities of the Atomic Energy Authority according to Act 288, (1). The regulations controlling the use of Ionizing Radiation and Protection Against their Hazards were specified in Governmental Decree No. 59, (2); which mandated that both Ministry of Health (MOH), and the Atomic Energy Authority (AEA) were the competent authorities, with specific jurisdiction for each. All Radation Protection matters related to Sealed Sources, X-Ray Machines, and Accelerators, are the responsibility of the ministry of Public Health; and matters related to Open Sources, and Nuclear and Radiological Accidents, are the responsibility of the Atomic Energy Authority.

During the mid seventies, and the early eighties, the need for updating the Radiation Protection Regulations in Egypt appeared an urgent issue to both competent authorities. The reasons behind this urgency were the exparding application of radiation sources in medicine, industry, and other fields; the increase in number of occupational radiation workers; and the drafting of a new Act in 1983 pertaining to the Nuclear Control and Safety of Nuclear Installations (3).

The period following that witnessed increased activity and efforts on the part of scientists from the Atomic Energy Authority, and responsible persons from the Ministry of Public Health. These efforts were mainly directed towards the formulation of a scientific and administrative structural hierarchy to promote Radiation Protection Activities in Egypt.

A concise presentation to this effect was reviewed at the IRPA-7 meeting (4). In 1988, both competent authorities the MOH and AEA sanctioned the recognition and adoption of IAEA, S.S, 90, 1989 "Basic Safety Standards" as the fundamental conceptual criteria for Radiation Protection Regulation in Egypt. This document was also used to help in the redrafting and updating of the Egyptian Radiation Protection Regulation.

THE MINISTRY OF HEALTH [MOH]

The general layout of the current organizational structrure of the Radiation Protection Activities of MOH is shown in the follwing blook diagram.

Ministery of Health	Chairs	Central Committee Radiation Protection (CCRP)
Vice Minister for Medical Practice		Technical Committee for Ionizing Radation (TCIR)
General Director for Radiological Practice and (EBRP)		Executive Bureau for Radiation Protection (EBRP)
Comptent Authority	For	X-Ray Machines, Accelerators, Sealed Sources

The MOH issued the Ministerial order No.265 (5) which stipulated the "Radiation Protection Procedures for Industrial Radiography". This was published in the Egyptian Gazzette in 1991. This ordinance was essentially based on ICRP publication 26, 1977(6), and IAEA, SS,9,1982(7); applying the System of Dose Limitation.

THE ATOMIC ENERGY AUTHORITY [AEA]

The new hierarchical organization of the AEA in Egypt was instituted (8) according to the following

AEA chairman		Board of Directors

- Nuclear Research Center (NRC)
- Radiation Research and Technology Center (NCRRT)
- Hot Laboratories and Waste Mangement Center (HLWMC)
- Nuclear Safety and Radiation Control Center (NCNSRC)

Comptent Authority	for	Reactors, Unsealed Sources

Several Committees were formed at the AEA in order to formulate and stipulate Radiological Protection and Nuclear safety regulation, and authorize their implementation. The following are the name of these Committees:

- The Nuclear Regulatory and Safety Central Committee as the competent Authority for Nuclear Installations . (1995)
- The Committee for studying the Safety Analysis Report for the Second Egyptian Research Reactor .
- The Consultative Committee for detection of Radioactivity in Foodstuff.
- The Lisensing Committee for Unsealed Sources.
- The Committee for Transport of Radioactive Materials.
- The Committee for Medical Response in Nuclear and Radiological Accident, 1993.
- The Radioactive Waste Disposal Regulations - Adopted by the Board of Directors of AEA, (9).

ACTIVITIES OF RADIATION PROTECTION GROUP.

The Radiation Protection Group of the [Egyptian Society of Nuclear Sciences and Applications (ESNSA), is the Affiliate society of IRPA in Egypt. This group although small in number of its members, is very active in promoting Radiation Protection concepts in most places where scientific activities exist. This activity comes under Radiation Protection Culture. Besides that, satellite meetings and discussions are always held with other concerned groups from the Ministry of Health, the Nuclear Materials Authority, the Nuclear power plants Authority, and other scientists from Universities; to consider issues related to updating Radiation Protection Regulations in Egypt. Other activities of this group include the participation in lecturing at training programmes in Radiation Protection. Also authorship of a Radiation Protection Manuscript for teaching purposes.

This Radiation Procection Group is intimately related with the National Network of Radiation Physics, and the activities of both are either incorporate or run in parallel. The most recent of such corporate performance was the two day seminar (11-12 Nov. 1995) organized under the title "Radiation Protection Regulations in Egypt - the need for updating"(10). This seminar was attended by about hundred scientists involved in Radiation Protection and Nuclear Safety, and also experts on the Law, legislative, and judicial matters.

Several important recommendations and issues related to updating of Radiation Protection Regulations in Egypt; were considered. The adoption of ICRP New Recomendations, publication - 60, (1991); and the IAEA S.S. 115-1 International Basic Safety standards for Protection Againts Ionzing Radiation, and the Safety of Radiation sources, 1994 (11); as the basis for Radiation Protection Practices in Egypt, and for the reformulation and updating Radiation Protection Regulations.

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THE IAEA RADIATION SAFETY STANDARDS

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INTRODUCTION

The objective of producing documents on radiation safety in the IAEA Safety Series known as RASS (Radiation Safety Standards) is to develop an internally consistent set of documents that reflects an international consensus on the principles of radiation protection and safety and their application in practice. During the final stages of preparation and approval of the International Basic Safety Standards (BSS) in 1994 (1), a comprehensive review of all the publications in the Safety Series related to radiation safety was initiated. The review was carried out mainly by staff of the Radiation Safety Section with input from Advisory Groups, Technical Committees and Consultants on particular topic areas. The results of the review include, for each existing document, an appraisal of its status, i.e. whether it is still valid or should be revised and updated to comply with the BSS or whether it should be declared obsolete and withdrawn. The most important output from the review is an overall structural plan for the RASS documents which clearly indicates their relationship to the BSS and identifies those areas in which documents do not exist and should, in due course, be developed.

During 1995 the Director General established five advisory bodies to assist the Agency in developing safety standards. The principal body is the Advisory Commission for Safety Standards (ACSS) which has an overview role in providing advice to the Director General on the safety standards programme and in reviewing the work of the four advisory committees. The Radiation Safety Standards Advisory Committee (RASSAC) provides advice to the Secretariat on the radiation safety programme and on the development and revision of the Agency's radiation safety standards, including publication of documents on radiation safety in the Safety Series. The other advisory committees have similar roles in the areas of nuclear safety (NUSSAC), waste safety (WASSAC) and transport safety (TRANSSAC).

The structure plan and proposals for documents described in this paper will be presented to the first meeting of RASSAC in 1996, any modifications can be found in the poster display.

THE STRUCTURE PLAN

All Safety Series documents should fit into an hierarchical structure with Fundamentals explaining the underlying rationale and the basic principles for protection and safety, Standards setting out in a more regulatory style the mandatory provisions, Guides that elaborate for general areas the manner in which the mandatory provisions are to be implemented and giving further explanations, and Practices developing the application of the Standards and Guides in particular areas or concerned with specific techniques.

The procedure that was adopted in structuring the radiation safety documents was to recognize the importance and broad scope of the BSS and to carry through the structure of the BSS, especially the appendices. The resulting structure is shown in Figure 1.

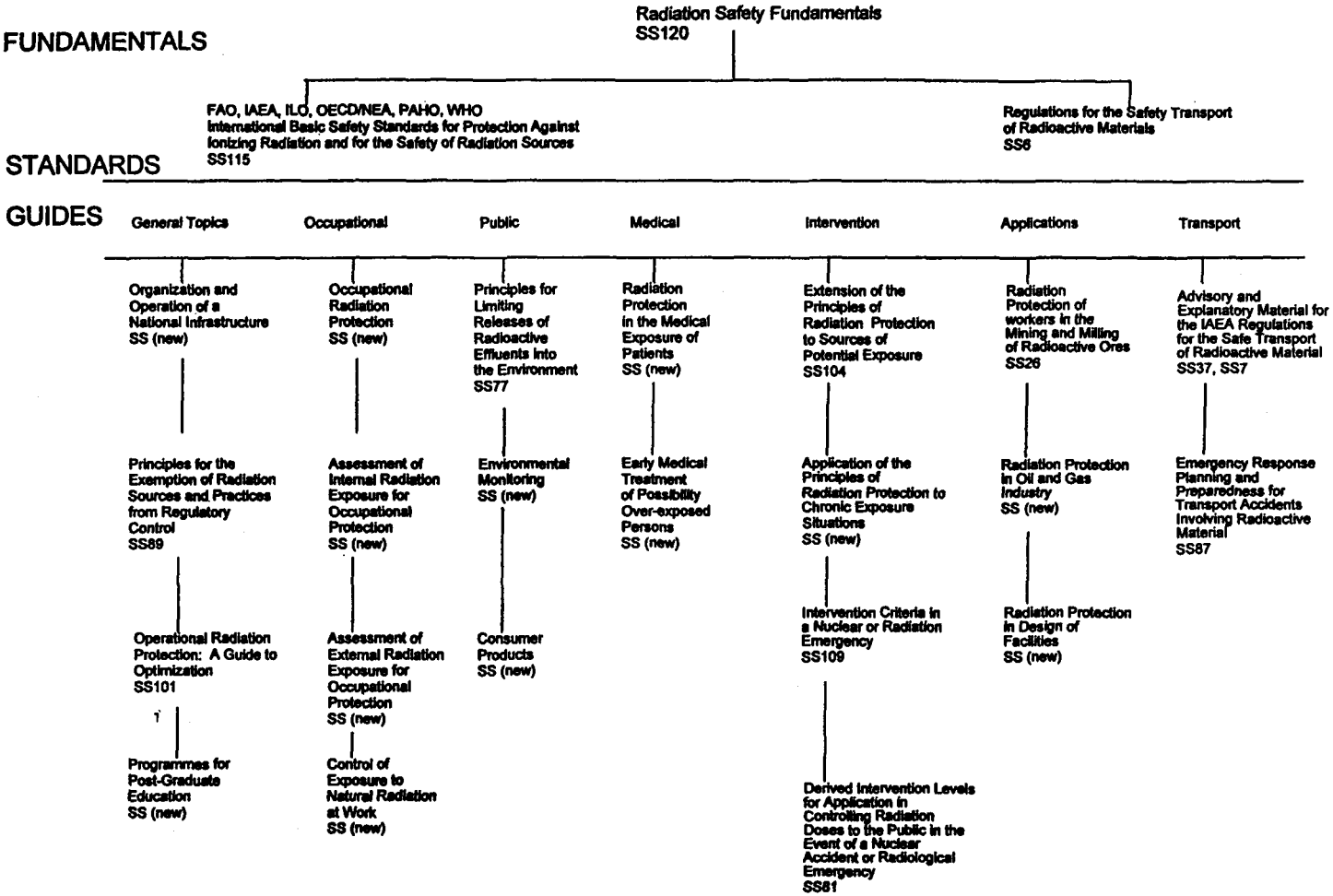

Figure 1. Safety Series Document Structure on Radiation Safety, Showing the Major Safety Guides

FUNDAMENTALS AND STANDARDS

The document at the Safety Fundamentals level concerned with radiation protection and the safety of sources is intended to complete the set of three Fundamentals documents; the other two, which have been published as SS110 and SS111-F, deal with Nuclear Safety and Radioactive Waste Management. The Safety Fundamentals explains the approaches to radiation protection and safety for persons in senior political or regulatory positions and persons who, although not safety specialists, make decisions relating to the uses of radiation in medicine, industry, agriculture and other areas.

The BSS were approved by the Board of Governors in September 1994 following a major effort over several years to achieve a consensus embracing all the Sponsoring Organizations. The BSS have been issued in English as an interim publication; the final publication (in Arabic, Chinese, English, French, Russian and Spanish) will be issued as soon as possible, now that all five other Sponsoring Organizations have given their formal approval and the final tables of dose per unit intake have been received from ICRP. The BSS establish basic requirements for radiation protection and safety, specify obligations and responsibilities and set out the requirements for application to practices and in intervention situations. The transport regulations are reaching the end of the review and revision process and should be published in revised form in 1996.

GUIDES AND PRACTICES

The major Guides are shown in Figure 1 for each of the areas. The Guides on general topics deal with the interpretation or implementation of the BSS and related general matters. A set of three Guides relating to the control of occupational exposures is being developed in a co-ordinated fashion. One Guide will deal with overall implementation of the requirements in the BSS, explaining and advising on how they are to be converted into practical control measures. It will be supplemented by two Guides, one on the assessment of internal and one on external occupational exposures. A Guide dealing with the application of the BSS requirements in limiting releases of radioactive effluents will be developed and supplemented by a revised Safety Practice on generic models and parameters and by a new Safety Practice on default release limits to be used when the situation does not warrant an in-depth analysis. The safety of consumer products containing radioactive materials will be the subject of a Guide which has been in production for some time and can now be finalized in compliance with the BSS. Although controlling the exposure of patients to radiation used for medical purposes is a very important aspect of radiation protection, it has only recently been fully dealt with in the BSS. There is now a need for a new Guide to supplement and expand on the requirements of the BSS. The systematization and extension of the approach to intervention to cover both emergency and chronic circumstances has been one of the major recent developments reflected in the BSS. A new Guide is under development dealing with the difficult practical matter of applying the principles to chronic exposure situations.

In order that users of the Safety Series may have accessible in one place the important aspects of major applications, guidance has been given in such a form. The most important area relates to the mining and milling of radioactive ores, and the relevant Guides and Safety Practices are being revised and updated. A Guide dealing with the problems of the oil and gas industry is planned. An important practical problem relates to the use of minerals containing thorium in industrial operations for which a new Guide is proposed.

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TRANSITION EFFECTS IN THE IMPLEMENTATION OF THE NEW RECOMMENDATIONS- BSS

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ABSTRACT

According to the Constitution, the Romanian nuclear national legislation, which is in a growing process of developing, is tightly connected to the actual international legislative framework.
The provisions and the recommendations lay down by the International Community in the ICRP-60 document and in the International Basic Safety Standards, led to specific approaches that take into account the different levels of development in the nuclear field as well as countries social and economic peculiarities. Thereby, a process of harmonization of the international provisions with the local features is a logic and necessary step toward a rational enforcement of these standards.
Both the principles and the quantitative values (limits and levels) acquired in the ICRP-60 document and in the International Basic Safety Standards after a mutual general consent have a specific impact on practices and interventions related to the existed sources as well as on the new sources, for instance the process of implementation of a multi-purpose irradiator in Romania.

GENERALITIES

The unprecedented international effort that has been started in 1992, oriented on providing standards for protection against the ionizing radiation and the safety of the sources, according to the spirit inspired by the International Commission for Radiological Protection in [1], has been fulfilled in 1994 through an interim IAEA' Safety series report, hereinafter BSS, [2].

Although, these are the first global international standards, with a comprehensive content of the domain, on which the all relevant international organizations have had a mutual consent, the harmonization of the Member State's different positions left the standards' provisions with some weaker points which could create confusion in the process of adoption and enforcement at a national level.

Besides, the efforts of tacking over the new terms are not always a smooth process and finally the funds for the measure implementation are not wholly available. Legislative and practical problems should consider the transition effects in the implementation of the new recommendations in Romania.

LEGISLATIVE FEATURES

Current Status:

The present legislative status is still dominated, unfortunately not for longer time, by:
the old Board of the Ministers' Decision, 1961, followed by regulations in which both nuclear practices have been classified in four distinctive categories and control as well as supervised areas are defined; the nuclear law, 1974, and the law for the quality assurance, 1982. All legislative approaches are obsolete and do not answer properly and internationally accepted at most of the problems. For instance these are not structured distinctly in requirements for the practices and requirements for the interventions, the organizational and institutional changes imply the redefinition of the competent authorities, medical exposure is not among its subjects. Also, our regulations for practices,[3], 1976, have outdated notions, quantities and values. Many basic concepts are no longer valid and even the internal structure of these regulations could no longer stand against BSS's provisions. The old custom of separate and distinct treatment of the radiation protection and safety of the sources missed most of the provisions on potential, emergency and chronic exposure.

For the intervention activities, we have distinct regulations, intensively used in the emergency preparedness plans, which could be considered as being bread- new, 1993, but the intervention levels and the action levels are different; also some terms have to be revised.

In absence of specific regulations, the IAEA's guidelines have been used.

Changes:

A general process of revising of the national legislative framework started several years ago with new proposed nuclear laws concerning the promotion of research and applications as well as the organization and the responsibilities of the Regulatory Authority. These laws are presently under the debates of the Parliament. Their content had to be changed quickly because at the time the BSS document was in work, drafts of the laws were ready. After it was accepted that the new international standards will have a strong impact on these laws, divergent opinions argued to what extend the BSS should be followed.

One opinion stressed out the necessity of inclusion as many as practicable provisions in a harshly translative manner so that the requirements contended do not be already outdated at the very day of the law's enforcement. The other opinion pointed out that only the BSS' structure should be retained and a carefully process of adoption should be focused on the basic and necessary provisions as well as on the definitions, leaving for the regulations the most delicate aspects together with the values. Anyway it is sure that these new laws will treat separately practices and interventions, will include medical exposure and will give an understandable meaning to radiation safety culture.

The new regulations that are expected to be drawn from the national nuclear laws but no effective initiative have been started so far. So, an anachronism situation is going to occur because we are expecting to have law s in compliance to BSS but older regulations in which terms and notions like: source, nuclear installations, irradiating installations, the distinctive meanings of dose and exposure, clearance and exception and the various guidance levels, dose constrains, kerma, radiation weighting factors and tissue weighting factors, risk, safety culture, controlled and surveillance areas, working levels, etc., are not met or have different sense.

Besides, there are others important issues that could not be easily eluded:

- the problem of fitting the BSS's concepts to the Romanian language; based on the experience gained trough our specialists involved in the drafting of BSS, we tried to avoid such aspects by organizing conferences, seminars and debates on the BSS in which we worked on the adoption of the new raised terms to Romanian language. Thus, we found real difficulties in deriving Romanian corespondents for clearance, exceptions- to obtain a different meaning vis a vis exclusions- projected and adverted doses or risks.
- the shortcomings of some data; because of divergent opinions, BSS looses aspects that affect the initial goal of comprehensively, we mention here: guidance levels for chronic exposure situations, others except the radon, values for the operational intervention levels for each radionuclide like the committed effective doses per unit of intake which BSS offers for the practices, values of the risk limits and constrains for the potential exposure.
- ambiguities in the definition of ionizing radiation- an energy threshold is asked by the regulations- and the lack of definitions for the source term, raw nuclear material and environmental restoration; all these could lead the considering of the other documents.

Here, we appreciate that a positive movement is that the Romanian Society for Radiological Protection, [4], is accepted as a counterpart of the civil society for legislative proposals analyzing.

On the other hand BSS should be followed by the guides and practices that have to develop in depth the generally provisions.

An internationally agreed content for license and registration should be recommended to gain a harmonization among various kinds of authorization.

PRACTICAL FEATURES

Current Status:

There are sources that are operated for years according to the older requirements, limits and levels; the decision to change should be carefully weighted, and without doubts, accompanied by cost -benefit analysis. Also there are installations under construction, on- going projects, sources near to be commissioned or decommissioned. Comprehensive assessments should be performed on what are we going to change, how will we change it and who will finally pay for that changes; anyway cost- benefit analysis should be carried out.

Specifically, the new limits for occupational exposure translated into the operational quantities, dose rate, together with quantitative values for risk of potential exposure, [5], are going to lead to re-assessing of the controlled and the supervised areas next to radiation sources.

There is a widely spreader restrained in switching to new terms and quantities. All the existent literature displays an old and slightly different approach of radiation safety. A similar situation has been met at the time of transition from a rontgenological way of approaching the field to an energetical one, particularly the movement from exposure to absorbed dose through intermediate status of rontgen echivalent physically. Even after years the old units are used in practice; for example, even whether ICRP and BSS recognize absorbed dose as the basic idea for radiation protection the calibration is still largely performed in terms of exposure. So, just for tacking over this particular situation it is necessary for training and funds.

Comparative it could be image that lectures, laboratory work as well as handbooks are conditions for a wholly offensive.

5 Changes:

To met BSS requirements for areas zoning a methodology was developed. Although some older sources comply even to the new limits, being considered at its time under more restrictive conditions, there are many of them that need revising of the normal and potential exposure's boundaries. Often the implementation of practicable measures is out of the question because of the lack of the funds at this stage. However, the methodology will be helpfully for the new sources, for example for the commission of a large scale irradiation installation, at the design phase at the time of BSS endorsement. This is an example of possible change with minimal social and financial cost.

By using the values of the new committed effective dose per unit of activity via inhalation or ingestion, instead of the older ALI values, a few source-related assessments and individual-related assessments were carried out for assessing both the impact and the new source constrains. Also through the same data new planning zones for intervention have been obtained, thereof it is a need to revise the emergency plans.

CONCLUSIONS

1. At this stage we are in a "legislative phase" in which we are expecting to have soon recent national nuclear laws in compliance with BSS provisions, limits and levels, and a "practical phase" of assessing the possible changes- new controlled and supervised areas, new environmental impact studies, new planning zones for interventions; for all these facts cost- benefit analyses are imperatively asked.

2. The presence of older regulations, which could not be discharged in the very next years, and de facto status of the sources, related to practices or interventions, are realities in the actual Romanian nuclear framework.

3. The transition will be a continuous process without thresholds; this aspect is conditioned even by the transition of the BSS itself.

4. Training courses and workshops are imperatively to be organized in this period of transition.

5. BSS should be followed by guidance and practices for the basic requirements supporting.

6. A free process of information and data exchange have to be supported through the Member States.

7. The overall social, economic, politic and institutional effect as well as the psychological impact ought to be assessed; generally people do not like changes, furthermore they have an aversion to them.

8. For some aspects BSS misses the declared comprehensively and even symmetry, the lack of: guidance levels for chronic exposure for radionuclides others than Rn, operational quantities for interventions, zoning of the nuclear areas, transport of the radioactive materials.

9. It is necessary to evaluate the degree of correlation between BSS and other international documents and agreements, a specific case is the liability in case of nuclear accident; IRPA could take the initiative on bringing all the internationally distinct actions at a common factor.

10. The long- time projects, decommissioning, waste management, repository for radioactive waste, should be treated carefully considering the actual requirements of the BSS as well the future expected changes that could be promoted by ICRP.

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BECQUEREL CENTURY: GOOD AND BAD IN RADIATION FIELD

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ABSTRACT

The purpose of this poster is to offer a synthetic as well as comprehensive image on this century of radioactivity with its beneficial and non-beneficial aspects. It is both a historical and an allegorical graphical presentation of the man's habitat, which have had in all the times a strong radioactive component. In order to paraphrase, but in a different way, another optimist message of this century perhaps we have to say that the cradle of man is radioactive and all the steps outside it should be very carefully considered. During this last century man started to challenge the nature in its intimate features, i.e. those associated to radioactivity phenomenon.

In the first phase the world was fully confident on the radiation peaceful applications, then in the second phase all these beneficial aspects have been detonated toward military applications which astonished all the people by the harmful effects of radiation, finally we have been passing in a new era dominated, all over the world by a spirit of harmonization and consent by which we are going to reach a globalization of radiation protection and safety principles and measures. Even if it is the latest source of energy which man has stolen from the nature, nuclear energy is now -from a technical point of view- a strong option for the human future and is going to gain more and more in the public acceptance branch. If this poster should have been confined in a phrase then the motto can be: with too much radiation we cannot be and live, but without radiation we cannot survive.

GENERALITIES

Human Being has been obstinately searching to understand what is good and what is bad around him. Along the centuries he has the intuition that the answer is probable the key that can unlock the universe for him. He has been groping for a clue in soil, water, air and fire, and far beyond these, sometimes he got a bit of true for which the dulls were happy and the sagas gloomy.

Man looked for an aid at his fellows being them priests or political rulers, monarchs or celebrities, scientists or prophets, and thought that they can help him to choose right. However, always just a moment separated him from the land where milk and honey are flowing, [1]. Invariantly, that hardly expected moment does not want to show up as it was promised and after that a terrified world raises instead and the man hidden his soul in myths, fairy stores or dreams. However, he learnt that at least he was at the very border between god and bad but missed something that nobody told him.

This old and new story is also the tale of radioactivity. She was a fairy living far from the people's eyes but making all the good that his father has learnt. Without any exceptions she modulated smoothly - with 2.4 mSv yearly- all the things she met on the Terra so that all the things lived peacefully. More than that she gave them a chance to grow. A fine balance ruled on the earth marking that without radiation we cannot live but with too much radiation we cannot survive.

It was the year 1896 when an inquisitive person, namely Henry Becquerel, rise a portion from the invisible veil, that covered the earth from the time when no person stepped on it, and said what he saw. Even that what he said was by far limited, humanity opens again the box in a harshly manner and rummaged till its bottom.

The early pioneers of this new field, among them: Herman Rontgen, M. Sklodowska Curie, Piere and Joliot Curie, were exuberance in announcing that a pure good thing finally shows up, and even they believe that. But later other reliable scientists, among them: L. Szilard, T.R. Oppenhaimer, E. Teller, I. V. Kurchatov, W. Heisenberg, A. Saharov, adulterate the initial aim and find something that is "good" only for the war time, i.e., nuclear bomb,[2]. From that moment man alienate and besmirched the mild face of the fairy of radioactivity.

WITHOUT RADIATION WE CANNOT BE AND LIVE

Man's cradle is without doubts radioactive. In the manner as any thing is neither good nor bad only the quantity can give a sentence, the habitual radioactive environment preserves life and helps the evolution at the same time. Man's features are tightly connected to the natural exposure, compiled by specialists around 2.4 mSv, [2]. This value is suspected to be among other natural constants that have been favoring the life on earth. Although there are both spatial differences- for example man's natural exposure varies with altitude and other local conditions- as well as temporary disrupters -for example at the times of pole inversion earth should had been devastated by deadly cosmic exposure- the value remained million of years roughly invariant.

Of course, the value itself does not tell us the true comprehensively, this depends by our degree of

knowledge and hypotheses made. It is obvious that the value has to be changed by considering other ionizing radiation that are now neglected. We mention here only the recognized ionizing effect of the electro-magnetic radiation, e.g. that part of visible and mainly ultraviolet bands. The intrinsic limitation of our knowledge should not restrict us at a border of a self- imposed frontier; other exotic ionizing radiation with biological effects will be assumed.

The life is strong conditioned by the influence of visible and UV photons in the processes of photosynthesis and vitamin production into our bodies. Human Being supplemented the influence of natural background with man- made exposure, also with beneficial effects: medical radiodiagnosis and radiotherapy, applications of the radioisotopes in industry, agriculture, hydrology and mainly to nuclear power production. Many of our commodities are indebted to man- made exposure.

However, human being in his inquisitive searching toward more safety radiation sources is exceeded by far by the nature. This "designed" a marvelous fusion reactor that is equipped, for the users' sake, with ingenious radiation protection in depth measures: distance, time, shields, magnetic traps, differentiable absorption layers, ozone filter.

WITH TOO MUCH RADIATION WE CANNOT SURVIVE

The loosing of one of the radiation protective means with which the sun- earth system is designed, for instance Terra's magnetic field, could lead to the extinction of the life on our planet. If we believe that the human being is among the most vulnerable creatures, his lethal dose is among 1 and 3 Gy, then we can see how easy the life can lose its witnesses.

Also the rarefaction of ozone layer could have acute biological effects on all the life forms.

Although at this stage, man cannot influence significantly the effectiveness of the natural safety barriers of the sun - earth system and because there are still doubts on an artificial depreciation of the ozone layer, man can significantly contribute brutally by nuclear bombing to the enhancement of the radiation exposure. Thousands of nuclear missiles installed on the ground, or submarines, or ships, or planes are frightening the natural fine balance. The nuclear testing in atmosphere from the 50s, 60s produced a total man-made collective effective dose of 30,000,000 man x Sv, [3].

We pay annually for our social comfort from nuclear power production with 400,000 man x Sv. In addition accidents involving nuclear power plants are sources of harmful effects. Chernobyl is equivalent to 600,000 man x Sv.

Radioisotopes' production and use contribute annually with 80,000 man x Sv, [3].

Moreover there are other significantly sources of exposure: for instance: fertilizer industry, 300,000 man x Sv on each year, nuclear weapons fabrication, 60,000 man x Sv, satellites re- entries in atmosphere, underground nuclear testing 200 man x Sv, [3].

However, finally it is up to us whether or not to preserve the earth and the life it has been developed for million of years.

CONCLUSIONS

1. God said " Do not eat from the Tree of the Good and Bad' Knowledge".
2. The Nature is just as it is.
3. Terra is a little and finite starship.
4. Demography, destruction, comfort, power, restoration. ecological development
5. Will finally win the human's reasoning faculty ?; Dubito ergo cogito....
6. The author propose a subjective separation of good and bad, however these could be reversed.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Legislating for occupational exposure to sources of natural radiation - the UK approach

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MAJOR SCIENTIFIC TOPIC NUMBER 1 (1.1) (see page 7)

ABSTRACT (See instructions overleaf)

The UK Ionising Radiations Regulations 1985, made specifically to implement most of the requirements of the 1980 Euratom BSS Directive, include provisions relating to occupational exposure to sources of natural radiation. The definition of "radioactive substance" includes any substance which contains radionuclides whose activity cannot be disregarded for radiation protection purposes . This means that some low specific activity ores and sands fall within this definition and are therefore subject to relevant requirements of the Regulations.

Radon is covered more explicitly by applying many of the controls where the work involves exposure to radon daughters at a level greater than 6.24×10^{-7} Jm^{-3} (0.03 Working Levels). For above ground workplaces, if a defined measurement protocol is followed, this level approximately equates to 400 Bq m^{-3} and is generally recognised as the workplace action level. The regulatory approach has been to seek remedial building measures so that the workplace is removed from control. Otherwise the full regulatory provisions, such as classification of people and areas, dose monitoring, medical surveillance etc., may well need to be applied.

The UK Health and Safety Executive, through commissioning research and use of existing information on workplaces and homes, has been able to target employers likely to have a radon problem and to raise their awareness of it. HSE is able to offer simple advice about getting workplaces tested and about remedial measures, if this proves necessary.

REGULATORY FUNCTIONS AND MEDICAL EXPOSURE

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INTRODUCTION

Must the patients subject to medical radiation practices be protected ?
What does the protection of patients means ?
Who must protect the patients ?
Is it not a sole responsibility of physicians ?
These are some of the questions this paper attempt to discuss.

MEDICAL EXPOSURE

Last international recommendations on radiation protection assign new emphasis to *medical exposure*, the expression used by the International Commission on Radiation Protection to name the exposure of patients to radiation for diagnostic or therapeutic purposes (1). The International Atomic Energy Agency, the World Health Organization, the Pan American Health Organization and other international organizations (2) have shown similar concern for medical exposure as for ocupational and public exposures. These are signs of a remarkable evolution toward a better protection of the patients.

Medical exposures have some particular characteristics:

a) Patients are *intentionally* exposed to radiation. Benefit from medical practices can not be obtained unless patients receive some dose of radiation (*necessary dose*).

b) Dose limits are not applicable because benefits and risks of radiation has to be evaluated on the same individuals and therefore there is not room for inequity..

c) Optimization has a lower constrain (necessary dose) while optimization of ocupational and public exposures has upper constrains (dose limit or another constrains)

d) Justification and optimization principles apply to medical exposure. They are more closely interrelated than in any other aplication of radiations. To be justified a medical exposure must produce some *benefit* to the patient health. Bennefit depends on the *right prescription* of the practice and the *good quality* of its performance it is to say optimized performance. A medical practice that has not been optimized may have poor quality and provide no benefit to the patient and therefore should not considered to be justified. Prescriptions rely exclusively on *human factors* (the knowledge and experience of physicians). Performance rely on human factors (mainly physicians and physicists) and on *physical factors* (proper equipment, its maintenance and calibration, rigth treatment planning, etc.)

e) Diagnostic practices involve relative small individual doses and very large collective dose. (3) Significant part of collective dose attributable to diagnostic practices in most countries could significantly be reduced without loss of benefit (4) through a better justification and optimization of procedures. Diagnosis is the field of radiation applications where the most substantial reduction of collective dose may be achieved at very low cost.

f) Therapy procedures involve large individual doses and the efficacy of treatments depends on the accuracy and distribution of doses in a very sensitive way. Significant part of therapy procedures may be inefficient due to a poor accuracy in delivering doses to the tumor or failures in protecting normal tissues.

g) Potential exposures asociated to medical sources have to be considered (5). Accidents caused by medical sources are a significant proportion of radiological accidents. Along last fifty years 38 % of deaths in radiological accidents were produced by medical sources (6) and most of them were patiens.undergoing radiation treatments. These only are the recognized accidents. Most of them were caused by significant deviations from prescribed doses. However it is well known that even a small deviation from the prescribed dose (10% for instance depending on the type of tumor) may cause a significant reduction on the cure probability. Unles good quality control procedures are implemented it is quite feasible that nobody realize that a significan fraction of treatments are carried out whith smaller probability of cure than what correspond to a good technique. Where is the boundary between an accident and a bad practice ? Is it a matter of

perception ? Many patients may not have positive reactions to radiation therapy (what may signify death) due to lack of good quality treatments. Should those situations be considered accidents or bad practices ? Radiation therapy is probably the field of radiation applications where more lifes can be saved through a human and physical resources of good quality.

h) There is not public perception of risk from medical sources . Since they are associated with preservation or recovery of health they seem to be inherently good. Medical practice with radiation sources may be carried out with great impunity.

REGULATORY AUTHORITY

Since dose limits are not applicable to medical exposure physicians have unlimited freedom to irradiate patients Regulatory authorities have to decide how deep their regulation on medical exposure should go. Whatever the decision be it will imply some sort of interference with medical practice. .

Regulatory Authorities, when are not the same as Health Authorities may find some conceptual dificulties to aplly regulations on medical exposure.. That means a sort of self constrain to impose requirements because they may collide with medical practice freedom

There can no be restrictions on medical procedures prescriptions Regulations on medical exposure can be oriented to enforce quality improvement through three main tools: by promoting good expertice of the human resources involved, by promoting the use of adequate equipment and techniques and by promoting quality control procedures.

ARGENTINE EXPERIENCE

Argentina started their regularory activities on Radiation Protection in 1958 when the National Atomic Energy Commission (CNEA) iniciated the promotions of radioactive material applications. The first regulations stablished the basic principles of the license process "to attend public utility reasons and to prevent the possible harm of nuclear trasmutation and reactions" (7).

Since that time individual and institutional authorizations were granted by the Regulatory Authority (CNEA) when specific requirements were fulfilled. There are evidences of concern about the protection of patients in that regulation. Physicians had to have specialized education and proper training to be authorized. "Medical facilities had to have adecuate means for a clinic attention of patients." .

Further regulations preceisely defined.the requirements to obtain authorization to utilize radioactive materials or accelerators in medical practices (8). These include specialized courses and training periods of about three years. Those requierements are still in practice and were important steps toward good quality medical exposures. Unquestionably these requierement interfere severily with the profesional freedom of physicians to use ionizing material in medicine. However as this process was accompanied by a true leadership in research and education from the same institution that applied the regulations there was a good degree of acceptation. Another important factor was the fact that every authorization request has been analyzed by an advisory committee integrated by experts in the use of ionizing radiation; most of them physicians or physicists specialized in medical application of radiations. So petitioners have been judged by their peers.

Ap to 1980 institutional authorizations were granted after verification of generic radiation protection and safety measures mainly considering the protection of workers and public There was not special regulation on physical aspects to assure good quality medical exposure.

In 1980 an additional regulation was adopted jointly by the CNEA and the Public Health Secretariat (SSP) exclusively to promote the improvement of medical exposure quality in radiotherapy and nuclear medicine.(9). This regulation was previously discussed with the most representative physicians and physicists in the fields of radiotherapy and nucler medicine.. Their main requirements are:

a) specifications of some characteristics of equipment to avoid excessive obsolescense.
b) specifications on maintenance and quality control of equipment
c) use of complementary equipment for simulation and calibration
d) specifications of minimun set of units to integrate radiation therapy faccilities to cover the most frequently techniques in teletherapy and braquitherapy.
e) minimun set of units to integrate nuclear medicine faccilities to cover the most common diagnostic techniques.
f) composition of minimun staff of physicians, physicists and technicians..
g) need for authorizations renewal every five years.

This regulation was thought as an evolutionary process since periods of time were allowed to fulfill the requirements..

Had the Regulatory Authority of Argentina (CNEA) enough power to enforce such a regulation ? The fundamental regulation (7) gives it sufficient faculties to regulate the conditions for granting individual and institutional authorizations. However, since the 1980 regulations on radiotherapy and nuclear medicine advanced significantly in the field of medical exposure it was though that the support of the Public Health Authorities (SSP) would help to achieve acceptation among physicians. As a consequence of an agreement signed between the Public Health and Regulatory Authorities the regulation was jointly enforced by these two institutions mainly through the regulatory faculties of the Regulatory Authority. As stated in the text the SSP "requires the concurence of CNEA to enforce the regulation." It must be noticed that Regulatory and Public Health Authoriies have kept an excelent level of cooperation in Argentina.

In 1994 Argentina updated its basic standards and the concept of medical exposure was incorporated as well as the optimization of medical exposure through the proper utilization of equipment and techniques (10). In 1995 the Regulatory Authority was separated form CNEA. The new institution is the National Nuclear Regulatory Entity (ENREN) and all the regulatory faculties were transfered to it. However there has been different interpretations on whether the ENREN has the same faculties as CNEA had to enforce the existing regulations on medical exposure. At present this is a matter of analysis. This an example of institutional caution to act in the field of medical exposure even when there has bee a generalized acceptation of requierements imposed by the 1980 regulations.

However a good degree of cooperatrion between the new Regulatority Authority and the Public Health Authority will most probably warranty the continuation of a policy unquestionably positive for the protection of the patients, avoiding an involution on the achieved goals.

CONCLUSION

Medical exposure is the field of radiation protection with the largest possibilities for reducing unnecesary doses and consequently negative effects of radiation without associated benefits. It also has the greatest posibilities of increasing benefitial consecuences of medical radiation sources by improving diagnostic and therapeutic techniques quality. Therefore Regulatory Authorities should not be absent in this field.

But they should not act alone. Regulations on this matter should not be prepared without the intervention of physicians and physicists with recognized expertise on medical aplications of radiations Cooperation of Public Health Authorities is most desirable. Even when the Public Health Authorities may not have operative capacity to enforce regulations in some countries Public Health Organizations have the faculty of judjing what is benefitial or not in the medical practice.. Agreement between Public Health and Regulatory Authorities may result in a sinergistic combination of resources. International Organizations have provide good example of cooperation between this two areas.

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The Russian State System for Quality Assurence of Radiation Measurements

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The State system for ensuring quality of radiation measurements (hereinafter referred to as "System") has been establishing in Russia throughout a long period of time. When arranging the System, any measurements of physical quantities and parameters, which characterize sources and fields of ionizing radiations, were considered as the radiation measurements (RM).

The System has some distinguishing features caused by the factors which are basically characteristic for this field of measurements made in Russia.

The first factor resides in the fact that in Russia, the RM are currently central problems because of a great number of installations where the ionizing radiations exhibit activity. Among them there are radiochemical enterprises, atomic power plants, research reactors, Navy and Civil fleet with atomic engines, radionuclide diagnosis, radiation therapy etc.

The second factor is a great variety of the RM types, stemming from a large number of physical quantities and parameters: specific activity - with numerous radionuclides being measured, the flux and flux density of the charged particles and neutrons - within a wide range of energy, power of the absorbed and equivalent doses - within a wide power range for different types of radiations etc., as well as from a number of problems, the solution of which require such measurements as: technological (at the enterprises and plants), diagnostic (in medicine and manufacture), radioecological (at monitoring laboratories) and so on.

The third factor consists in the necessity to conduct quite often the RM for economic subjects such as atomic power plants or ecology organizations which have the same interests and should come to some responsible conclusions related to the fate both of manufacture and of population.

To satisfy the above requirements, the System should be arranged in such a manner that it could be used as an objective basis for solution of any problems including the disputable ones which refer to the RM.

Thus, from the very beginning, the System has been created with the aim to provide the metrological assurance of all economy branches of Russia in the field of the RM, using as a basis the centralized principle which means:

- reproduction of measurement units which are the main ones for a given field of measurements with the help of state standards;
- the legalized hierarchy chart for calibration of measuring means with an approved list of secondary and working standards and definite intercalibration intervals;
- availability of specifications regulating methods and accuracy limits of measurements for the whole park of the working measuring means being in usage.

At present the System is based on 11 state primary and special standards which provide to reproduce the main measurement units for the activity of radionuclides in sources, gases, aerosols, flux and flux density of the charged particles and neutrons, power of the absorbed dose of photon, neutron and beta-radiation as well as some other measurement units.

The size of the units reproduced by the standards is transferred to the working

measuring means according to the hierarchy calibration charts including more than 50 working standards operating in the form of suitably designed dosimetry and radiometry set-ups installed at the leading enterprises or at the main metrological laboratories. As the certified radiation sources, they use radionuclide sources, x-ray apparatus, reactor channels, output accelerator beams. All measuring means including working standards, are tested every 3-5 years, the same interval being used for metrological certification of the measurement methods.

The System described, which to some extent appears to be bulky, has demonstrated its efficacy in carrying out the works on liquidation of consequences of the accident at the Chernobyl Atomic Power Plant.

First of all, it was found to be very important that all radiometers and dosimeters being used at the accident site, had been calibrated in the established units and that the measurements were performed according to the specified procedures. This enabled to assess the accident scale correctly, to bring out the major seats of origin of the damage and to forecast the development of radiation conditions. Thus, it was possible to organize and monitor the radiation dose of individuals promptly and on a large scale.

Coincidentally with the liquidation of the direct accident consequences, a demand arose for a radiation inspection of population, food, soil, water. This problem was promptly solved at a high level of reliability with the help of the standard solutions of radionuclides (SSR), which were used for manufacturing the standard specified measures of activity (SSMA) serving as full-scale models (simulators) of the test samples taken at the contaminated locality. Certification of the SSR was carried out with the help of the standard set-ups being a part of the state standard for the radionuclide activity. The SSMA were used to calibrate scales of the radiometers and spectrometers, as well as whole body counters at the laboratories of sanitary-epidemiology services including those situated in the regions of the radiation contamination. At the same time, to certify the radiometer set-ups and to measure the radionuclide activity in the samples, some methods and corresponding procedures were developed. Thanks to the System, they were brought up to their direct executors.

Thus, owing to the existence of the System, it was possible to solve the problem of carrying out reliable radiation measurements under the conditions of liquidation of the large-scale accident consequences. This in itself is a good evidence that the choice of a centralized system for metrological assurance within this field of measurements in Russia has been chosen correctly.

Once the System has been created, it is being continuously developed and improved. One of the most urgent problems which it faces now, is the creation of a network of laboratories for monitoring the radiation. These laboratories will be accredited for the right to carry out the radiation measurements according to the rules regulated by the ISO/IEC standards for the testing laboratories included in the system of radionuclide product certification. This will permit to harmonize the structure, rates and rules accepted for the System functioning in Russia with due regard to the corresponding elements of the testing laboratories of the other countries where the international standards are being used.

FROM REGULATIONS TOWARDS RADIATION PROTECTION CULTURE

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INTRODUCTION

Compliance with the technical standards and specifications is a necessary but not sufficient condition for quality in radiation protection. Reaching this quality objective is not a matter of forcing improvements by a regulatory policy of reducing dose limits, but of promoting a real radiation protection culture. The spread of such a radiological protection culture encourages the deliberate adoption in everyday practice of behaviour likely to reduce exposure to ionising radiation as low as reasonably achievable.

The aim of this paper is to demonstrate that the need to diffuse a radiological protection culture is inspired by the philosophy behind the system recommended by ICPR Publication 60 (1) on the management of residual radiological risk and, in particular by the behavioural and incentive approach implied by the optimisation principle. Special attention will be given to the fundamentals likely to contribute in a definition of radiation protection culture.

THE PHILOSOPHY BEHIND THE RESIDUAL RADIOLOGICAL RISK MANAGEMENT SYSTEM RECOMMENDED BY ICPR PUBLICATION 60

The changes to the place and functions of the concepts of dose limits and optimisation of radiological protection as set out in ICPR Publication 60 are contributing to the promotion of a dynamic action based on a behavioural approach, and this should no longer be based simply on the strict regulatory application of individual dose limits.

The system of prevention based on the threshold concept has given way to a system of radiological risk management system based on the "prudent avoidance" principle associated with the recognition in the 1950s of stochastic effects and the later adoption of the hypothesis of a linear relationship with no "safe" threshold in terms of low doses.

The adoption of this hypothesis, which is not the result of a scientific knowledge but of an intellectual approach aimed at encouraging action in radiological protection, has had the following consequences: limits are now considered to be the lowest boundaries of unacceptable doses. Values above this limit should be very strictly controlled, and doses under the limit should only considered to be acceptable in so far as residual exposure levels are optimised.

According to this approach, we should implement a management approach to radiological protection in terms of individual and collective doses, based on the establishment of optimised dosimetry objectives, in addition to controlled radiological protection based on compliance with individual dose limits. The implementation of the principle of optimisation in no way does away with the need for conventional radiological protection monitoring, which it should complement.

Quite simply, compliance with dose limits no longer ensures high quality protection. As ICPR Publication 60 recalls, dose limits can no longer be seen as the boundary between that which is safe and that which is dangerous nor as the simplest and most effective way of maintaining exposure at sufficiently low levels and of encouraging improvements.

It is the principle of optimisation which initiates action in the area of residual radiological risk management and which motivates operators to try to reduce doses within an optimal allocation of resources. The approach recommended by the principle of optimisation is to encourage initiative and innovation rather than "imposing" a limit to be complied with. Thanks to the specific nature of the principle of optimisation, we are able to go further than a merely legal perception of standards represented by given quantitative data. The principle of optimisation can be compared to a behavioural requirement which encourages operators to on-going efforts in achieving a given result. The operator may therefore be held responsible not for not having achieved a specific result which corresponds to an optimised dose, but rather for not having implemented reasonable actions to reduce doses (2).

In so far as optimisation is inciting and qualitative, regulations may only involve optimisation as a general requirement, since flexibility is required in its application within the regulatory framework together with the use of guidelines. This is the type of approach which the European Commission recommended in its Communication

of 31 December 1985 (3) and which was adopted for example when drawing up the French regulations concerning the principle of optimisation.
Radiological protection regulations should therefore have two qualities. They should not only be restrictive in terms of compliance with dose limits within the framework of a posteriori radiological protection monitoring but above all, they should encourage the promotion of the optimisation and improvement of behaviour in terms of a real management of radiological protection. Compliance with this principle is as essential as compliance with individual doses, in accordance with the logical sequence which French regulations represent in transposing the basic principles of radiological protection: "Equipment, procedures and the organisation of work should be designed so that occupational individual and collective exposure is kept as low as reasonably achievable below the limits which have been laid down (4)."

TRIAL CONTRIBUTION TO THE DEFINITION OF RADIOLOGICAL PROTECTION CULTURE

This quest for the "best" in the on-going effort to improve the quality of action required by the principle of optimisation is behind the "culture of radiological protection". If we use the definition of safety culture given in INSAG Document 4 published by the IAEA, we could define the culture of radiological protection as follows: "the radiological protection culture represents a state of mind in terms of radiological protection, that is, the view of radiological protection held by individuals and organisations, the value given to it and the interest shown in it. This state of mind influences occupational attitudes and practices" (5).
In practice, the dissemination of this radiological protection culture should involve a dynamic review when necessary of methods of thinking and working, so that radiological protection is given the place it deserves.
The key factors for the dissemination of this culture, which is based on individuals' awareness of the importance of radiological protection based on the assumption of prudence in the management of residual risk, are as follows:
- the commitment of the management to a policy of radiological protection and the adoption of the concept of optimisation within the management structure of the company (6-7);
- the definition of responsibilities, particularly through the clear allocation of tasks to those concerned;
- the motivation of the personnel, which takes the form of the establishment of objectives, personal responsibility and self-discipline in terms of radiological protection, knowledge and skills in the field of radiological protection as provided by theoretical and practical training, through good distribution of information and by extensive experience feedback at both international and national level.

Commitment of the management and adoption of the concept of optimisation within the management structure

Such a commitment may only exist and last if the management has the desire to encourage a culture of radiological protection because it considers it to be in the interests of the company and two main reasons for this can be put forward.
On the one hand, the cost of achieving the level of performance required in the field of radiological protection should not affect the attainment of the main objectives of the company. Since the principle of optimisation is a reference point for the company in that it underlines the necessity of improving the allocation of protection resources, thus avoiding waste, it is integrated in the management culture of the company.
On the other hand, the dissemination of such a culture, based on a policy of dose reduction can contribute to improving the image of the company in the eyes of the public and of its own personnel.
The expression of the will to encourage such a culture must be sufficiently clear as to convince the personnel that its performance in the field of radiological protection will be as important an assessment criterion as its performance in the area of production.
It is essential that the management confirms this movement by making a general policy statement and then goes on to demonstrate to the personnel, through its attitude, its total commitment to promoting the culture within the company. In practice, it should adopt a clear and effective approach by which the processes affecting radiological protection are reviewed on a regular basis and by taking a direct interest in the more important matters concerning radiological protection, whilst bearing in mind the basic message which is that exemplary dose management requires the implementation of the principle of optimisation.

Definition of responsibilities

The competent authority for radiological protection at national level should be able to oblige the company to promote a radiological protection culture by, in particular, holding the management responsible for allocating the necessary resources to comply with the principle of optimisation. In France, the regulations make the operator responsible for the organisation of work and, in addition he alone in practice can control dose rates, the operator bears the main responsability for the optimisation within nuclear installations.

This review of the legal responsibilities of the management of the company should be combined with the practical delegation of responsibilities together with a clear definition of tasks and the corresponding resources. Such a definition is essential to avoid any ambiguity and problems of shared responsibility, which could lead to overlaps, omissions or the watering down of responsibilities. The necessity for a rigorous definition of the tasks of all and the corresponding delegation of responsibility should be seen, first and foremost, as a policy for prevention aimed at identifying the errors made in order to avoid any repetition and not as a means of punishing the shortcoming which led to such errors.

Attitude of personnel

Motivating the personnel both during the preparatory phase and during operation, increasing their awareness of personal and collective responsibility in the field of radiological protection is one of the essential conditions for the success of any dose management system. The aim should be to ensure that all accept the principle of optimisation as a "state of mind" rather than an additional workload imposed by the management.

Their cooperation should go beyond a specific commitment for a given operation and justifies an on-going effort, since experience feed-back shows that nothing can ever be taken for granted in the field of radiological protection. Those involved should systematically adopt a disciplined and prudent approach based on professionalism, communications and "time for thought" giving pride of place to a questioning approach.

The development of professionalism implies the implementation of an ALARA training programme and the integration of the ALARA approach within the personnel training structure, followed by regular retraining. The mobilisation of the personnel during operations requires on-going information as to objectives and results and a policy of motivation, which consists, for example, in taking into consideration commitment to the ALARA approach in the annual personal assessment. The stimulation provided by a questioning attitude can be seen in the ALARA check lists used during project reviews, before and after completion, which correspond rather to encouraging thought rather than the drawing up of restrictions.

CONCLUSION

The introduction of a culture of radiological protection based on the principle of optimisation within the management structure of the company, will modify both the philosophy of action and behaviour. In speaking of the ethics of radiological protection, experts are not referring to a phenomenon of fashion but rather to awareness, in that the analysis of radiological risk includes both behavioural and technological aspects. The development of a radiological protection culture, based on the principle of optimisation, responds to these ethical concerns through reasonable behaviour based on social, economic and health protection value judgements and through the responsibility for the actions this requires. The success of the optimised management of exposure levels, through the motivation of all those concerned, relies on a policy of information and transparency in so far as the residual risk of ionising radiation and the resources used to reduce such risks to levels as low as reasonably achievable are concerned. The dissemination of this culture of radiological protection will probably be a determining factor in the dynamic improvement of information concerning nuclear risk and the public's acceptance of it.

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INCOHERENCES OF BRAZILIAN LABOUR LAWS FACE TO PRESENT RADIOPROTECTION CONCEPTS

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ABSTRACT

The brazilian labour legislation establishes, since 1950, some privileges for people working in activities which imply exposure to ionizing radiations. Comparing the present legal framework with technical radioprotection knowledge, one can detect several incoherences covering : classification of such activities ; additional payments ; reduced labour journey ; more vacations ; medical surveillance ; early retirements ; special norms for women .An analysis of these incoherences lead us to propose a new frame of labour rigts and radioprotection norms , coupling brazilian juridical principles and modern radioprotection knowledge .

INTRODUCTION

The brazilian labour legislation have established, since 1950,several privileges for workers exposed to ionizing radiations but, time evolution of the juridical scenery raised some incoherences, mainly when faced to time evolution of radioprotection knowledge. Topics covered by this study were : 1-classification of activities where exposure to ionizing radiations occurs;2-additionals of salary; 3-special work conditions : reduced labour journey,increased vacations, specific medical surveillance, and restricted norms for women ; 4-early retirements.The following text is divided in three topics : the present scenery,incoherences detected and suggestions for a new legal scheme.

THE PRESENT SCENERY

A- Classification of activities

Decree 3214 of the Ministry of Labour, in 1978, edited that an activity is considered **insalutary** when it is done over a ``tolerance limit'':concentration or intensity of a agent, bellow which a worker will not suffer any health damage during his professional life . Decree 83080, in 1978, recognized ionizing radiations as patogenic agents, harmful to health .The Labour Laws Consolidation - CLT , establishes that an activity is considered **dangerous** when, due to its nature or methods, it implies permanent contact with a dangerous agent , under considerable risk conditions.Regulation 3393 of the Ministry of Labour, in 1987, considering that ``any exposure to ionizing radiations is potentialy harmful to health'',classified such

activities as dangerous (where and when a potential risk is always present).

B- Additionals of salary

Law 1234, in 1950, established an additional of 40% for federal sector workers,when direct and permanently exposed to R-X or radioactive substances. Law 8270, in 1991, reduced this percentual to 5,10 or 20% , according to duration of exposures and potential doses of each worker.All other workers could also receive an additional of 40%, calculated not over their salaries but over the official national minimum salary, situation changed by regulation 3393, above mentioned, which estipulates an additional of 30% over each worker's salary.As a particular category , technicians in radiology, through Decree 92790, since 1980, receive 40% over their salaries, as ``life risk and insalubrity.

C- Work conditions

Brazil adopts convention 115 of the International Labour Organization, concernig protection of workers against ionizing radiations. Law 6189, among others, establishes that the National Nuclear Energy Commission - CNEN , is the authority encharged of editing norms concerning the safety of activities which implies exposure to ionizing radiations. Present general and basic norm is the NE-3.01 Diretrizes Basicas de Radioproteção (Basic Radioprotection Rules) [1], that defines as ``nuclear worker''somebody that, due to his professional activity, is exposed to doses over the level established for the public (1 mSv/year).

REDUCED WORK JOURNEY - federal sector workes and technicians in radiology have the legal right to work only 24 hours/week (normal journeys are between 40 and 44 hours/week).

VACATIONS - Once again, only federal sector workers and technicians in radiology have the legal right to vacations of 20 days/semester (normal vacations are 30 days / year). Since 1991, law 8112 prohibited the federal sector workers of ``selling''one third of their vacations period, in exchange of an additional payment.

MEDICAL SURVEILLANCE - CNEN [1] establishes a periodical examination ``according to the nature of the activity and dose absorbed''. Workers of the federal sector must be submited to this examination each semester , no matter the nature or dose of their activities. Examinations are also requested before job engagement , at resignation or retirement and when dose limits are surpassed.

NORMS FOR WOMEN - CNEN [1] establishes that pregnants cannot work in areas classified as ``controlled''(doses above 3/10 of the worker's limit),and also lower limits of accumulated dose to foetus (1 mSv) and to the abdomen of

women with reproductive capacity (10 mSv/three successive months).Law 8112, since 1991, establishes that women working for federal organims, must be excluded of activities and areas where they could be exposed to ionizing radiations ``during pregnancy and lactation periods''.

D- Early retirements

Federal Constitution establishes the right to retire after 35 years of work, for man, and 30, for woman,or after a reduced time(special retirement) for activities under special conditions, defined by law,considered harmful to the health or physical integrity of a worker.In consequence, 25 years in activities with exposure to ionizing radiations are sufficient for a special retirement, but federal sector workers are still waiting for a specific law to receive this constitutional benefice.

INCOHERENCES DETECTED

Analysing the present frame of the brazilian legislation [2], one can detect the following incoherences:

1- ionizing radiations are either considered as insalutary and dangerous agents, but since CNEN doesn't specify clearly a tolerance limit, at least two levels could be considered :
a- 1 mSv/year, which characterizes a ``nuclear worker'';or
b- 50 mSv/year, maximum recommended for nuclear workers;
but , for stochastic effects , this is a mere speculation, since the increasing of dose levels only increases their probability of occurrence.There is not a threshold dose isolating a safe from a harmful zone, as is the case for deterministic effects, where it is clear that the tolerance limit should be the deterministic levels for occurrence of each effect;

2- CLT specifies that, to be considered as dangerous, an activity should imply permanent contact with a dangerous agent, in a high risk situation. conditions not demanded by regulation 3393 that, in 1987, classified as dangerous,``any'' activity with exposure to ionizing radiations;

3- since 1950, workers of federal and others sectors are treated in different ways,even when doing similar activities, despite Decree 97458 that, since 1989, edits that ``characterization and classification of insalutary and dangerous activities done by workers of a federal organism, will be the same prescribed for others workers;

4- limitation of exposure time to a maximum of 24 hours/week conflicts with modern concepts of dose limits, admited as the best criterion for a safe activity; semestral vacations and early retirements present the same kind of incoherence (people with these privileges usually work in two or even more institutions, in similar jobs).In 1995, law 9032, prohibited

REVISION OF NATO STANDARDIZATION AGREEMENT (STANAG) 2345 "EVALUATION AND CONTROL OF PERSONNEL EXPOSURE TO RADIO FREQUENCY FIELDS"

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North Atlantic Treaty Organization (NATO) joint operations, which reach across borders of countries that have different standards, could present a logistical nightmare were it not for Standardization Agreements, called STANAGs. STANAGs are established to provide uniform policies and procedures to insure international cooperation and to maintain the most effective levels of commonality, compatibility, inter-change-ability and inter-operability in military operations.

Since signals produced by radar and communications equipment are not contained by national boundaries, a STANAG for Radiofrequency Radiation (RFR) is essential. The STANAG-2345 *"Control and Evaluation of Personnel Exposure to Radio Frequency Radiation"* establishes criteria for the evaluation and control of personnel exposure to radio-frequency radiation within NATO forces. It defines hazard assessment, allows for control measures, indicates actions in case of accidental overexposures and establishes permissible exposure limits.

The promulgation of this STANAG in 1979 was the culmination of deliberations among scientists and health professionals within the NATO community that began in May 1973. At the time of adoption it contained guidance based on the then state-of-knowledge. The standard has not been updated since it was issued, despite great advances in the knowledge-base defining health and safety aspects of RFR. Several standards setting groups recently updated their standards, including the American National Standards Institute revised safety guidance for RFR. This has left a large technology gap between the newer standards and STANAG-2345.

It is critical that a revision of the STANAG be developed that is in consonance with current scientific understanding and with modern national safety standards. In 1992, a program to update STANAG-2345 was initiated. Obtaining wide scientific consensus was deemed to be critical and many NATO countries participated in the effort. The Armstrong Laboratory co-directed, with the Italian National Institute of Health and the Italian Air Force, a NATO Advanced Research Workshop *Developing a New Standardization Agreement (STANAG) for Radio-Frequency Radiation* held in 1993 at Pratica di Mare Italian Air Force Base near Rome. The workshop served as a springboard for revision of the STANAG. This workshop brought together over 40 world class scientists from nine NATO countries to review the state-of-the-science on the biological effects of electromagnetic energy, with particular attention to radiofrequency radiation. Most of the participants either are currently heading or have had senior leadership positions in major research and standards setting organizations. Over forty presentations were made and papers have been published as a NATO Proceedings *"Radiofrequency Radiation Standards: Biological Effects, Dosimetry, Epidemiology, and Public Health Policy"*. The volume provides updated review by leading research and policy experts of the knowledge-base as it is transitioned to the standards setting process.

The process of rewriting the standard involved many hours of meetings in both the United States and in Europe and many very animated discussions. In October 1994, the United Kingdom Ministry of Defence and the Armstrong Laboratory co-directed a second workshop in London where general agreement on language was reached.

In February 1995, the United States Technical Representative briefed the 44th NATO General Medical Working Party on the progress made toward updating the STANAG. Custodianship of STANAG-2345 was transferred to the United States and the RFR Division of Armstrong Laboratory was appointed United States Technical Representative. Technical representatives from NATO countries met at Brooks Air Force Base, San Antonio, Texas, in May 1995. They made minor modifications in accordance with their rational concerns and then signed a consensus statement supporting the final draft STANAG, marking a further step in the process of transitioning knowledge-base to international policy.

The United States, United Kingdom and French Technical Representatives briefed the NATO Radio and Radar Hazards Working Party in September, 1995 in Brussels, Belgium. Involving the member nations in the entire process has paid off. It is anticipated that the technical consensus that has been formed will transition into ratification by all NATO members.

Highlights of the revised STANAG 2345 (Edition 2) are the following:
(1) The proposed STANAG 2345 (Edition 2) is a single tiered standard. It is based on the "controlled" level concept which is similar to the occupational level concept. Although both levels are safe, the military operational situation suggested that the controlled value was the most appropriate.
(2) The STANAG (Edition 2) is frequency dependent and covers the range from 3kHz to 300GHz. International safety standards have long recognized the frequency dependence factor and the revised STANAG has incorporated this essential concept.
(3) Most important, the STANAG 2345 (Edition 2) provides minimal safety guidance thereby insuring safe operations. National standards which are more restrictive or conservative are not superseded.

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1996 International Congress on
Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE Is Ionising Radiation a fact of life?

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MAJOR SCIENTIFIC TOPIC NUMBER ..... (see page 7)

ABSTRACT (See instructions overleaf)

Everyone attending the IRPA meeting will give a simple answer: YES

All of us know that natural ionising radiation arises in outer space, where cosmic rays are formed, and in earth, where radionuclides normally present in soil, air, water, food and even our bodies undergo radioactive decay.

How many among the "general public" know that radioactivity is the natural phenomenon of disintegration accompanied by the emission of ionising radiation? How much has been done to teach the population that penetrating radiation (gamma rays, alpha and beta particles, neutrons and muons) and radioactive material pervade the natural environment. And that exposure occurs by irradiation from sources outside the body and upon the decay of radionuclides taken into the body through ingestion and inhalation. Very little.

Even today, a hundred years after humankind learned about ionising radiation and radioactivity, the subject is not addressed at any stage in either primary and secondary schools world wide. One can not forget that an effort has been done, may be biased to gain public acceptance, by the nuclear industry to produce leaflets, brochures, teaching aids, even comics aimed to child and adolescent understanding but because there is not national curriculum requirement their impact is small.

If we agree that ionising radiation is a fact of life, it has to be thought as such in primary and secondary schooling. Then it is time to start some action to encourage the education authorities to introduce radiation and radiation protection in primary and secondary schools curricula in our own countries.

THE ETHICAL DILEMMA POSED BY OVERLY CONSERVATIVE ENVIRONMENTAL STANDARDS

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The searing introduction of the public to the human health effects from large doses of ionizing radiation came from the nuclear weapon explosions at Hiroshima and Nagasaki, Japan at the end of World War II. The horrors of a war using such weapons are part of the human consciousness. This introduction created fear and concern over ionizing radiation from man-made radionuclides or natural radionuclides enhanced by man's activities. This leads to a public over-emphasis on reducing exposure to man-made ionizing radiation whatever the expense might be. A study that evaluated costs of 500 life saving interventions found that radiation control activities had a median cost of $27 million (US) per life per year and were four of the top ten most expensive [up to $34 billion (US) per life per year]. [Tengs, 1994]

We were personally involved in a cleanup in northwestern Alaska where 1.1E8 Bq (3 mCi) of ^{137}Cs, buried over one meter deep in a mound 51 km from the nearest village, were removed because of fear and mistrust even though there was no physical hazard to anyone from the buried material. In the late 1950's an experiment, called Project Chariot, was proposed under the United States Plowshare Program (peaceful uses of nuclear explosives) to create a harbor along the northwestern coast of Alaska using nuclear explosives to move earth. The nuclear experiment was never conducted. However, as part of the extensive ecological and environmental studies in preparation for the experiment, the United States Geological Survey did tracer experiments in 1962 to determine how fallout radionuclides [total of 9.6E8 Bq (26 mCi) at the time of the experiment] might move through the tundra. At the end of the five day experiment, and in violation of the license authorizing use of the material, the material (contaminated soil, vegetation, wood and plastic) was buried on site in a mound about two meters high. (The material was to have been removed from the site.)

A combination of factors lead to the decision to remove the mound. The mound was re-discovered in September 1992, 30 years after burial. It reminded people of a difficult political battle against Project Chariot that was waged at that time by the local populace as well as a number of other residents in the Alaska. The concern felt in those days was rekindled by this rediscovery. The mound's location in a pristine hunting area, where people are dependent upon hunting, caused significant local concern about contamination of the food supply. The facts that the burial was in violation of the license and that the license permitted 1.85E11 Bq (5 Ci) to be used [although all evidence indicated only 9.6E8 Bq (26 mCi) was used] fueled the public outrage because some felt that the remaining 1.84E11 Bq (4.974 Ci) were used on site and were now missing. Also, the re-discovery occurred two months before a major election. The outrage created pressure on the politicians to act quickly and decisively. The Department of Energy (DOE) accepted the responsibility for subsequent activities. The isolated location meant that investigation, and removal,

if necessary, could not proceed along normal investigative lines where surveys and sampling and analysis are followed by a risk assessment and presentations to affected stakeholders before a decision on the appropriate remedial action. The cost of setting up the infrastructure to do the assessment work was so high that the extra cost of doing actual mound removal (which was the only acceptable alternative to the local populace) did not add that much more to the cost. Surveys were done, the mound was removed, and the contaminated soil was shipped to Nevada for burial in five weeks during the summer of 1993. Total costs were around $7 million (US). Fortunately, this work in a harsh environment where the only form of transportation to and from the location was by small aircraft landing on short dirt airfields, was accomplished without any injury to the workers although worker risk was always a prime concern.

This removal is a classic example of what Dr. Peter Sandman, a specialist in risk communication, talks about when he says that risk is the hazard (known to most technical people as the technical risk of probability times consequences), plus the public concern or outrage, i.e.,

RISK = HAZARD + OUTRAGE

Thus, hazard, or technical risk, can be zero and the "risk" can still be high because of public outrage. [Sandman, 1993] In ethical debates this concept might be framed with the understanding that one cannot divorce ethics from community and the values of that community. Often, the community value on man-made radiological contamination of the environment is not equal to biological likelihood of damage from that contamination. Therein lies the ethical dilemma that confronts many in the radiological protection community. How much effort, time, money and worker risk should be expended to mitigate a non-hazard as understood by the professional? How much better would society be if the money were not spent or were spent on higher priority activities? Most radiation protection professionals have probably confronted this dilemma sometime in their careers.

Another place where this dilemma occurs is in the setting of environmental standards for the cleanup of radionuclide contamination of soil. The United States Environmental Protection Agency is being pressed to set a single risk level for contaminates in soil and is giving consideration to a lifetime cancer risk of 1E-6. When the pathway analysis is done, cleanup is not feasible for a number of radionuclides (such as ^{137}Cs, ^{90}Sr, ^{232}Th) because the calculated concentrations are a fraction of background concentrations or are below detection levels of laboratory instruments. Besides, a risk level of 1E-6 is well below the natural background lifetime risk which ranges from 1E-3 to 1E-2.

It is interesting that for radon, the outrage factor is generally low to non-existent. In areas where the radiation protection professional might urge radon mitigation, the public is often apathetic. Is this because radon is a natural substance and ubiquitous? Or is it because the action falls to the individual and his/her pocketbook and is not something that can be blamed on others? Or, might it be a combination of these factors?

What might be done? As a matter of ethics, radiation protection professionals should speak out wherever the opportunity presents itself. Seek out ways to communicate beyond the typical dry, technical approach always hedging for uncertainty however small.

The United States Congress is currently considering legislation that would make funding for environmental cleanup based on risk. This would mandate that technical considerations be given primacy in funding priorities. This would significantly reduce the dilemma of the radiation protection professional.

The DOE is working diligently on interactions with community and regulatory stakeholders at DOE sites. This has led to some interesting results. At one facility, activists were given play money equal to the budget for environmental restoration and the costs for desired remedial activities and then asked to prioritize activities based on the money available. Various taxes and infrastructure charges were taken away before actual prioritization could begin. Several activists completely changed their positions on their individual items of concern when faced with the decision of how to prioritize across the whole site. At another site, when the activists were given access to areas that were formally off limits, their fears subsided and consequently their opposition to the DOE priorities also subsided. Across the complex, community groups are involved in future land use planning. Their recommendations have often been in favor of recreational and industrial uses for land lightly contaminated with radioactive materials. This mitigates the need to clean up to the more conservative residential-farmer scenario at much larger costs.

Another communication method is to bring in unbiased professors, who are skilled teachers, to teach risk assessment to stakeholders. Courses, given free of charge, have been taught by Dr. Genevieve Roessler (radiation risk) and Dr. Margit von Braun (chemical risk). Attendees have included national laboratory scientists, regulators, DOE staff, and representatives of women's groups, Native American tribes, homeowners associations, Congressional staff, etc. There is an interesting synergism that occurs when this mix of people learns together. Mistrust is replaced by understanding and comraderie. This enables all stakeholders to talk from a common basic technical understanding.

The communication efforts must be strong, deliberate and unending to overcome what we call the information half-life factor. Over time, knowledge of events decays away (similar to radioactive decay) among those who did know and a new generation appears that has no political awareness of past events. Any event that triggers a new wave of concern among those who have forgotten or were unaware of a past event creates a political firestorm which overcomes reasoned debate, as happened in Project Chariot. In new situations, it is better that the public learn of a possible concern by our proactive approach to informing them rather than learning from newspaper headlines or television soundbites.

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HUMAN RADIATION EXPERIMENTATION: A HEALTH PHYSICS PERSPECTIVE

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ABSTRACT: This paper observes ethical human experimentation can be considered in terms of two basic principles or tests: informed, willing and knowledgeable subjects; and expectation of benefits. A number of human experiments are evaluated in terms of these principles, including a sixteenth century toxicology experiment, the deliberate exposure by an x-ray pioneer, and the plutonium injection cases of the 1940's. The following rational ethic is proposed for the practice of health physics with respect to human radiation experimentation: ***At all levels, the health physicist has a professional as well as personal obligation to ensure that proper humanitarian requirements, including proper informed consent and willing subjects, are carried out with respect to human radiation experimentation, and must be convinced that the real or potential benefits to be derived from the experiment clearly exceed the potential detriment and risk.***

INTRODUCTION

Since the primary mission of the health physicist is protection of people and the environment from the harmful effects of radiation while at the same time permitting (and indeed promoting) its beneficial applications, human radiation experimentation is a topic of particular interest to the health physicist. However, the leading professional societies involved with radiation protection, including the International Radiation Protection Association and most national associations, provide little, if anything in the way of guidance. The Code of Ethics of both the oldest and by far the largest national organization, the American Health Physics Society, is published annually on the inside back cover of the Society Handbook, and provides nothing specific with respect to human radiation experimentation or public benefit. Indeed, this Code is, in a general way, patterned after the Code of Practice for professional engineers, minus the business aspects. Similarly the current Code of Ethics of the American Academy of Health Physics and its professional certification arm is vague and general, and refers only to obligations and responsibilities of the health physicist to the profession.

In developing a health physics perspective regarding human radiation experimentation, it is important to bear in mind what might be called the fundamental guiding principle of health physics. As elucidated in numerous publications, and in particular by the International Commission on Radiological Protection, this principle can be paraphrased thusly: *No person should be subjected to any additional radiation exposure unless the benefit to be derived clearly exceeds the risk to be incurred.* This is the fundamental underpinning of the concept of ALARA and the day to day practice of health physics. But, whose benefit and whose risk? Is it the individual, or the society, or somebody else who derives the benefit? And, is it the individual, or the society, or somebody else who incurs the risk? These are important questions that need to be considered by the health physicist in the overall context of human radiation experimentation.

TYPES OF HUMAN RADIATION EXPERIMENTATION

Human radiation experimentation can take many forms, and can be divided into two categories. Category I exposures are those in which the purpose is to direct administration of radioactivity or external radiation in order to study the radiation in relation to the individual. Perhaps the most common or obvious Category I exposures involve new and unproven radiological therapies or diagnostic procedures with significant exposure to radiation. A corollary, perhaps, to the preceding, are radiation effects studies in which a person or persons may be exposed to radiation to determine the effects on the healthy body. Category I exposures would also include biokinetics studies in which radioactive substances, albeit in small quantities, are administered for studies of uptake, distribution and excretion, and determination of various biokinetic parameters such as biological half-life for medical or health physics applications.

Category II exposures are those in which the radiation exposure is incidental to the primary purpose of the investigation or treatment. These include tracer studies of various kinds; additional diagnostic or evaluation procedures involving radiation exposure required by a nonradiological experimental treatment; and major exposure ancillary but essential to a particular treatment. Category II exposures might even be extended to include the exposures incurred by personnel involved in the preparation, handling, and administration of the radioactivity or external irradiation procedures. The radiation experimentation in the United States over the past half century has encompassed all types of Category I and Category II exposures.

POTENTIAL LEVELS OF INVOLVEMENT OF THE HEALTH PHYSICIST

The health physicist can become identified with human radiation experiments on many levels. He or she can be a member or even the leader of the experimental team carrying out the research. A somewhat lesser level of involvement would be to provide direct health physics support to the experimental team. More indirect but still at a higher level of involvement would be review and possible approval of the experimental protocols as a member of an Institutional Review Board, Human Subjects Committee or similar body, or review and approval of the experimental protocols from a professional health physics standpoint only. At a lower level is incidental involvement, which may arise out of and in the course of employment, such as when the colleagues of the health physicist seek advice or consultation on a specific aspect of the research, or when the health physicist is aware that the research is planned or being carried out by the institution. The lowest levels of potential involvement are incidental and not related to the employment or professional practice of health physics. For example, the health physicist may be aware that a human radiation experiment is planned or being carried out at another institution, or that waste shipped to his or her institution has been generated as a byproduct of a human radiation experiment.

ELEMENTS OF APPROPRIATE HUMAN EXPERIMENTATION

Although there are many points of view ranging from impermissibility of any type of human experimentation to the permissibility of extensive types experimentation on certain classes of people, it is the generally accepted viewpoint that human experimentation is permissible and even desirable under certain circumstances, and with certain assurances and controls. Much has been written with regard to ethics of human experimentation, and specific cases and scenarios present conundrums and paradoxes with ample basis for discussion and serious consideration, but in sum and substances, the basic principles are twofold. First, the subject(s) must be fully informed, knowledgable and understanding of the procedure and its attendant risks and discomforts and detriments, and must be unreservedly willing to participate. The second principle is that there must be some reasonable expectation of benefit.

The above principles are generic, much like the Biblical injunction "Thou shall not kill". And, like the injunction, detailed examination of each of the basic principles poses a host of further questions -- e.g. What is 'reasonable'? What constitutes 'knowledgeable'? Or 'understanding'? Whose benefit? Is it acceptable to experiment on a comatose patient if there is reasonable hope of achieving an otherwise impossible cure? -- are might best be left to detailed discussion and consideration by ethicists, scientists, semantisicists, clergy and laity and as such outside of the scope of this paper. More instructive would be an examination of past practices in the light of these two principles or tests, with an eye towards developing a rational ethic for health physicists with respect to human radiation experimentation.

PAST EXPERIMENTS IN THE LIGHT OF THE TWO GUIDING PRINCIPLES

In his diary *Journals Into Diverse Places*, the great French barber surgeon, Ambrose Paré (1510-1590) describes a visit to another country in which a test was made of the efficacy of the bezoar stone at preventing poisoning. It had long been believed that the bezoar stone, the hairball of a calf, would prevent the action of certain poisons. A cook, condemned to death the next day for stealing, was offered the opportunity to test the bezoar stone, with the understanding that if it worked, he would be set free. If, of course, it did not work, the cook would still die, although much less pleasantly and humanely. The cook cheerfully and willingly chose to participate in the experiment. He was administered a caustic poison, but the bezoar stone had no effect. Despite the ministrations of Paré and the attempts to

ease his suffering, the cook died an agonizing and horrible death.

Did this experiment, carried out nearly five centuries ago, meet the two tests? The answer is clearly affirmative. The cook was willing and informed, and understood the potential consequences of his participation. Thus the first test was met. There was also expectation of direct benefit to the subject, who would be set free and not executed if the experiment worked, as well as to the greater community from the knowledge gained vis-a-vis the effectiveness of this potential treatment.

The second experiment involved a group of men in the U.S. suffering from syphilis in the 1930's who were denied treatment so that physicians could observe the end stages of that disease. This experiment met neither of the tests; the subjects were completely and deliberately uninformed and presumably would have unwilling to forgo treatment. There was no expectation of benefit either to the subjects or to anyone else, for the end stages of tertiary syphilis were already well known, and had been for many years.

The third example is that of the first known human radiation experiment. On November 18, 1896, less than a year after the discovery of x-rays, American physicist Professor Elihu Thompson reported the results of deliberately exposing the little finger of his hand to x-rays, describing the physical damage, pain and suffering that he had incurred. Thompson's classic experiment met both tests: he was both willing and knowledgeable (although he did not anticipate such serious consequences, he knew as much about potential biological effects of x-rays as virtually anyone) and there was a clear expectation of benefit in the form of better understanding of the possible biological effects of x-rays, and of protection of people from excessive exposure.

Another example of radiation experiments, this one with a more equivocal answer to whether the two tests are met, relates to the plutonium injection cases. In the 1940's, 18 individuals thought to be suffering from incurable diseases that would result in death within ten years were injected with plutonium so that the biokinetics of this new manmade element could be studied and documented. Whether this experiment met the test of willing, informed, and understanding subjects is open to question; there was no documentation, and it is impossible to reconstruct what may or may not have been told to the subjects as they and the scientists carrying out the experiments are largely deceased. that there was an expectation -- indeed almost a guarantee -- of benefit in the form of new knowledge potentially useful for establishment of safety standards is unequivocal. But whether this benefit would have overridden the potential detriment, even if the subjects had been fully informed and willing is also open to question.

One final human radiation experiment bears consideration. In the 1960's, a group of mentally impaired children at the Fernald School in the northeastern United States was administered tracer doses of iron in a nutrition study. The fact that these were children, with serious mental deficiencies, raises the ethical question of the acceptability experimentation on groups or individuals whose ability to evaluate the merits of their participation is lacking or minimal. Even though the consent of the parents (who were likely not informed that a radionuclide was to be used in the study) was obtained, the ethical question remains. Permitting such a practice would seem to open the door to justification of the grotesque medical experiments carried out by the Nazis on groups they considered inferior and hence incompetent to judge whether they should participate as individuals.

TOWARD A RATIONAL ETHIC FOR HEALTH PHYSICISTS

The above lead to a simple rational ethic for health physicists with regard to human radiation experimentation: ***At all levels, the health physicist has a professional as well as personal obligation to ensure that proper humanitarian requirements, including proper informed consent and willing subjects, are carried out with respect to human radiation experimentation, and must be convinced that the real or potential benefits to be derived from the experiment clearly exceed the potential detriment and risk.*** In practice, this means that the health physicist should not participate in or provide support to experiments that do not meet the two basic criteria for human experimentation. Moreover, the health physicist has a professional and societal obligation to attempt to prevent any such studies with human subjects that do not meet humanitarian considerations. This is a fundamental responsibility of the professional practice of health physics, which after all, is devoted to the betterment of mankind.

In conclusion, it perhaps bears mention that the term *humanitarian requirements* as used in the above proposed ethic is of necessity dynamic and not static. What is acceptable today may not be so tomorrow. Consider, for example, the bezoar experiment described by Paré, which met both of the tests or criteria for human experimentation. By the standards of today, such an experiment would be unthinkable for manyethically based reasons. But we can and must learn from the past, so that as Santayana has pointed out, we are not condemned to repear our mistakes.

PERCEPTION OF RADIATION RISK FROM A CROSS CULTURAL PERSPECTIVE

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INTRODUCTION

Regarding radiation risk individual coping strategies range from apathy, no worry, avoidance, information seeking, changes in life style, inter alia. How they occur and when, is a necessary information for the development of better risk communication programmes. To address these points four particular situations involving radiation were chosen, namely indoor radon exposure, X-ray diagnostic, consumption of irradiated food, and radioactive waste management. Situations correspond to very different contexts, natural exposure (with indoor radon), daily life (with medical diagnostic and food consumption) and the industrial and energy context (with waste). From a cross-cultural perspective it was deemed fruitful to compare these situations in various countries.

RESEARCH METHODS

Three groups of scientists were involved in the study which was funded by the European Commission (1). IPSN (France) and CFR (Sweden) used the conventional approach to risk communication, that is considering successively the involved parties (transmitters, receivers), their goals, the content of the exchange and the exchange process. They designed large surveys respectively in France and Sweden to study public opinions, and analysed the four radiation risk situations by looking over health risk, authorities' actions, people opinions, experts' views, and recommendations about risk communication. IFS (Germany) designed communication strategies in relation with the social representations of the four situations. They did an inventory of risk debates and risk communication strategies in Germany, Netherlands and Belgium, evaluated those strategies and stressed the importance of the local contexts. Their approach, more qualitative, was based on in-depth interviews and the work in focus groups.

INDOOR RADON EXPOSURE

Indoor radon represents an everyday exposure affecting all the individuals who normally should be concerned with. In fact public opinion surveys show, first, that the radon risk is more or less well known by a majority in the US, UK and Sweden and by a minority in France and, second, that the radon risk is underestimated. Possible explanations are: radon is a natural gas and nature is seen as benevolent and friendly; the gas cannot be sensed so the risk is forgotten; the possible effect, i.e. lung cancer, takes a long time coming; the exposure-effect relationship is uncertain particularly for low exposures; private homes are under people own control so the risk may be considered as voluntarily taken; some equity exists, indeed everybody is exposed. Moreover information is lacking about radon, its possible effects, how to test and mitigate.

To increase people awareness, communication campaigns were launched (2,3). In all countries actions were under the responsibility of public authorities which consider radon as a public health problem. Actions in the US and Sweden were extensive and all the channels available were used (booklets, newspapers, TV, meetings, etc.). In the UK, communication was rather information (guides and brochures). In France actions were very few. In general people show apathy and a very low motivation to have their homes monitored and to reduce the radon concentration indoor without any monetary incentive from authorities. Indoor radon mitigation remains in most countries a home owner's decision and, up to now, authorities have not defined when and how mitigation should be compulsory. The radon risk situation is a case for which information can be successful, i.e. memorized (see the US experience), but fails to improve the situation, i.e. lower the radon levels. A possible way among others for improving the situation is the setting of new regulatory requirements on housing by public health authorities which simultaneously would have to inform the public on the reasons of such regulations.

X-RAY DIAGNOSTIC

In Sweden the general public does not see X-ray diagnostic as a possible health risk and its trust in medical personnel is high (2). Articles in the press upon the risk of cancer associated with X-ray examinations do not seem to have affected the perceived risk. Authorities have avoided to directly inform the Swedes about such a risk. Most of the governmental actions have been aimed at the medical profession with the purpose of reducing the patient dose without alarming the public. In France people who consider X-ray examinations dangerous are very few, from 11 % to 23 % depending on the survey, even if they are 40 % on the average to

think that the truth about the associated risks is not said (1). Efforts to reduce doses from X-rays are directed towards physician's practice rather than patient's behaviour.

In general there are no large differences in perception between countries. The risk is not taken into account by people and risk control is let to physicians. Communication with the public consists in giving neutral information for general education purpose. Major communication efforts are directed towards medical and dental practicians to make them aware of dosimetric data and quality insurance necessity.

FOOD IRRADIATION

Irradiation is a method for food preservation. It reduces the microbial contamination, removes bacterias and avoids germination. The process does not contaminate the food. No toxic effects have been found when the irradiation dose is delivered following an appropriate protocol. Intensive discussions about food irradiation took place in Sweden during the 1970's (2). A strongly negative public opinion led authorities to prohibit irradiation of food and importation of irradiated foodstuffs. The most common misconceptions are: the food becomes radioactive and unsafe; the food changes in terms of smell and taste; nutrients are less. In France food irradiation is permited and debates about the risks of the practice have not reached the public. The use of irradiation for food preservation is not known in general. In a 1995 French representative survey, people who feel personally endangered by the practice are 29 %, 26 % do not know precisely, and they are 41 % to think they are not (1). Food irradiation cannot be considered as a French issue.

To summarize, the technology and the sale of irradiated food are not allowed in many countries (for instance Sweden and Germany) although scientific evidence of negative effects on health is missing. This fact makes the difference between countries. People living in countries where food irradiation has been banned have a better knowledge of the practice due to the controversy which took place at the ban period. In other countries where the practice is authorized, as France, very few people know what it is.

RADIOACTIVE WASTE MANAGEMENT

The radioactive waste issue is a popular topic in Sweden because the site of a final repository for high level waste must be found (2). The waste management agency SKB has begun preliminary studies in different regions (Storuman, Malä, and Östhammar). Afterwards SKB must propose at least two locations candidate for the repository site, and to end the local population must take part in a referendum before any final decision. Regarding radioactive waste, public attitudes are affected by how people perceive nuclear power risks and if they feel that the responsible authorities can be trusted. Many public opinion surveys have been made in Sweden. Lay people do not consider themselves very well informed. For them science is incomplete and experts disagree; these opinions are clearly opposed to those of experts in radioactive waste who consider the risks as small and the technical problems of the disposal as solved. Radioactive waste are associated with the probability of large disasters, non reversible effects, as well as injuries to vegetation and animal life, and to future generations. Respondents think that waste risks stand for a large share of all nuclear power risks. In most cases they are strongly opposed to any possible siting of a repository in their community. A waste facility is considered to lead to negative health effects, to give the region a bad reputation, and not to bring large economic advantages. However the Swedes are convinced that the management of their nuclear waste is their own problem. SKB informs the public with booklets and articles. Up to now information campaigns conducted by SKB have not affected people opinions. The solution for going ahead in the siting problem is in introducing the concerned citizens in the decision making process.

In France the high level waste siting has not been solved and the waste management agency ANDRA is searching one or two underground laboratories for testing (1,4). The siting procedure was established in 1991 at the National Assembly level by the French deputees. The waste issue is perceived as an important society problem by a very large majority of people, as underlined by the results of the many surveys performed in France since 1976. Consistently lay people feel endangered by radioactive waste and express a very strong demand for safety, the logic here emphasizing social control rather than an individual control. Their trust in authorities is also very low.

Is the radioactive waste situation very different from those of chemical and industrial wastes, especially with respect to siting ? Not really. The siting issue for any type of waste is difficult to solve in all developed countries. Analysing risk communication in case-studies of the past, or discussing with experts in focus groups lead to similar views. People are always very interested and have an « intellectual civic » representation of such issues. There are no solutions based only on the technical arguments of authorities. Information given has no visible influence upon individuals' opinions and behaviours. The policy is to involve concerned communities in a global formalised negotiation process driven at a political level, which means the participation of elected bodies, interest groups and the operator which is a national agency in most countries. Health and social considerations enter in the process, but economic incentives as well. Anyway, the negotiation is completely site and social context dependent. Experience has shown that lessons gained from the analysis of success or failure at one site do hardly transpose to an other site and social context.

FROM SOCIAL REPRESENTATIONS TO COMMUNICATION POLICIES

The four radiation risk situations described above show many similarities from country to country but also some differences. There are no simple links between perceptions and the various risk coping actions which may be taken by people and authorities. A tentative to structure those links was carried out by IFS (Germany). To study the interaction between the individual and each of the four risk situations, that is how opinions are elicited about each particular risk, discussions in focus groups were conducted in Germany, Netherlands and Belgium. The concept of social representation was used to characterise the process of opinion formation. Six different representations formed the background of risk evaluation by focus group members. They correspond to specific argumentation patterns used when considering a particular risk: traditional indifferent, individual distant, rational economic, intellectual civic, environmental apprehensive, and to end progressive modern. The IFS study (1) showed first, that an individual had not only one representation for the four situations evaluated but took one or another depending on the situation, and second that the social image of a situation within a society might be derived from individual's representations. Then it was proposed for each situation a communication policy in accordance with its social representation, as illustrated in table 1. Communication policies are six : neutral information - general education; context information - local emphasis; comparing information - involving decision; feedback information - set up dialogue; contrast information - achieving discourses; no information - shift strategy.

Table 1 : Social Representations and Risk Communication Policies (IFS Study)
for indoor radon, X-rays, food irradiation and radioactive waste
in Germany(D), Belgium(B) and Netherlands (NL)

↗ = high → = medium ↘ = low ● = nothing

social representation	technology / process / product												risk communication strategy
	radon			*x-ray*			*nucl. waste*			*food irrad.*			
	D	B	NL	D	B	NL	D	B	NL	D	B	NL	
traditional indifferent	→	↗	↗	↘	↘	↗	↘	↗	→	↘	→	↗	❶ neutral information general education
individual distant	→	↘	→	↘	→	●	●	●	●	↘	↘	↘	❷ context information local emphasis
rational economic	↘	↘	●	↗	→	↘	●	→	→	→	↗	↘	❸ comparing information involving decisions
intellectual civic	↘	→	↘	↘	↘	→	↗	→	→	↗	↘	↘	❹ contrast information achieving discourse
environmental apprehensive	●	●	●	●	●	●	↘	●	●	●	●	●	❺ [no information] shift strategy
progressive modern	●	●	●	↘	●	●	↘	●	●	●	●	●	❻ feed-back information set up dialogue
	↓↓↓↓↓↓↓↓ social image of technology in society			↓↓↓↓↓↓↓↓ social image of technology in society			↓↓↓↓↓↓↓↓ social image of technology in society			↓↓↓↓↓↓↓↓ social image of technology in society			recommended risk communication strategy related to a particular social image

CONCLUSION

For the four radiation risk situations retained more similarities between countries have been observed than differences with respect to perception and communication. In a search of rationalization, a tentative has been made to relate social representations to communication policies which proposes interesting views which must be precised in future work. Results obtained within the cross-cultural approach offer fruitful comparisons for researchers and decision makers involved in risk management, and for this reason more cross-cultural studies in these fields must be encouraged.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE

Preliminary study of risk comparison and risk perceptions on selected population groups in Yugoslavia

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MAJOR SCIENTIFIC TOPIC NUMBER 8.10. (see page 7)

ABSTRACT (See instructions overleaf)

The public is constantly harangued about all sorts of risks and its perception of risks plays an important role in governmental decision making.

This paper presents results of one of preliminary studies of the attitudes to ionizing radiation compared to other risk factors.

In the area of social psychology attitude is defined as a tendency to positive or negative reaction towards certain characteristics, objects and situations. It is caused by structure and content of the whole personality, especially by its subconscious part, and defined by the past, education and social surroundings.

Primary goal of this research was to divide risk perception attitudes according to different variables such as: sex, education level, years of life, history of occupational exposure and so on.

SOME ETHICAL PROBLEMS IN RADIATION PROTECTION

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ABSTRACT

The paper is focussed on the work of international organisations to establish ethical principles for protection against ionizing radiation.

INTRODUCTION

The United Nations have devoted much efforts to the protection of man. Already in the Universal declaration of human rights adopted in 1948 (1), it was stated in article 3: "Everyone has the right to life, liberty and the security of person".

In the field of establishing ethical principles for radiation protection the work of the International Radiological Protection Commission (ICRP) has been of outmost importance (2).

The ethical issues in radiation protection are receiving an increased international interest (3). Several radiation protection recommendations of importance from an ethical perspective have been published since 1990 and a review is timely to summarize the present ethical issues in the radiation protection field.

INTERNATIONAL RADIATION PROTECTION STANDARDS IN AN ETHICAL PERSPECTIVE

ICRP Publication no. 60

The recommendations of ICRP have today a profound influence on radiation protection all over the world. The ICRP has in Publication no. 60 (2) elaborated a conceptual framework for radiation protection mainly of ethical nature but also based on experimental work and risk assessment. An important presumption of the ICRP conceptual framework is that even small radiation doses may produce some deleterious effects. The three main principles of ICRP for proposed or continuing practices are the following:

1. The justification of a practice,
2. The optimisation of protection (ALARA: As low as reasonable achievable radiation doses, considering economic and social factors),
3. Individual dose and risk limits.

Medical exposures are usually intended to provide a direct benefit to the exposed individual. If the practice is justified and the protection optimised, the dose in the patient will be as low as is compatible with the medical purposes. Any further application of limits might be to the patient's detriment. ICRP therefore recommends that the dose limits should not be applied to medical exposures.

The system of radiological protection recommended by ICRP for intervention is based on the following two principles:

1. The proposed intervention should do more good than harm,
2. The form, scale, and duration of intervention should be optimised (according to ALARA).

IAEA Fundamentals for radiation protection

The International Atomic Energy Agency has in 1995 adopted safety fundamentals relating to radiation protection and the safety of radiation sources (4). These fundamentals are mainly of ethical nature and they are partly based on the protection principles given in ICRP publication 60 (2).

According to IAEA the primary aim of radiation protection is to provide an appropriate standard of protection and safety for humans without unduly limiting the benefits of practices giving rise to radiation

as well (9).

ICRP in para. 16 in (2) believes that the standard of environmental control needed to protect man to the degree currently thought desirable will ensure that other species are not put at risk. The same idea is stated by IAEA in (5) in principle 2 (radioactive waste shall be managed in such a way as to provide an acceptable level of protection).

IAEA in (5) states in principle 4: Protection of future generations as follows: Radioactive waste should be managed in such a way that predicted impacts on the health of future generations will not be greater than relevant levels of impact that are acceptable today. This principle is derived from an ethical concern for the health of future generations. In the establishment of acceptable levels of protection the latest ICRP and IAEA should be taken into account.

The fact that ICRP does not happen to use the term precautionary principle does not mean that it does not use the concept, which, of course it does. In fact, the whole philosophy of ALARA and protection against stochastic effects is based on not proven assumptions on radiation harm (the hereditary harm from radiation has never been demonstrated in humans, nor has cancer at low doses), because these assumptions seem to be scientifically justified.

The substitution principle (to replace a harmful agent with a less harmful) is not used by ICRP but follows directly from ICRP's first principle for practices (justification). Inter alia ultrasound is today used for prenatal examinations instead of X-rays.

Some moral problems to be deliberated in the future are: Adaption to radiation, The existence in nature of higher doses than the established limits, and The variation of the radiation risk among individuals.

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exposure or incurring disproportionate costs in the case of intervention. This primary aim is expressed by the following specific protection and safety objectives:

Protection Objective: to prevent the occurrence of deterministic effects in individuals by keeping the doses below the relevant threshold and to ensure that all reasonable steps are taken to reduce the occurrence of stochastic effects in the population at present and in the future.

Safety Objective: to protect individuals, society and the environment from harm by establishing and maintaining effective defences against radiological hazards from sources.

These objectives are achieved by the application of eleven fundamental principles.

IAEA Fundamentals for radioactive waste

Many of the hazards associated with radioactive waste are similar to those associated with other toxic waste. However, the nature of radioactive waste implies the possibility of exposure to ionizing radiation. An acceptable level of protection therefore needs to be provided against radiation hazards as the main source of harm of radioactive waste. The principles of radiation waste management have thus a bearing on protection also against radiation.

The International Atomic Energy Agency has in 1995 adopted Safety fundamentals relating to radioactive waste management (5). These fundamentals are mainly of ethical nature and they are also as the IAEA Fundamentals for radiation partly based on the protection principles given in ICRP Publication no. 60 (2).

According to IAEA the *objective of radioactive waste management* is to deal with radioactive waste in a manner that protects human health and the environment now and in the future without imposing undue burdens on future generations. This objective is achieved by the application of eleven fundamental principles.

ILO Convention 115

There exist also some ethical principles in the International Labour Convention no. 115 concerning the protection of workers against ionizing radiations (6). This convention came into force in 1962. In article 5 of the Convention the main protective objective is formulated as follows: "Every effort shall be made to restrict the the exposure of workers to ionizing to the lowest practicable level, and any unnecessary exposure shall be avoided by all parties concerned."

OTHER RECENT WORK ON ETHICAL ISSUES IN RADIATION PROTECTION

Ethical issues in radiation protection have since the latest ICRP recommendations in Publication no. 60 were adopted in 1990 (2) been dealt with by inter alia Silini (7), Prêtre in ref. (3), and Shrader-Frechette in ref. (8). Silini tries to answer the question if the present system of radiation protection was founded on sound ethical principles. His answer is yes but with some remarks. Clarifying the meaning of the principle of justification and including environment in addition to humans are improvements proposed. To protect individuals as human beings and not as workers or public is another suggestion. Prêtre attempts to connect the three ICRP protection principles with the four ethical principles: Responsibility, Respect of life, Justice, and Common good. He shows that, in general, all ethical aspects in some way or other flow into the ICRP principles of radiation protection. Shrader-Frechette studied risk and ethics in a radiation protection perspective. In her paper she argues that evaluation of radiation risk often include a spectrum of extreme and unrealistic views. Ethical analysis of risks in terms of uncetainty, equity, consent, and compensation can help to provide more just, moderate, and realistic evaluations.

DISCUSSION

The linear hypothesis of radiation protection and the ALARA principle are considered to be ethically acceptable guidelines and similar guidelines should influence our attitude toward exposures to harmful chemicals

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ABSTRACT

THE "G.R.R.I.N.S." AND THE INFORMATION OF HEALTH PROFESSIONS

Professeur J.C. ARTUS, Dr BATT/SANSON, Dr G. BROGLIA, Dr H. FROSSARD, Mr R. GARCIA, Dr J.M. GELAS, Dr R. GRANIER, DR C. MAYER, Mr J. MERCIER, Mme J. RIOU, Dr J.M. VINOT.

The "Groupe Régional de Reflexion et d'Information sur le Nucléaire et la Santé " (Régional Group of Reflexion and Information about Nuclear energy and health) is a working party created after Tchernobyl disaster in order to provide the framework for a pluridisciplinary reflection about the risks induced by the development of Nuclear technology on health and to carry out informing actions towards professionnal working in health services.
(Association law 1901)

The public preoccupation on the sanitary incidence of the ionisant radiations have to induce health professions to be able to answer theirs patients interrogations.
Experience shows that practitioners are not sufficientty informed and we have to propose them correct informations outwards all the polemics. All this is done in a health worry.
This is the GRRINS goal fixed to itself.

MEMBERS :

Those taking place in the working party : Doctors from occupational medecine or Nuclear medicine, biologists, physicists, hygienists, radio toxicologists, chemists and radiologists.

AIMS :

- Training students, doctors, chemists.
- Informing medical professionnals, newspapers, decision makers.
- Contributing to the actions of local information committees close to energy plants and other publis structures (department board for hygiene).

PLACE OF SETTLEMENT :

South of France (Rhône Valley), region with a large nuclear establishment.

ACTION :

- Investigation towards 13 000 practitioners, doctors, chemists, dentists, vets from Vaucluse, Gard, Hérault, Aude, Lozère and Pyrénées Orientales.
- Spreading of educational documents (films, video films, lectures, slides, books) produced under the aegis of learned companies.
- Participation in various teaching (post graduate and ordinary degrees, dissertations, training courses).
- Publication of a report "the GRRINS'Letter" with a circulation of 10 000.
- Yearly organisation of a day's work on a particular subject entitled "the Lascours Talks" :
  * 1990 : Ionizing radiances in daily life.
  * 1991 : Medical radiation (diagnosis and therapy).
  * 1992 : Ionization of food products.
  * 1993 : Are there more cancers around Nuclear centers ?
  * 1994 : Radioactive waste : which risks for health ?
  * 1995 : The domestic radon.

RESPONSES TO RADON REMEDIATION ADVICE

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Abstract

It is estimated that around 100,000 homes in the UK exceed the Action Level of 200 Bq m^{-3}. About 20,000 of these homes have been identified and the householders have been advised to reduce the radon levels. As part of an initiative to encourage remedial action a short questionnaire was sent to around 10,500 houses with radon at or above the Action Level to obtain data on any remedial action taken and the factors which influenced the decision. The results showed that around 20% of householders who replied had taken some form of effective remedial action and that cost was the major consideration. The best estimate of the overall rate of remediation is about 10%.

1 Introduction

Doses from radon are the largest component of the average radiation exposure in the UK and are estimated to account for about half of the annual dose of 2.6 mSv received by a typical member of the population (1). The majority of the exposure to radon occurs in dwellings as most people spend most of their time at home. Levels in homes can vary, and an extensive survey of the radon levels in UK dwellings has been carried out. The UK Government Action Level for radon in dwellings is set at an annual average whole-house concentration of 200 Bq m^{-3}. Householders with measured concentrations at or above this Action Level are advised to reduce the level in their homes and given information on mitigation techniques (2).

To date, measurements have been made in over 200,000 homes in the UK as part of a programme supported by Government Departments (3). One area which has been found to have a higher proportion of homes above the Action Level is the southwest of England and in particular the counties of Cornwall and Devon where it is estimated 36,000 homes are affected. Householders who had measurements of radon at or above the Action Level in these two counties were surveyed by postal questionnaire to obtain information on the types of remediation , if any, that they had undertaken.

2 Survey

Some 10,500 householders where surveyed by questionnaire. Questions included the remedial method used, the costs incurred and also reasons why they had decided not to undertake mitigation. Multiple choice reply was possible for several types of method or reason and householders could add any points that were not covered by the questionnaire. Data were obtained from over 5,000 completed questionnaires and then analysed. Around a third gave data on remedial measures and the remainder gave reasons for not taking advice to remediate.

Information on remedial measures used by the householders indicated that around 20% of those who replied had used an effective remedial measure such as underfloor depressurisation, increased underfloor ventilation, positive pressurisation of the living space or permanent additional ventilation . Multiple methods were employed by many householders, some having used five different types of remediation. The types of effective remedial measure used are shown in Figure 1. The costs of installing each type of remedial measure ranged up to £6000, but over 75% of householders spent less than £1000. The average cost incurred was about £700.

3 Remediation

The methods of remediation usually employed in homes fell into two main categories: those which prevent the radon from entering the living area of the house, for example underfloor depressurisation (sump), increased underfloor ventilation or sealing of cracks or other entry points in the floor and walls; those which reduce the level of radon in the dwelling, for example permanently increasing ventilation by the installation of trickle vents in windows. Sealing alone is not recommended but should be combined with another effective method. Householders are encouraged to choose the most effective methods of remediation. The proportion of householders installing underfloor depressurisation and positive pressurisation is banded by original radon level in Figure 2. This illustrates that the proportion of householders using these effective methods increases with radon level.

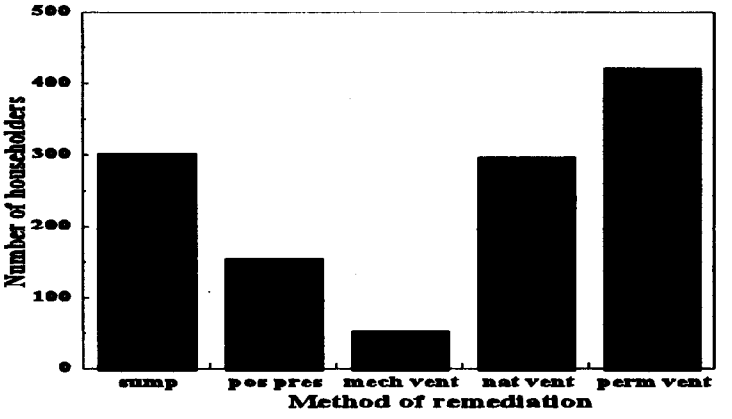

Figure 1 Types of effective remedial measure used

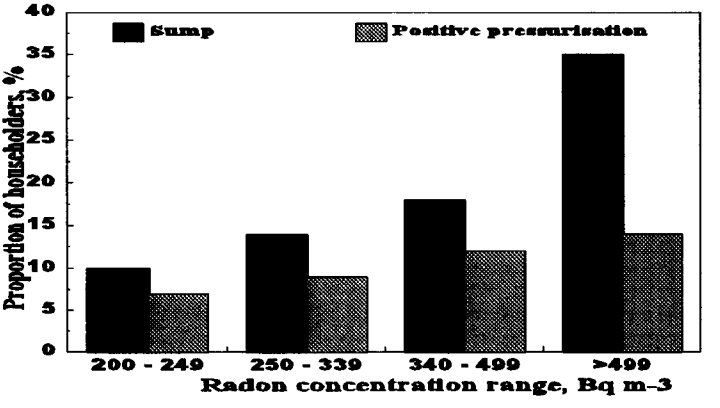

Figure 2 Comparison of the proportion of householders using the most effective remediation methods with radon level

The most effective form of remediation, installation of an underfloor depressurisation system or radon sump, was chosen by over 300 householders, with some of these using additional measures such as sealing. Underfloor depressurisation is estimated to reduce the radon level by about 90% on average (4). The use of this method should reduce the average concentration well below the Action Level, with an estimated reduction in the average level of around 700 Bq m^{-3}. The average cost to install this type of system was found to be £950, but for those householders who installed the system themselves the cost reduced to about one third or £300, making it very cost effective.

4 No action

Of the 5,000 householders who provided information, around 4,000 gave reasons why they had not carried out remediation. The main reasons can be divided into four categories: cost; no perceived risk; difficulty in implementing remediation; soon to move house. Some householders felt that the responsibility lay with the landlord or that taking the remedial measures would cause too much upheaval. The majority of householders gave more than one reason for not taking action. The distribution is shown in Figure 3.

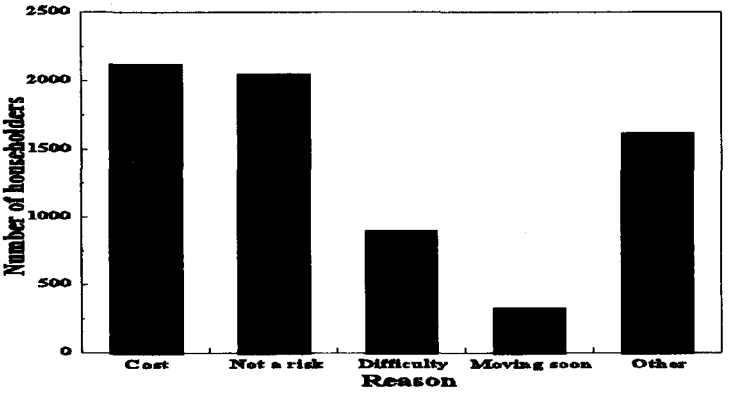

Figure 3 Reasons given for not taking remedial action

Over 50% cited cost as one of the reasons for not remediating and a similar percentage did not perceive there to be a risk. However, around 700 householders, some 17% of those who gave reasons for not remediating, did indicate that they still intended to remediate.

5 Conclusions

Of 10,500 householders with radon above the UK Action Level of 200 Bq m^{-3} who were given advice to reduce their radon levels, around 20% of the half who responded had used at least one method of remediation which would reduce levels by 50%. Thus the overall remediation rate, including householders who did not respond, is around 10%.

The methods chosen by the householders reflect the radon level in their homes, those with higher levels having used more effective methods. This suggests that the provision of clear and firm advice is an important part of any radon programme. Over 300 householders installed the most effective form of mitigation, underfloor depressurisation, with an estimated average reduction in their level of around 700 Bq m^{-3}.

The average cost of mitigation was around £700, with the most effective method of remediation, underfloor depressurisation, costing around £950 on average. Costs are tending to fall as experience is gained. Costs were reduced significantly if remediation was carried out by the householders themselves.

For householders who had not remediated, a major factor was the cost involved, but, radon was not perceived as a risk by many householders despite the clear information provided by Government and NRPB. This phenomenon of denial is well known (5) and indicates the need for more perceptive ways of providing advice and encouragement.

6 Acknowledgement

Radon surveys in the UK are supported by the Department of the Environment and the Departments of State for Wales, Scotland and Northern Ireland.

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RADON: A CASE FOR PUBLIC PERSUASION

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Abstract

The importance of reducing individual exposure to elevated levels of radon is well understood by radiation protection specialists, and successful methods of locating the areas most at risk have been developed. However, less attention has been paid to informing the general public about the health risks and encouraging those in radon-prone areas to take action. In the United Kingdom, techniques have been developed to persuade householders in high radon areas to take advantage of a Government scheme that provides free long-term measurements of radon in the home. Improvements in the methods of contacting householders in the target areas and in the presentation of the facts has resulted in a twofold increase in the rate of take-up of measurements since the first large-scale surveys.

Introduction

In 1990, the National Radiological Protection Board recommended to the Government a strategy to tackle the domestic radon issue [1]. The Government accepted the advice and commenced a programme to identify the majority of homes with elevated levels by the year 2000 [2,3]. The plan, based on the concepts of a radon Action Level, set at 200 Bq m^{-3}, and the designation of radon Affected Areas, is consistent with the recommendations of ICRP[4]. In addition, provision was made by the Government to fund long-term radon measurements in the homes most at risk. The standard measurement method is to install two passive integrating radon monitors in each home to be tested, one in the living area and one in a bedroom.

NRPB has designated radon Affected Areas in several parts of the UK and, on behalf of the Government, carried out large-scale surveys in those parts most at risk [5,6,7,8,9]. The initial objective of these surveys has been to encourage householders to take up the offer of free radon measurements in their homes. The outcome is the identification of dwellings above the Action Level, the first step towards the ultimate objective of persuading householders to carry out remedial measures in their homes. This reduces the high radiation doses received by the householder, his immediate family, and subsequent occupiers of the dwelling. It is estimated that there are some 100,000 dwellings above the Action Level in the UK.

Identification of households at risk

The methodology used to define radon Affected Areas in the UK is based on assessing the risk of dwellings being at or above the Action Level by 5 km grid squares of the Ordnance Survey Grid [10]. The technique results in an estimate of the percentage of dwellings in each grid square that will exceed the Action Level. The squares are ranked in descending order of probability to give a listing of the most radon-prone areas. The addresses of all dwellings in each square are obtained from the Post Office Address File (PAF) maintained by the Post Office and available on CD-ROM. These addresses are used in postal radon campaigns as discussed below. The PAF, which is regularly updated, also provides information on local council and health authority districts.

Contacting the householders

The first large-scale offers of free measurements to householders were made using a specially designed leaflet delivered direct to every dwelling by the normal mail. The leaflet gave simple facts about radon and the free measurement scheme and contained a small form to apply for the test. The householder was asked to complete the form with his name and address and return it in an envelope to NRPB at his own expense. Posters advertising the scheme were distributed to local authorities for display in council offices, libraries, and so on. Further supplies of the leaflet were also made available at these locations.

Subsequent campaigns, in different parts of the country, used similar leaflets but these were enclosed with a letter individually addressed to the householder. The individual address was already printed on the reply form.

To apply for the test, the householder needed only to complete the form with his name and return it to NRPB in a pre-paid and pre-addressed envelope.

In the final campaign reported here, in parts of the English county of Somerset, an extra leaflet was included in the package sent to householders. This leaflet, prepared by the local council and health authorities, contained a strong anti-smoking message as well as encouraging the householder to apply for a free radon measurement. The local authorities also organised a vigorous media campaign to coincide with the mailing of the first batch of letters.

The response

The difference in the response rates to these techniques is quite pronounced as illustrated in the table. The most striking point is the high response rate compared with leaflet campaigns on other topics when a response of a few percentage might be expected. Even in the early campaigns in Cornwall, Devon, Derbyshire and Northamptonshire, when the householder received only the leaflet and was required to provide an envelope and pay the return postage, the response rate was over 10%. When the leaflet was delivered in an envelope addressed to the occupier at an individual address, a return envelope provided and the return postage pre-paid, the response rate doubled to 20% or more as demonstrated by the second campaigns in Northern Ireland and Somerset. A further significant increase in the response rate appeared when the mailing was accompanied by a media campaign as was the case in the Scottish regions and in the first Somerset mailings. However, the responses from the two campaigns in both Somerset and in Northern Ireland indicate that the effect of media coverage is quite short-lived and that the response rate drops back to a lower rate once the initial media interest wanes.

Conclusions

Positive response rates of around one in four have been achieved with carefully prepared literature and the provision of return stationary with pre-paid postage. Local media coverage increases the number of positive replies to almost one in three. The same approach could be used to maintain contact with householders with elevated radon levels so as to encourage them to undertake remedial action.

Acknowledgements

Radon surveys in the UK are supported by the Department of the Environment and the territorial Departments.

Table. Response rates to large-scale offers of free radon measurements

Area	Technique	Number sent	Positive replies	
			Number	%
Cornwall & Devon (first campaign)	Leaflet alone	640000	77000	12
Derbyshire & Northamptonshire (similar approach)	Leaflet alone	350000	38500	11
Grampian & Highland Region (concurrent LA campaign)	Leaflet, letter and return envelope	2260	720	32
Northern Ireland I (within a year of designation)	Leaflet, letter and return envelope	20900	5100	24
Northern Ireland II (over a year since designation)	Leaflet, letter and return envelope	26700	5500	20
Somerset I (concurrent media campaign)	Leaflet, letter and return envelope	40000	12400	31
Somerset II (no concurrent media coverage)	Leaflet, letter and return envelope	60000	14300	24

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FUZZY MEASURE ANALYSIS OF PUBLIC ATTITUDE TOWARDS THE USE OF NUCLEAR ENERGY

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(1) INTRODUCTION

It is important to identify the structure of public acceptance or rejection when new technologies are developed and implemented. The structure of attitudes should have the essential attributes and their interrelation. In such a structural analysis the attitudes need to be decomposed into meaningful attributes by a suitable model. However, the data obtained in this type of study may be more or less subjective and fuzzy, and the following problems may be pointed out:

(1) A man does not always have an additive measure such as probability to evaluate fuzzy objects.
(2) The attributes of an object in his evaluation process are not always independent of each other.

In either case a linear model may not be applicable.

This paper is concerned with applying fuzzy measures and fuzzy integrals to analyze public attitude towards the use of nuclear energy.

Probability theory deals with the definition and description of models involving the probability concept. Probability judgements are concerned with repetitive events which have basic similarity. Most applications of probability theory may be interpreted as special cases of random processes. If, as in many games of chance, equal probabilities are assigned to each of the simple events of a given finite fundamental probability set, then the probability of realizing a compound event (success) defined as the union of specified simple events ('favourable' simple events) can be computed as the ratio of the number of favourable simple events to the total number of simple events. It is on the basis of these considerations that probability is deduced by resolving the various outcomes into a number of equipossible alternatives. However, when we apply the term probability to a non-repetitive event or an isolated case, for example, probability of 'Julius Caesar's visit to Great Britain', or probability of a certain country secretly producing nuclear weapons, it is impossible, at any rate in any obvious way, to generate a sequence of trials and thus measure the probability of its occurrence by means of a frequency ratio. We have, therefore, to estimate it by a more or less subjective intuitive appraisal of such evidence as we may have. Since in such cases a universally acceptable quantitative estimate of the degree of our confidence in the statement cannot be given, probability when used in this sense cannot form a part of a scientific assertion in the conventional sense.

To avoid confusion, it is, therefore, better to restrict the use of probability in a scientific sense to repetitive events only and to use the term 'possibility' when we wish to speak of our expectations of non-repetitive events, as suggested in fuzzy set theory. In addition to the probability and possibility measures, belief, credibility, certainty, necessity, plausibility measures, etc. are also introduced as special cases of fuzzy measures.

(2) FUZZY MEASURES

A fuzzy measure of Sugeno is an extended probability measure in one sense, which assumes, in general, only monotonicity without additivity.

Let X be a universal set and $\mathscr{B}$ be a Borel field. Then a set function g defined on $\mathscr{B}$ with the following properties is called a fuzzy measure.

(i) $g(\emptyset) = 0, g(X) = 1$ (boundedness)

(ii) If $A, B \in \mathscr{B}$ and $A \subset B$, then $g(A) \leqslant g(B)$ (monotonicity)

(iii) If $F_n \in \mathscr{B}$ for $1 \leqslant n < \infty$ and a sequence $\{F_n\}$ is monotone (in the sense of inclusion), then $\lim_{n\to\infty} g(F_n) = g(\lim_{n\to\infty} F_n)$ (continuity)

A triplet $(X, \mathscr{B}, g)$ is called a fuzzy measure space, and g is called a fuzzy measure of $(X, \mathscr{B})$.

For applications, it is enough to consider a finite case. Let K be a finite set $K = \{s_1, s_2, ..., s_n\}$ and P(K) be a class of all the subset of K. Then a fuzzy measure g of (K, P(K)) is characterized by the first two properties since the third one implies continuity.

In particular, $g(\{s_i\})$ for a subset with a single element s_i is called a fuzzy density like a probability density. We denote $g^i = g(\{s_i\})$. As a special form, g_λ fuzzy measures have the following characteristics:

$$A \cap B = \emptyset \Rightarrow g_\lambda(A \cup B) = g_\lambda(A) + g_\lambda(B) + \lambda g_\lambda(A) g_\lambda(B) \quad (-1 < \lambda < \infty)$$

When the fuzziness coefficient $\lambda = 0$, λ-fuzzy measures are probability measures. $\lambda > 0$, belief measures; $-1 < \lambda < 0$ plausibility measure.

(3) FUZZY INTEGRAL

Let h be a measurable function from X to [0, 1]. Then the fuzzy integral of h over A with respect to g is defined as

$$\int_A h(x) \circ g = \sup_{\alpha \in [0,1]} [\alpha \wedge g(A \cap F_\alpha)]$$

where $F_\alpha = \{x, h(x) \geqslant \alpha\}$ and $\wedge$ stands for minimum.

In the above definition , A is the domain of a fuzzy integral which is omitted if A is X.

Now let us see how to calculate a fuzzy integral. For simplicity, consider a fuzzy measure g of (K, P(K)) where K is a finite set previously defined.

Let $h : K \to [0, 1]$ and assume without loss of generality that $h(s_1) \geq h(s_2) \geq \ldots \geq h(s_n)$. Renumber the elements of K, if not in descending order. Then we have

$$\int h(s) \circ g = \bigvee_{i=1}^{n} [h(s_i) \wedge g(K_i)]$$

where $K_i \triangleq \{s_1, s_2, \ldots, s_i\}$ and $\vee$ stands for maximum.

A fuzzy integral can be used as a model of subjective evaluation of fuzzy objects where the attributes of an object are measured by a fuzzy measure and the characteristic function of an object is integrated with respect to a fuzzy measure. In the example about the nuclear power plant let h : K [0,1] be the characteristic function of the plant, i.e. the function expressing the characteristics of the plant. For example, we set h ('output'), h ('safety'), h ('price') etc. Then the overall evaluation of the plant is given by the fuzzy integral of h with respect to g, i.e. the grade of subjective importance of each attribute. As is clear from the definition of a fuzzy measure, a fuzzy measure is not only a subjective scale for guessing whether an a priori non-located element in X, a universe, belongs to a subset A of X, but also concerned with such cases as the grade of subjective importance of an attribute referred to in the foregoing from a practical point of view. The fuzzy integral model is applicable to non-linear cases, where one does not have to assume independence of one attribute from another. In the example given it is highly possible that there is a certain dependence between 'safety' and 'price'. If this is the case, we cannot use a linear model as far as we regard 'safety' and 'price' as the attributes of the plant.

We have to consider the dependence of attributes from two points of view. One is objective dependence such as between 'safety' and 'price', and the other is subjective dependence. Even if an attribute seems physically independent of another, one may consider that they are subjectively dependent in some cases.

The uncertainties involved in human judgement are usually non-additive and fuzzy, and therefore could be treated more adequately with the fuzzy theory.

(4) CONCLUSION

We applied the fuzzy measures and fuzzy integrals to analyze public attitude towards the use of nuclear energy by distributing questionnaires to about 100 students of engineering department of Kinki University, Higashi-Osaka, Osaka, Japan. Before and after the Chernobyl Accident we noticed a distinct difference in mental structure. Before the accident, the students of pro-nuclear group were whole-heartedly in favour of the use of nuclear energy, based on fringe benefits, impacts on society and economic progress, but after the accident they showed a favourable attitude towards the use of nuclear energy based on economic progress, but with some reservation because of the potential threats.

This calculation was conducted by Mr. Sen'ichi Mokuya in Master's Program of Science and Engineering, Tsukuba University, Japan.

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IRPA9
1996 International Congress on Radiation Protection
April 14-19,1996
Vienna, Austria

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PAPER TITLE

The HPS Position on Radon Health Risks

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MAJOR SCIENTIFIC TOPIC NUMBER 11. (see page 7)

ABSTRACT (See instructions overleaf)

The Health Physics Society first published a position statement on radon in October 1990. In mid-1995 that statement is undergoing revision by the HPS Radon Section to reflect the continuing controversy on radon health risks. In particular, a January 1995 study by the National Cancer Institute (the Missouri Study) concluded that no correlation was demonstrated between radon and lung cancer incidence. This study provided material for the news media to widely proclaim that radon was a huge, costly, government hoax.

In June 1995, another study by the National Cancer Institute demonstrated a statistically significant increased lung cancer risk at levels equivalent to a lifetime exposure at 4 pCi/L. This study also confirmed previous estimates of 15,000 lung cancer deaths a year in the United States due to radon exposures. Among these estimated deaths, 10,000 were among smokers and former smokers, while 5,000 were among never-smokers (a much greater proportion than previously estimated).

Some health physicists are saying the radon action level at 4 pCi/L should be increased to at least 8 to 10 pCi/L. Some are also saying that the linear non-threshold dose model is not applicable to radon health risks and that radiation hormesis may apply. At the same time, other health physicists point out that radon is not only the greatest source of radiation energy deposition in the human body, it is greater than all other sources combined on the average. They contend that if we are concerned with any risks from radiation, then radon has to be the major focus of that concern.

The HPS position will provide guidance that recognizes the controversy on radon health risks and recommend prudent actions in keeping with the HPS mission to assure public health and safety.

THE WHITE PAPER ON RADIATION PROTECTION AT ELECTRICITE DE FRANCE

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The general public is increasingly interested in questions concerning the environment, and people are particularly sensitive to the consequences arising from the operation of nuclear power plants. Radiation protection standards are becoming increasingly stringent: recommendations drafted in 1990 by ICRP (ICRP 60) will soon be applied on a general scale. In this context, EDF became aware of a deterioration in the results of dosimetry monitoring in its plants[1] (an increase in the collective dose per nuclear power unit from 1989). This encouraged the Company to undertake a critical analysis of the situation, and to comprehensively rethink all the questions linked to radiation protection. This work led to the publication of the White Paper on Radiation Protection in mid-1993.

The White Paper deals with various aspects:

- medical and biological fundamentals,
- worker protection,
- protection of the public and the environment,
- information provision and training,
- emergency situations.

The major principles to note are:

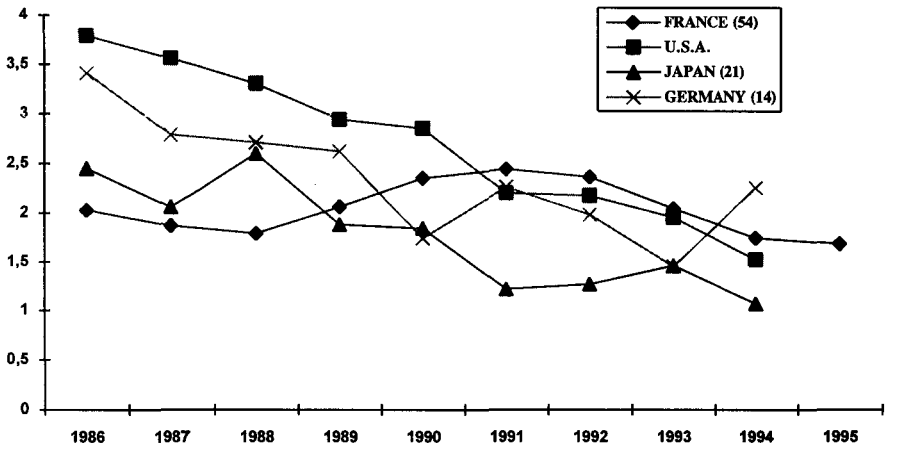
- The application of optimization techniques ("ALARA") to **work organization** is a key point in dose reduction; national and regional structures were set up to deal with this; and ambitious targets were established (35 % reduction of annual collective doses between 1992 and 1995).

Dose reduction started effectively; doses went from 2.36 man.Sv per unit in 1992 to 1.68 man.Sv in 1995.

[1] As of 01.01.1996, EDF operates 54 PWR nuclear power plant units of 900 and 1,300 MW.

Collective dose per unit

man.Sv



- Efforts have also been made **to reduce doses** by other methods:

  - design of facilities: study of new materials, better use of materials and premises to reduce exposure time during maintenance,
  - study of robotic implements,
  - study of new decontamination techniques, etc.

- Many provisions have been made to bring **radiation protection of contractors' workers** up to the same level as EDF workers. In particular this has involved:

  - developing an EDF system, called DOSINAT, to follow up dosimetry nationally using daily individual dosimetry data obtained from electronic dosimeters; this computerized system can follow up migrant workers who move from site to site, preserving dosimetry records, which were previously difficult if not impossible to obtain. An overall system for all French nuclear operators, DOSIMO, is being developed.

  - developing formal agreements with these companies, in particular to achieve a better division of work in nuclear and non-nuclear zones that takes account of dose limitations.

  - drafting training requirements both in radiation protection and on working in nuclear environments, formalized by the creation of an "access book". Training to be supplied by approved organizations (on the basis of training specifications and quality organization auditing).

- In the **environmental field**:

  - radioactive waste from nuclear power plants, already very low, will be brought to the level of the most efficient generating plants in this field. Note that going any further in this offers no advantage in health terms.

  - quantities of radioactive waste are gradually being reduced with the development of enhanced compacting, incineration and recycling techniques.

  - studies are underway on the elimination, or at least the reduction in volume, of long life waste.

- Improvement of overall radiation protection requires the general awareness of personnel ("radiation protection culture") as well as a redefinition and promotion of the "radiation protection skills".

- **Information provided to the public** is very important. EDF undertakes to supply everyone with clear and credible information, based on openness and honesty. But informing people well is difficult because of the variety of publics; which is why EDF has an information policy adapted to various public sectors which can act as select information channels, in particular:

  - health professions
  - media
  - teachers and students,
  - elected representatives.

Documentation for the various public categories is being updated.

- Finally, one of the White Paper conclusions is that, in the French context, **cooperation with Public Authorities should be developed**. The facilities to be set up are significant, and everybody should be able to make contributions while maintaining their respective roles. This concerns both the realization of a large national database, and the improvements in the company medical follow-up structures, which are currently too scattered.

  By mid-1995, 2 years from publication of the White Paper, the results obtained are very encouraging. Undoubtedly the most important thing is the awareness of the problems, and then dealing with them with the necessary energy and determination; that is what has been done and what augurs well for the future.

RISK COMPARISON AND COMMUNICATION IN RESEARCH AND POLICY

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INTRODUCTION

In the field of ionizing radiation quantitative risk analysis is a commonly accepted tool in assessing the acceptability of practices and applications of radioactivity. Harmonization has been greatly promoted by the efforts of the International Commission on Radiological Protection and other international bodies. In the last few decades, risk is also becoming one of the yardsticks in many areas of environmental policy in measuring both the acceptability of an activity and the effectiveness of specific measures. Risk analysis is thus developing into a major tool and risk assessment into one of the primary procedures with which standards are being developed and policies being set [1]. The increasing pressures of economic market forces and the globalization of the economy is urging organizations and countries to be cost-effective, competitive and also to define their position in environmental matters in relation to others. The importance and the broad scope of applications of risk analysis are illustrated by many studies (for a review see, for example, reference 1). The process of environmental risk analysis often follows a similar path, as illustrated by the diagram in Figure 1.

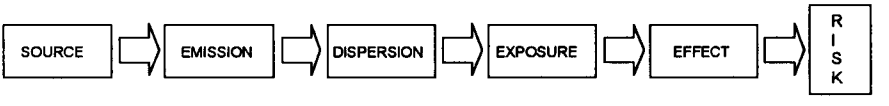

Figure 1. A diagram of environmental risk analysis processes.

Risk as an expression of the result of one's effort can be a binding quantity. However, as risk analysis and assessment have been developing in different scientific disciplines and on the borderline of science and policy, a lot of divergence in terminology and methodology has entered into the field. In practice, this leads to considerable confusion to the extent that assessment sometimes is used while meaning analysis. Such confusion weakens the possibility of defending any resulting policy decision in the political arena, defying the original purpose of the risk analysis exercise. Simply expressed, the more widely risk analysis is accepted as a tool of policy support, the less widely the results of such risk analyses may be accepted by politicians and the general public in defining their political choices.

PROBLEMS

In the development of the techniques of risk analysis many choices as to the definitions of key elements in the analysis had to be made. Consider the definition of the risk target in most studies: the human being. In some cases this target is defined as a "spinach-eating infant", in others one takes a human to be a 20-year-old well-trained marine. Sometimes emergency-response actions are considered to be immediate and effective, sometimes they are supposed to be too little too late. Comparable variations exist with regard to almost all parameters being used in risk analyses in human and ecotoxicology and in chemical safety studies. These observations are not new, and are in fact well documented in several papers and publications (e.g. 1, 2, 3). However, they have led to surprisingly few changes in day-to-day applications of risk analysis.

In various fields of environmental policy, and in some countries even within the same field of environmental policy, differing assumptions have found their way in the risk analysis process and subsequently in policy development. In the field of major chemical hazards, in the United Kingdom the individual risk is expressed as the frequency of a protected individual receiving a dangerous dose, whereas in the Netherlands it is expressed as the frequency of an unprotected individual receiving a

[1] Many definitions of risk exist. We mean by *risk* the probability, or frequency, of occurrence of a particular effect (adverse event); by *risk analysis* the process of obtaining a numerical outcome for this risk; by *risk assessment* the valuation of the calculated risk.

fatal dose. The different numerical outcomes of analyses based on such widely different presuppositions create profound difficulties in policy-making and negotiations with the parties involved, such as industries. Recently, in a report on problems associated with risk analysis and environmental risk management (4), a committee of the Health Council of the Netherlands gave a general indication of the problems that may arise from confusing terminology.

Even more complications are met in risk communication, in explaining the results of risk studies to - often non-expert - politicians and, most importantly, to the public. The public may be well aware that it functions as potential victim in risk studies, and therefore claim a position in discussions on risk outcomes in its role as the primary driving force behind the (democratic) policy process. Unclear statements and diffuse terms may influence the debate on the acceptability of risks, and act as a trigger for far-reaching action with possibly unpredictable results. The situation is even more aggravated by virtue of the fact that interested parties may want to introduce definitions to steer the outcome in a desired direction. For instance, the comparison between driving by passenger car and flying with a commercial airline will have a markedly different result if compared on the basis of numbers of casualties per calendar year, per passenger-hour or -mile.

COMPARABILITY OF RESULTS

Nevertheless, the possibility of comparing the results of analyses and the policy indications is the original idea behind the development of structured risk analysis techniques and should remain so in the future. Any difference in the results of decision-making between countries or fields of policy should be the result of a conscious weighting of risks against profits and not the accidental result of unknown or unexpressed differences in definitions, techniques or assumptions. Adequate risk comparison is related to making the criteria for risk limitation explicit, for instance "individual vs. collective risk", "fatalities vs. morbidity", "regular vs. potential emissions or explosions" etc. When this explicitation is lacking, discussions on risk analysis and assessment may lead to lengthy and fruitless sessions.

EXAMPLES

The environmental risk policy debate in the Netherlands forms a typical example for the developments described. Originally, risk policy was developed for petrochemical installations. A maximum risk exposure was defined as the chance per year of an unprotected individual being killed as a result of an accident (5). The limit was set to 10^{-6} per year per installation. Subsequently, the same limit was declared applicable for all possible confrontations with risks induced by exposure to human-induced hazards, including chemicals in the environment, accidental releases of nuclear power plants and applications of ionizing radiation in general. This creates some interesting problems.

Firstly, exposure to a chemical disaster - which was the original subject of risk policy - has immediate lethal effects stemming from intense thermal radiation, explosive effects and high dosages of toxic chemicals. Delayed effects are rare, to the extent that they are almost never considered. An exception may be the long-term effects of exposure to accidentally emitted chemicals such as dioxins (Seveso). These effects, however, are never included in "normal" risk analyses for chemical plants.

On the other hand, when assessing the environmental aspects of possible applications of radioactive substances and radiation, the main effects of ionizing radiation taken into consideration nowadays are delayed effects, with the possible exception of the planning of new nuclear power plants. Assumptions on these delayed effects involve assumptions on the duration and pathways of exposure. Often complicated processes and pathways of environmental dispersion related to inhalation and ingestion of contaminated food are critical in the risk calculations.

Furthermore, risk analyses for the collective exposure of populations to ionizing radiation are based on assumptions related to the age distribution of the population, food intake etc. The individual population risk is obtained by dividing the collective risk by the total population. This approach is hardly or not at all comparable with the assumption of the "unprotected individual" as used in the field of chemical safety. Nevertheless the resulting figures are both designated individual risk and as such are taken to be directly comparable.

The same applies when this "individual risk" is being compared with other risk sources, such as the aforementioned driving or flying. Yet, many publications do present seemingly unambiguous lists of risks, usually trying to prove the apparent innocence of some source of risk.

WHAT SHOULD BE DONE?

There is a clear and urgent need for clarification and, if possible, harmonization of terms, procedures and assumptions, driven by the continued difficulties that arise in making policies and explaining them to the public. Several efforts have been made, or are still in progress, to at least clarify the terms of reference in the various fields. The efforts of the ICRP in radiation protection have already been mentioned. A recent document of the Health and Safety Executive in the United Kingdom (6) is an example of an attempt in the field of industrial safety to clarify several important aspects. A project under the auspices of the expert group on chemical accidents of the OECD is under way. However, to our knowledge all these activities are limited to only one of the fields previously mentioned, whereas in the opinion of policy-makers and the general public clear interactions exist.

What is even more striking is that, with the exception of ICRP, all attempts at clarification and harmonization originate from the policy scene. The participation of the scientific and technical community is limited. However, scientists are the originators of information forming the basis for policy decisions. It is therefore vital for continued or reinstated trust in these analyses that scientists be unambiguous about what they mean when presenting a number as the result of an analysis. For the further development of the field of risk analysis it would be of great importance if scientists and experts came together to clarify and organize their doings in a more interdisciplinary way.

A simple academic exchange of ideas and viewpoints alone does not seem to suffice. This year a project was started at the National Institute of Public Health and the Environment (RIVM) to present different types of risks geographical and coherently. In this so-called CUMU-project geographical mapping of risks and the use of GIS technology is proving to be a way to a useful organisation of discussions between experts on risk from different disciplines. Mapping risks forces the participants to take the whole chain, from source to risk, into consideration, and to reflect on their own premisses and axioms. During the mapping process a confrontation with the more philosophical aspects of risk comparability is almost inevitable. However, such discussions take place in the positive atmosphere of working together towards a collectively produced "risk map". In this way, making the relevant positions explicit is more fruitful than in comparing risk approaches as such without a common goal. We feel that this approach may be of a wider significance than for this project only.

To facilitate further discussion, RIVM plans to organize a conference on risk-map technology and risk comparison in the spring of 1997. In our opinion, active participation in activities aimed at crossing borders between the different fields of "risk" and contributing to more broadly coherent ways of risk assessment and risk communication could be of great interest to the radiation protection community.

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HEALTH PHYSICS EDUCATION AND TRAINING IN IRAN

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ABSTRACT

Health physics education and training (HPET) are close counterparts for an effective enforcement of radiation protection (RP) regulations and development of an advanced RP infrastructure in a country. The related history in Iran dates back to over 30 years ago advancing towards promotion of a "Sustainable Training Program" (STP) through programs such as academic courses, intensive courses, research, on-the-job training and media training. The STP has been effective in development of an advanced national infrastructure for effective enforcement of regulations in different applications and provision of self-sustained national services. In this paper, the elements of a long-term national STP are discussed with a hope it could act as a model in developing countries.

INTRODUCTION

The history, status (1) and infrastructure of radiation protection (RP) in Iran as well as on health physics education and training (HPET) up to 1990 have been reported before (2). As regards to HPET, this author believes that they are powerful counterparts for enforcement of regulations and they should be implemented far before promulgation of a law (3,4). This is the case especially in Iran where RP regulations have been implemented based on international regulations even far before promulgation of the AEOI Act of Iran (3) and the RP Act of Iran (4). The author also believes that although international agencies could support RP manpower training of developing countries, experiences in Iran shows that developing countries should develop a "Sustainable Training Program" (STP) through academic courses, intensive courses, research, on-the-job training and media training. The elements of STP as implemented in Iran are discussed in this paper with a hope they could also be applied as a model in developing countries.

1. ACADEMIC TRAINING PROGRAMS

The university academic degree programs can be a major part of a STP in a country for development of qualified manpower and radiation protection infrastructure. Early in 1960's, the first formal courses related to radiation protection offered in the MSc program in physics of the University of Tehran (UT) including three credit hour courses on health physics, nuclear physics, reactor physics, cosmic ray physics, radiation detection, nuclear electronics, etc. Further potential for specialization in health physics has been through MSc theses in health physics from different universities (e.g. see ref. 5 and other papers of the author at this congress). Also, a MSc degree program was organized by the Health Faculty of the UT in cooperation with the World Health Organization (WHO) twice during 1973 to 1975. In 1977, the NRPD at AEOI also organized a MSc equivalent program through which about 30 health physicists have been trained. The graduates of this program and many others trained are now serving as health physics officers at the AEOI and in many other institutions. At present, courses in health physics, nuclear physics and radiation detection are being offered at BSc level in almost all technical universities in Iran. Since 1987, MSc and PhD programs in nuclear science and technology with options in Health Physics and in Radiation Medicine Engineering have been established at the Amirkabeer Technical University in Tehran in cooperation with the AEOI. A MSc program in Nuclear

Science and Technology has also been established around 1988 at the Sherif Technical University in Tehran with options in Health Physics and in Medical Radiation Engineering. Some MSc programs in Medical Physics are also being offered in some medical universities. All these programs will assist in manpower development for a strong national infrastructure on radiation protection.

2. INTENSIVE COURSE PROGRAMS

Intensive short courses have been a major component of development of RP infrastructure in Iran for over thirty years and in particular within the last decade. A six weeks course on "Radioisotope Applications" has been offered many times since 1960 for safe peaceful applications of radioactive materials. Since then, many intensive courses related to health physics have been conducted. In 1985, to establish a strong base for implementation of regulations of RP Act of Iran to be promulgated (4), a STP by provision of intensive courses was established for workers in medicine, industry, research and education. The courses consist of 100 hours of theory and practice in four weeks. The courses are mandatory to be taken before starting any radiation work of potential health hazard such as for example in industrial radiography. Up to now, 44 intensive courses have been organized with about 2300 participants. The courses usually have one preliminary exam at the beginning, one at the end of each week and one exam two weeks after the termination of the course. About 50% of the participants usually pass the course and receive a certificate. Those failed in the final exam have the chance to take it once again; if not successful, the course should be taken again.

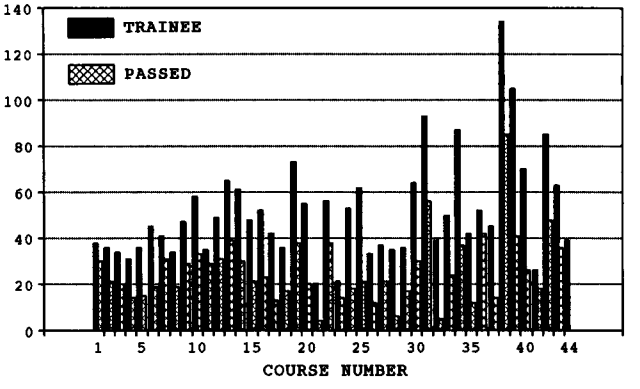

Fig. 1. Number of examinees (solid bar) and those passed the exams (dotted bar).

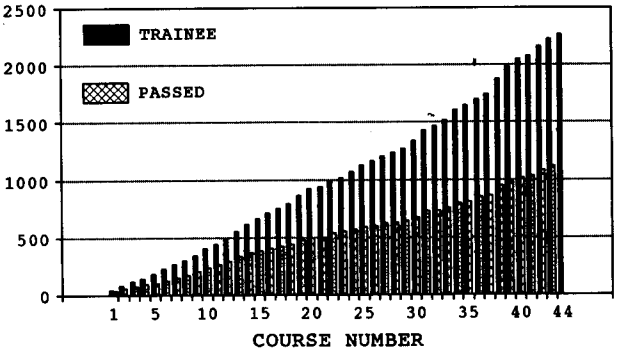

Fig. 2. Cumulative number of participants and those passed the exams.

Figure 1 shows the number of participants taking the final exam for the first or second time (solid bar) and those passed the exam and received a certificate (dotted bar). Figure 2 shows the cumulative bar chart of the number of participants (solid bars) and those passed the exam and received a certificate (dotted bars). This author believes that even those failed, just by their presence in the lectures have gained enough knowledge and experience to protect themselves and the others.
Also, regional and interregional IAEA courses have also been organized; e.g. a two-weeks interregional course on "Handling of Radiation Accidents" organized by TUNC in 1967 and a four-weeks regional training course on "Notification, Registration, Licensing and Control of Radiation Sources and Installations" again sponsored by IAEA and organized by the NRPD in August 1995.

3. RESEARCH TRAINING PROGRAMS

Research programs in health physics through MSc and PhD theses are also an important component of a STP. For example, so far as the NRPD alone is concerned, about 50 MSc and PhD theses have been conducted under the supervision of this author on topics related to radiation detection and dosimetry development and measurements using SSNTDs, TLDs, gas detectors, etc.; determination of genetically significant dose (GSD) in Iran; quality assurance in radiotherapy, diagnostic radiology, mammography, CT scan, nuclear medicine, etc.; radon dosimetry; studies in high level natural radiation areas; non-ionizing radiation, etc. Such programs are also conducted in medical and technical universities and some other research institutions. BY these programs, the country is relatively self-sufficient for development of health physics manpower bearing in mind that always international exchange of information and cooperations are vital.

4. ON-THE-JOB TRAINING PROGRAMS

In parallel to the manpower development given above, on-the-job training of staff and radiation workers have been advanced at the NRPD and other institutions such as in hospitals and through fellowship programs of the IAEA. This is also a major counterpart of development of manpower. The NRPD has also been the host country for many IAEA fellows to have special training at the NRPD.

5. MEDIA TRAINING PROGRAMS

The media training programs have been carried out by TV, radio, journal articles, brochures, posters, etc. This has been in particular very effective for control of public exposures as regards to medicine and fetus exposure control. Such topical programs are occasionally carried out for public awareness to an extent not causing radiophobia.

CONCLUSION

The above programs have been very effective for efficient implementation of radiation protection regulations in the country. Based on experiences gained, the STP is the only apporach to meet the minimum radiation protection requirements in a country and to promote an advanced program.

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Remember When Science Was Fun? Encountering "Nuclear Fallout In Your Wood Stove" and Other Mysteries at the Northwestern New Mexico Regional and State Science and Engineering Fairs

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INTRODUCTION

The Rio Grande Chapter of the Health Physics Society is a proud supporter of the Northwestern New Mexico Regional and State Science and Engineering Fairs. In this role, the Chapter provides judges and furnishes monetary awards to recognize those students, between grades 6-12, and their teachers whose projects include the utilization or investigation of ionizing (e.g., gamma) or non-ionizing (e.g., UV exposure, microwaves) radiation. The Chapter promotes public information and education about health physics by sending every award winner and sponsoring teacher a copy of *Career Opportunities in Health Physics*, including information about degree programs and scholarships. Also, the Chapter provides a 1-year free subscription to the *Rio Grande Chapter Newsletter*, and publishes the names of the award winners, the titles of their projects, the names of their teachers, and the names of their schools. Furthermore, Chapter members are encouraged to assist contestants and award winners by providing mentoring opportunities, and educational resources such as textbooks. This paper reviews the Rio Grande Chapter Science and Engineering Fair Program with respect to judging categories and criteria, project titles, what the Chapter has learned from the students, and an overview of the 1995 Regional, State, and International Science and Engineering Fair Programs.

JUDGING CATEGORIES AND CRITERIA

The students may enter their projects in one of the following categories:

Behavioral and Social Sciences	Environmental Science	Botany
Mathematics	Biochemistry (Senior Division Only)	Medicine and Health
Chemistry	Microbiology	Computer Science
Physics	Earth and Space Sciences	Zoology
Engineering	Team Projects	

The judging criteria issued to the students by the International Science Fair Program provides guidance regarding the display, use of materials, and safety. Nevertheless, the primary responsibility of the judge is to evaluate the collection and interpretation of the data, and only secondarily, to scrutinize the physical display of the project. The actual judging criteria, the percentage contributing to the student's score, including questions and considerations during the evaluation process are as follows: **Scientific Thought/Engineering Goals (30%)** - A) Does the student understand the fundamental scientific principles involved with the project? B) Are the conclusions formulated clearly and follow the presentation? C) Does the student recognize sources of error or alternative interpretations? **Thoroughness (15%)** - A) Does the project show care and dedication, or was it thrown together hastily? B) Does the quality of the project show consistency with the stated objective? **Skill (15%)** - A) Does the project reflect the ability of the student? B) The student is not penalized for lack of craftsmanship or experience in working with materials, since many students may have highly developed skills in certain areas. C) Reservations about the ability of the student to complete the type of project on display can be evaluated during the interview. **Creative Ability (30%)** - A) Does the student present a fresh approach to a problem, or merely copied an experiment already documented in the peer-reviewed literature? B) Does the student make ingenious use of materials and/or equipment? C) Does the student formulate the problem to obtain meaningful answers that are within the intellectual capacity, training, and means of a middle school or high school student? **Exposition (10%)** - A) Proper exposition of the information using graphs, illustrations, and the written report will convey the main ideas and results after a few minutes of observation. **Teamwork (Team Projects Only)** - A) Does the project identify the tasks and contributions of each team member? B) Does each team member appear to be fully involved, and familiar with all aspects of the project? C) Does the final exhibit reflect the coordinated efforts of all team members?

Perhaps the most educational and rewarding aspect of the science and engineering fair, for both participants and judges, is the interview. The interview allows the judge an opportunity to discuss one-on-one with the student the source of ideas, the basic scientific principles and methods used to develop the project, and significance of the conclusions. During the interview, the judge can also suggest methods to improve collection and interpretation of scientific data. In addition, the judge can notify colleagues at nearby laboratories and universities to either donate equipment to the schools directly or make equipment and facilities available to conduct additional experiments. The access to advanced equipment provides an opportunity for the student to collect supplemental data to test their hypotheses using hands-on experimental techniques.

PROJECT TITLES

A sampling of the project titles from the Regional and State Science and Engineering Fairs are as follows:

Radiation Roundup (Natural and Man-Made Sources).
Does Radiation Affect Seed Fertility?
WIPP - How Does it Affect You?
How Radioactivity Levels in Soil Differ Across the North-Central Region of New Mexico.
Radon vs. Home Construction.
Nuclear Energy (Observing Radioactivity).
Effects of UV-B Radiation on the Biosynthesis of Amino Acids in Plants.
Nuclear Fallout in Your Wood Stove.
Does an Electromagnetic Field Affect the Germination and Growth of Plants?
Can Microwaves Kill Bacteria on a Contact Lens?
Does an Electromagnetic Field Affect the Rate of Mutation in *Drosophila melanogaster*?
Making Water Safe: UV the Culprit Becomes UV the Hero.
What are the Effects of Microwave Radiation on Bean Seeds?
Zap It - Microwaves vs. Bacteria.
What are the Effects of UV-B Radiation Upon the Biosynthesis of Amino Acids in Plants and Upon Plant Growth?
Particle Acceleration.
Light Wavelength and Fungal Growth.
Does Sunscreen Really Block Ultraviolet Light?
The Effects of UV Light on the Chlorophyll of *Isochrysis galbana*.

WHAT THE CHAPTER HAS LEARNED FROM THE STUDENTS

After interviewing teachers and students, the representatives from the Chapter have discovered that the immediate availability of equipment determines the scope of the project. Typically, the students will conduct experiments monitoring the effects of plants, etc. following exposure to ultraviolet light, microwaves, and electromagnetic fields (i.e., non-ionizing radiation). Usually, students whose parents are employed in the health physics profession (e.g., X-ray technician) will conduct their experiments using ionizing radiation.

THE 1995 REGIONAL, STATE, AND INTERNATIONAL SCIENCE AND ENGINEERING FAIR PROGRAMS

The following information provides an overview of the 1995 Regional, State, and International Science and Engineering Fair Programs.

Northwestern New Mexico (NWNM) Regional Science and Engineering Fair Program

- The NWNM Regional Science and Engineering Fair takes place every March.
- The NWNM region includes approximately 70 schools.
- 737 (162 senior and 575 junior) students participated in 1995.
- 110 students qualified to compete at the State of New Mexico Science and Engineering Fair.
- 2 students and 2 alternates qualified for the International Science and Engineering Fair held in Hamilton, Ontario, Canada.
- The Program Committee hosts a teacher workshop every August to evaluate suggestions for enhancing the science fair program. A sampling of enhancements have included, but are not limited to the following:
    - A) Informing new middle school and high school science teachers about the NWNM Science and Engineering Fair Program, and establishing a database for future correspondence.

B) Generating new ideas for future science fairs (e.g., team projects).
C) Involving corporate and organizational displays.

- The Mentorship Program provides professional contacts with experts, and produces national champions at the International Science and Engineering Fair.

The 43rd Annual State of New Mexico Science and Engineering Fair Program

- The State of New Mexico Science and Engineering Fair takes place every April.
- The State of New Mexico Science and Engineering Fair includes 6 regions.
- 325 total (136 senior and 189 junior, 12 senior and 14 junior team projects) students participated in 1995.
- Winners from the senior division proceed to the International Science and Engineering Fair.

The International Science and Engineering Fair Program

- The International Science and Engineering Fair represents the United States, U.S. Territories, and 29 International Affiliated Science Fairs throughout the world.
- 1019 students participated in 1995.
- Primary communication is with the Fair Directors of the various Regional, State, and International Affiliated Science Fairs.
- The 47th International Science and Engineering Fair will take place in Tucson, Arizona, May 5-11, 1996.

CONCLUSION

The Regional, State, and International Science and Engineering Fair Programs provide tremendous opportunities for students and teachers to meet and interact with professionals from the field of health physics. Although the professionals possess strong backgrounds in the application and control of ionizing radiation, most students conduct experiments using non-ionizing radiation because of the availability of generating sources. Based on personal interviews, students and teachers will exclude the use of ionizing radiation because of the perceived dangers, or the lack of access to specialized equipment. Nevertheless, the Rio Grande Chapter is notifying teachers and students about the availability of resources (e.g., radioactive sources, detection and measurement equipment, neutron activation analysis) at nearby laboratories and universities to perform experiments using ionizing radiation. The information contained herein furnishes valuable insight to enhance future science fair programs, and serves as a guide for those individuals wishing to increase their involvement with these and similar programs (e.g., mentoring) that are available to students wishing to explore career opportunities in health physics.

TRAINING COURSE FOR BORDER GUARDS ORGANIZED BY THE INSTITUTE OF ATOMIC ENERGY IN ŚWIERK, POLAND

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INTRODUCTION

Danger of illegal trade in radioactive and fissile material has recently increased due to disintegration of the former Soviet Union. A substantial part of these materials is suspected to be smuggled through Poland to Western Europe. Proper countermeasures like establishing radiation detecting gates at border crossings and specific training programs for border guards have been set up by Polish authorities.
On request of Polish Border Guard Command the Institute of Atomic Energy (IAE) has prepared a series of training courses for border guard officers. The courses covered both theoretical and practical subjects concerned with radiation safety and were focused on detection and safeguarding of radioactive or fissile material at border crossings.

THE PURPOSE OF THE COURSE

The main subject areas covered by the course included: transport regulations for radioactive materials, radiation protection safety procedures, operation of radiation survey meters and physical properties of fissile materials. Since border guard officers could not spend much time off their posts, the duration of the course had to be limited to three days of intensive training. Emphasis was laid on developing practical skills in detecting and safeguarding of nuclear material. The practical part covered also hands on experience in measurement of radiation, decontamination of areas and persons, and setting up radiation safety zones.

The syllabus of the course was evaluated and approved by the Polish Atomic Energy Agency. A handbook covering the course material was edited in Radiation Protection Division of IAE and published by the Institute's Publishing Division. The handbook contained texts of all lectures and technical parameters of radiation monitors used during the course. There was also included a complete set of example documents accompanying isotopic sources in transport and their marking.

SYLLABUS OF THE COURSE

1. Elementary information on ionizing radiation

- types of radiation and their properties (alpha, beta; range, penetration power)
- natural and man-made radiation sources (exposure of the population to these sources)
- basic concepts and quantities (activity, Radioactive Decay Law, half-life)
- types of radiation sources (radioisotope sources opened and sealed, radiation generators)

Demonstration: basic properties of alpha, beta and gamma radiation.
The aim was to distinguish different types of radiation and act accordingly in protecting yourself and the general public when the source is found. To know what activity means and how it is changing with time.

2. Exposure to ionizing radiation

- biological effects of radiation (health hazard resulting from exposure to ionizing radiation)
- internal and external exposure and methods to limit these
- dose limits for occupationally exposed persons and for members of general public
- characteristics of potential exposure at border crossings

Demonstration: penetrating power of alpha, beta and gamma radiation; shielding.
The aim of this lecture was to make trainees aware of the danger of exposure to ionizing radiation. They were also practicing with the role of distance, time and shielding in limiting external exposure with numerical examples. Past incidents with radiation sources on border crossings were called on and evaluated as examples.

3. Assessment of internal and external exposure

- measurement of external exposure (dose rate and contamination measurements, personal dose)
- measurement of internal (whole body counter, thyroid activity counter, activity of excreta)
- basic rules of work in contaminated areas (radiation safety zone, decontamination procedure)

Each trainee was asked to perform decontamination of himself and of a given area. Direct and indirect (smear test) measurements of contamination were performed using alpha and beta calibration sources.

4. Operation of dosimetric equipment

- radiation monitors different types
- contamination monitors for measurement of alpha, beta and gamma contamination
- radiation detection gate (type installed at border crossings)

The trainees were operating monitors under conditions of a simulated radiation accident. They were also taught how to prepare and check the monitor before starting measurement.

5. Isotope transport regulations

- dose limits and transport regulations for radioactive materials (according to international standards)
- transport of radioactive materials (containers, marking, surface dose rate limits, documentation)
- regulations concerning vehicles for transport of radioactive materials

Demonstration: containers, packets and their marking

6. Practical training and demonstrations

- practical operation of radiation measuring devices
- detection and safeguarding of radiation source (use of manipulators, containers and shielding)
- demonstration of different radiation source packing and transport documentation
- demonstration of specialized laboratories of IAE (whole body counter, thyroid activity counter, spectrometric laboratory)
- a visit to fissile material store (uranium of different shapes, plutonium containers and fuel elemnets)

CONCLUSIONS

Two series of courses were organized and altogether 300 border guard officers were trained in 1994. Each course was concluded with a theoretical and practical examination. The practical part consisted of a simulated radiation accident arranged with low activity calibration sources. The trainee was asked to resolve the situation according to the rules. All important elements from the practical part of the syllabus were involved in the examination together with relevant regulations.
Each trainee was given a copy of the handbook for future reference.

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THE ROLE OF INFORMATION, EDUCATION AND TRAINING IN REDUCING DOSES

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SUMMARY

During the early years of EDF's nuclear reactor construction programme, there was intensive initial training aimed at informing new generations of workers of the work they were going to doing. This ambitious objective, aimed at putting an increasing number of units into service in total safety and at rates of up to 6 to 8 units per year, was a success.

The mass training programme which centralised and concentrated on knowledge of the process and procedures, was ideally suited to the requirements of a standard population of reactors and to the various trades concerned, which were less varied and numerous than those of today. The present situation is slightly different and training should be directed towards the experience of operations, retrofits and knowledge of new technologies.

Accordingly, we have moved towards a more individual style of management for a greater body of knowledge.

The population concerned consists of some 50 000 workers. This document sets out the policy implemented by EDF:

- mobilisation of EDF managers and contractors,
- integrating the ALARA concept into specific operations,
- reducing the amount of information.

Today, the ALARA concept represents an excellent opportunity for making progress. Man, his knowledge, his experience and initiative, represents the most important source for progress in the field of radiological protection. In order to be able to use his creative potential to the full, man must be imbibed with both a radiological protection culture and a safety culture. This must be the main priority of nuclear operators.

FOREWORD

In order to provide quality electricity generation at the lowest possible cost and under the best possible conditions of security and safety within its installations, EDF must:

- develop the professionalism of the people concerned through operation itself (EDF personnel) and maintenance operations (EDF and contractor personnel);

- improve and maintain the safety and quality culture;

- improve the quality of relationships between those involved in maintenance by enabling them to become part of a close partnership approach.

EDF's DOSIMETRY POLICY

Since 1991, EDF has committed itself to a policy of optimising collective dosimetry and this has already resulted in a reduction of approximately one third over five years. This will to succeed is being maintained to make EDF one of the best nuclear operators in the world.

In so far as personal dosimetry is concerned, the objective of not more than 20 mSv for each worker should be achieved before the year 2000.

For this, it was decided to implement systematically the concept of the optimisation of dosimetry (ALARA). The various aspects which influence dosimetry (training, operation, maintenance, experience feedback, information processing system, etc.) are described in terms of progress.

INFORMATION AND TRAINING ON THE OPTIMISATION OF DOSIMETRY

An initial analysis of the reasons for high dosimetry has shown that significant improvements can be made by basing information on an extremely simple concept, the ALARA concept: defining dosimetric targets during operational preparation, measuring performances through site monitoring, analysis of discrepancies (and, if necessary, corrective actions) and obtaining feedback of experience after completion.

It was decided to embark on a campaign of awareness for EDF managers and for contractors. The performance and durability of radiological protection depend first on the total commitment of senior management followed by the rest of the employees. Getting men to reflect carefully before taking decisions, rejecting improvisation, adopting a systematic approach to solving problems, identifying and controlling doses is not a natural process and only senior management can initiate and develop this style of management in accordance with the requirements and specific nature of the company (heat lagging, auxiliaries, valves and fittings).

The second stage could be called "communications in the field". Communications with workers is the responsibility of both the contractor and of EDF personnel (training on arrival on site, ALARA posters, displaying dosimetry targets and performance for unit shutdowns, presence of an ALARA team on the site, etc.).

The sites with the worst records were analysed according to the ALARA concept and dosimetry targets are fixed together by the contractor and EDF when orders are placed. For certain operations, models are used to train operators.

Extension towards more detailed analysis, through collective dosimetry targets, is being considered for 1996.

In so far as prototype sites are concerned, EDF operates a specialised centre, the CETIC, where it is possible to develop certain operations (actions and positions of workers, tool modifications, etc.).

CONCLUSION

EDF, together with its contractors, should return to being one of the leading nuclear operators in terms of radiological protection. The required optimisation of dosimetry will be achieved through the awareness of all those concerned (senior managers, managers and other workers), training in the ALARA concept, increased professionalism including careful preparation, monitoring of operations (dosimetry and quality), training on models, if necessary, and experience feedback from sites.

All this, of course, goes to create the radiological protection and safety culture of everybody concerned within the nuclear industry.

SYSTEM OF REGIONAL CENTRES OF FIRST AID IN CASES OF RADIATION ACCIDENTS IN GERMANY

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When in the seventies the number of occupational radiation exposed persons in the Federal Republic of Germany increased from about 35,000 (1974) to about 160,000 (1978) the Industrial Injuries Insurance Institutes felt prompted to reflect about special measures to prevent radiation accidents and provide health care for this special cases. They did so without any actual occasion: accidents were persons have been exposed by ionizing radiation were in the seventies just as rare as today. But that fact does not allow the Industrial Injuries Insurance Institutes to neglect the existing potential for severe accidents.

So the Industrial Injuries Insurance Institute for the Electrical Industry including Precision Mechanics and the Industrial Injuries Insurance Institute for the Chemical Industry created the Institute for Radiation Protection in 1978. The primary task of that Institute is to guarantee an effective first aid in the case of a radiation accident.

To realize that task the Institute contracted 11 wellknwon institutions like radiological departments of large hospitals or the medical departments of research centres where the knowledge on diagnostic and therapy of radiation effects is present. They are called „Regionale Strahlenschutzzentren", Regional Centres for Radiation Protection (RCRP) (Fig. 1).

In the case of radiation accidents these RCRP are the logistical centres for all arising questions of treatment. They have facilities for reconstructing exposure situations and assessing and evaluating doses, including measurements of internal contamination, as well as for medical in-patient or out-patient treating like internal or external decontamination.

Another important task of the RCRP is to advise employers in all radiation protection questions which arise with the industrial application of ionizing radiation. Of course the centres give also answer to many question from members of the public, for example the personal effects of the power plant accident at Chernobyl.

The Institute for Radiation Protection offers regularly training courses and lectures to the stuff of the RCRP. That is necessary because radiation accidents are seldom in the Federal Republic of Germany and therefore the stuff can not keep practical experiences. For that reason the offered training courses impart practical training as well as theoretical foundations. Confronting the participants with hypothetical radiation accidents is basic part of the courses.

These training courses also provide the contact between the stuff of the RCRP among each other and to a team of excellent experts on the field of radiation protection and radiation protection medicine. These experts can be contacted from the RCRP in the case of a radiation accident. Additionally the Institute invites representatives of the state authorities and of the

industrial branches where ionizing radiation is applied. In the last 13 years about 500 persons have taken part in the training courses.

Additionally to the training courses the Institute supports the RCRP by sending information about actual news in the fields of radiation protection and radiation protection medicine.

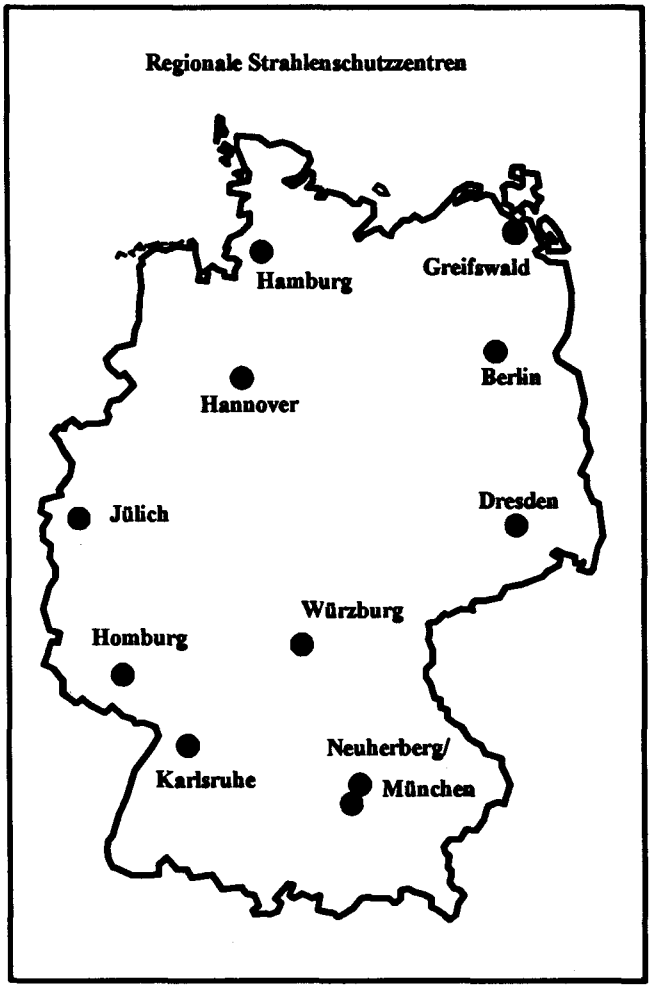

Figure 1. The Regional Centres for Radiation Protection in the Federal Republic of Germany.

THE CONCEPT OF RADIOECOLOGY EDUCATION IN BELARUS

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1. INTRODUCTION

The lack of adequate radioecological education and hence, 'radiation ignorance' is almost notorious for the whole world. During the last decade the world power producing companies have issued a great number of colourful booklets and information materials designed for totally unprepared public. They undoubtedly contribute to advertising the atomic energy but do not provide useful radiation related information as mostly comprise general knowledge on e.g. the atom structure or NPP operation principles covered in a very much elementary if not primitive way.

Meanwhile the atomic energy is becoming a substantial element of the environment and calling for much more serious and adequate information. In this context the population of Belarus as the most suffered from the Chernobyl accident with the anthropogenious radioactivity continue affecting almost all the spheres of man's activity must know about radiation no less then about electricity.

Had the people of Belarus been duly informed and at least initially educated in the elementary methods of individual radiation protection and agriculture production control, had they been aware in psychological peculiarities of life in the contaminated territories along with the sufficient number of specialists with the expertise required for working out the recommendations for those who were to make decisions, the consequences of the Chernobyl disaster, including social, psychological and medical aspects, could have appeared not so heavy; the efficiency of protective measures undertaken at that time might also have been much higher then that had been observed. The necessity of general and obligative radioecological education follows from the long-term character of Chernobyl disaster consequences, geographical position of Belarus (some nuclear power plants (NPPs) of other countries are operating now near its boarders), the claim of the strict radiation control in all branches of economy including export and import.

A certain work to develop the system of radioecological education in Belarus was carried out. According to decision of the Ministry of Education and Science in 1989 the radiation safety disciplines were included in the academic plans of all secondary and higher schools. In 1992 the International Sakharov College of Radioecology (now it is the Institute, ISIR) was opened in Minsk and the training of the specialists with higher education in radioecology has been begun. In some institutes like Belarusian Agricultural Academy special radioecological chairs are opened. The training and retraining of the specialists for radiation control network and activities is targeted to mitigate the consequences of the Chernobyl accident and it is supervised by the Training Informing and Radioecological Centre of the Management Academy of the President of Belarus.

At the time being the text books on radiation protection, unique as they are as there are no analogous in the world, have already been published and in mass edition in Belarus.

Nevertheless, there has been neither a unified concept of radioecological education developed yet nor the integral educational system purposefully designed for teaching radioecological disciplines at various educational levels. The latter is of special concern and importance as the social tension is being induced by the lack of knowledge in behaviour of the radioactive materials and substances exercised by all age groups of population living in the contaminated territories. The problem is even more aggravated by the fact that before the Chernobyl accident the radiation culture, i.e. standard

knowledge on radiation protection did not exist at all. Consequently, the decisions on elimination the consequence of the Chernobyl accident made at the amateur level were inefficient and regretfully still remain so.

It has led to the following objective discrepancies arisen, developed and/or become profound between

- necessity to provide vital activity of the Belarusian population within complicated radioecological conditions and the level of the population's awareness in questions of radiation safety;
- necessity of psychological rehabilitation of Belarusian population and the absence of corresponding programmes of radioecological education and enlightenment;
- nessecity of qualified and efficient managing the mitigation of the Chernobyl accident consequences and the state of the radioecological knowledge among decision-markers of all levels;
- nessecity of wide range employment of ionising radiation in different human activities and unsatisfactory knowledge of its features among workers;

The concept proposed below is worked out in correspondence to the more general "Conception of protection measures in reconstructing period for population resided at the Belarus territory suffered by radioactive contamination because of the Chernobyl disaster" adopted by Cabinet of Ministers of Belarus. It is based at the current view of the Chernobyl problems in reconstruction period and on experience of the world science.

2. BASIC STATEMENTS OF THE CONCEPT

2.1. The radioecological education is the intermediate part of the Belarus national educational system and is targeted to form a current knowledge for adequate understanding and apprehending of wide range of problems connected with radiation impact on the environment and a human being, employment of ionising radiation in national economy and alleviation of consequences of radiation incidents, accidents and disasters.

2.2. The radioecological education should be spread over the whole adult population. It is suggested as for citizens educated at traditional educational levels (secondary, special secondary and higher levels of education, the network of training and retraining specialists) but for another part of population through their enlightenment and informing on questions of radiation safety and hygiene.

2.3. The content of the radioecological education at different levels is formed by taking into account the basics of scientific knowledge just formed among students and is the same for definite categories of students .

2.4. In secondary and special secondary schools the radioecology is introduced as the obligative subject at graduate courses.

2.5. In higher institutions the radioecology is delivered obligatevely. Its content depends on a training speciality profile. In higher institutions training specialists on radioecology the separate subject on general radioecology can be not introduced.

2.6. The content of radioecological education within the network of training and retraining specialists is defined by corresponding programmes of the Ministries, Councils and Boards supervising the organisations which deliver their workers to train or to retrain.

2.7. The enlightenment and informing of population in questions of radioecology is conducted at the base of special permanently operating studying, consulting and informing centres of radiation safety and hygiene placed in large towns and district centres of polluted areas of Belarus. The special attention should be paid to residents of

contaminated territories, resettled citizens, clean-up workers and people residing within the 30-km zones round operating NPPs.

2.8. To satisfy needs of the Republic in specialists on radioecology including training the academic staff the creation of profiled higher and special secondary institutions or opening special departments in existing institutions is required.

3. REALISATION OF THE CONCEPT

Taking into consideration the difficult economic situation in the country it is clear that the realization of the conception must start with the most effectual measures, which would lead to the best results with minimum expenditures.

3.1.At the secondary educational level the compulsory radiation protection course must be introduced in the school-leavers forms in the most suffered Gomel, Mogilev, Brest regions using the textbooks editted in 1994. It needs to found special educational rooms to consult teachers (radiometric instruments, textbooks of methodics, posters carried out in ISIR) in the regions which were mentioned already.

3.2.To develop and to introduce special radiation safety programmes according to the age of the children from the suffered regions in health-improvement and rehabilitation centres using the textbooks editted in Belarus and equipping them with educational consulting rooms.

3.3.Prolonged consequences of the disaster demand urgent planning of the gradual substitution of the expert posts in the State Authorities making decisions connected with the problems effected by Chernobyl accident, by the specialists who received a special training in ISIR and retraining at the Republic Academy of management.

CONCLUSION

In the future radioecological education must become a multilevel training embracing various strata of society. In Belarus where the fifth part of the territory is polluted by the radiowastes it is necessary to know the personal protectional rules. It provides not only successful radiation safety but it also creates normal conditions of living and managing in the radiation environment.

The concrete ideas discussed at the special seminar where the interested organisations took part were presented to the Government of Belarus.

In the face of developing nuclear technologies the radiation accidents can not be excluded completely in future, and the efficiency of eliminating their consequences will greatly depend on the knowledge and expertise available, thus the experience acquired in Belarus in providing radioecological education, the various forms of which are being currently introduced into the national system of education might be helpful for many other countries.

THE NEED OF EDUCATION OF BIOTECHNICAL SPECIALISTS IN THE FIELD OF RADIATION PROTECTION

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ABSTRACT

Education is the base for a successful carrying out of radiation protection measures. Starting from this fact, in the field of biotechnology protection measures should be carried out by biotechnical specialists (veterinarians, agronomists, technologists). In FR Yugoslavia, at the Faculty of Veterinary Medicine a separate course "Radiobiology and radiation hygiene" was introduced in undergraduate and postgraduate studies in 1976. However, other biotechnological specialists do not study the field of radiation protection separately at their faculties. Because of this, the Expert Group for Radiation Protection in Biotechnology formed at the Federal Ministry of Economy initiated the introducing of a course for this field in undergraduate and postgraduate studies at the faculties of agriculture and technology in FR Yugoslavia.

This paper presents the basic elements of the educational plan and program of the course "Radiobiology and radiation hygiene" for students of biotechnical faculties in FR Yugoslavia and discusses the results obtained until now.

INTRODUCTION

Radiation protection is a wide and a complex field with a range of specific problems. To solve these problems, a multidisciplinar approach and team work, with participation of specialists of different profiles (physicists, chemists, veterinarians, agronomists, doctors, food technologists, lawyers, sociologists etc.) is necessary. Taking into consideration that for an effective undertaking of radiation protection the knowledge of main characteristics and effects of ionizing radiation is necessary, as well as of the characteristics of the milieu where radioactive substances are located and which they affect, the protection can only be successfully fulfilled by reciprocal connection of these knowledge (1,2).

In addition, the basis of each protection, and so of radiation protection, is the prevention, i.e. undertaking of measures that prevent the intake of radioactive substances into human organism. Since the main way of the intake of radioactive substances is the ingestion (intake through food - it makes 70% of the total intake), the most effective way from the aspect of prevention and protection is to break the chain, i.e. to prevent the food contamination in the process of production, processing and turnover. This task can most successfully be fulfilled by biotechnical specialists (agronomists, veterinarians and technologists), whose main task is the production of "healthy" food and indirectly also the protection of human health (3-5).

THE BASIS FOR INTRODUCING OF TEACHING

Taking into consideration that the danger of radioactive contamination of greater proportions considerably increased in the last period, not only because of the possible use of nuclear weapons but also because of a real danger of accidents on nuclear power plants in peaceful times, the education of biotechnical specialists from the field of radiation protection deserves much bigger attention.

In order to include biotechnical specialists equally and competently in solving complex problems of radiation protection, it is necessary to introduce a separate course from this field in undergraduate studies and to organize postgraduate studies at all biotechnical faculties (Faculty of Veterinary Medicine, Faculty of Agriculture, Faculty of Food Technology).

SITUATION IN FR YUGOSLAVIA

In the previous period, the education from the field of radiation protection in undergraduate and postgraduate studies of biotechnical specialists had only an informative character. An exception is the Faculty of Veterinary Medicine in Belgrade, where since 1976 the field of radiation protection has been separately studied in undergraduate and postgraduate studies.

At the end of 1992 a meeting of representatives of biochemical faculties in FR Yugoslavia, of the Federal and Republic Ministries of Agriculture as well as of the permanent Expert Group for Radiation Protection in Biotechnology was held, that was devoted to measures that should be carried out in the field of biochemistry to reduce the radiation risk. On the basis of adopted conclusions, the Expert Group for Radiation Protection worked out The Bases for Elaboration of Educational Plan and Programme for teaching in the field of radiation protection in biotechnology at biotechnical faculties (Faculty of Agriculture, Faculty of Veterinary Medicine and Faculty of Food Technology) in FR Yugoslavia.

BASES OF THE EDUCATIONAL PLAN AND PROGRAMME

1. *Title of the course:* "RADIATION HYGIENE AND PROTECTION IN BIOTECHNOLOGY"

2. *Aim:* Education and training of biotechnical specialist to became able to carry out production and population protection, to reduce the radiation risk and to provide the production of healthy food and fed by complete perceiving of the production cycle and food processing as well as by applying measures of radiation-hygienic prevention, control and protection.

3. *General teaching plan:* Teaching would be carried out during one semester at the end of the third or at the beginning of the forth year of studying (semester VI or VII) with 3 + 2 classes per week, i.e. 45 + 30 classes per semester.

4. *General teaching programme:* The teaching programme is conceived and standardized for all biotechnical faculties, but in certain chapters the focus and the content of the subject are adapted to particularities of individual faculties and majors (Faculty of Agriculture - Husbandry, Cattle-Raising, Fruit and Wine Growing, Plant and Food Protection, Melioration, Food Technology of Plant Products, Food Technology of Animal Products, Agroeconomy; Faculty of Veterinary Medicine; Faculty of Food Technology).

a) *Theoretical teaching - lectures:* The lectures include the following topics:
1. Introduction with theoretical base and elements of radioecology, radiobiology and radiation hygiene and protection;
2. Sources of ionizing radiation in the biosphere;
3. Ways of spreading, migration and transfer of radionuclides;
4. Characteristics and effects of ionizing radiation;
5. Detection and dosimetry of ionizing radiation, radiation values and units;
6. Radiation load and radiation risk;
7. Endangering of biotechnical production by radioactive contamination in normal occasions as well as in case of emergency;
8. Preventive protection measures from radioactive contamination;
9. Radiation protection in biotechnology;

10. Rehabilitation measures and radiation decontamination in biotechnical production;
11. Radiation-hygienic control of biotechnical production;
12. Radiation-hygienic expert opinion on soil, plants, animals, food, water and feed;
13. Organization of the biotechnical monitoring system in normal situations and in case of emergency;
14. Carrying out of measures of radiation-hygienic control and protection from the field of biotechnology during export, import and transport on the territory of Yugoslavia;
15. International and domestic standards, conventions and legislation from the field of radiation protection.

b) *Practical teaching - practice:* The practice includes the following topics:
1. Introduction and work in radiological laboratory;
2. Measuring instruments for the detection and dosimetry of ionizing radiation - introducing with characteristics and principles of work;
3. Measuring of the radiation phon and determination of KonZ;
4. Estimation of the semilethal and the lethal radiation dose;
5. Instruments for measuring of radioactivity level - their characteristics and principles of work;
6. Ways of sampling and sample preparing for measuring of radioactivity level;
7. Measuring of the total beta activity by using the anticoincidental beta counter - LARS-5;
8. Measuring of the total beta activity by using the method of thick-layer-sample in normal conditions - LARA-86;
9. Measuring of the total beta activity by using the method of thick-layer-sample in case of emergency - LARA-10;
10. Estimation of the consuming value of food;
11. Gamma and alpha spectrometry - demonstrative;
12. Decontamination of space and equipment - estimation of the decontamination effect.

The Expert Group took care that the plan and the programme are uniform, consistent and compatible for all biotechnical faculties and that they correspond to modern scientific knowledge from this field.

c) *Literature:* For carrying out the teaching and preparing exams there is a contemporary domestic literature, which is cited as references.

CONCLUSION

Introducing of irradiation protection in education of biotechnical experts would provide reducing of radiation risk for the population, more effective protection and control during import as well as the possibility of an equal and recognized participation in the international turnover and exchange.

The proposed bases of the teaching plan and programme can be very useful during introducing the radiation hygiene in the educational system for biotechnical specialists.

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INFORMATION DES PROFESSIONS DE SANTE A ELECTRICITE DE FRANCE

INFORMATION FOR HEALTH PROFESSIONALS AT ELECTRICITE DE FRANCE

C. GALLIN-MARTEL[1], J. LALLEMAND[2], C. VROUSOS[3], H KOLODIÉ[4], H. PONS[5], M. DURR[6]

L'information des professionnels de santé dans le domaine des effets médicaux et sanitaires des rayonnements ionisants et de l'utilisation industrielle de l'énergie nucléaire reste trop souvent insuffisante et limitée aux spécialistes. Or de la qualité de cette information dépend pourtant en grande partie l'acceptation par la population d'un voisinage à risque potentiel et bien entendu ses réactions en cas d'incident ou d'accident. En effet, des différentes enquêtes réalisées, il apparaît qu'en situation normale, les professionnels de santé constituent des référents essentiels pour le public et qu'ils sont des relais d'information crédibles. En situation de crise, le champ de l'information rationnelle sera altéré par les rumeurs et la désinformation dont les conséquences en terme de santé publique peuvent être caractérisées par l'émergence de comportements inadaptés à l'échelle individuelle et collective. Dans cette situation, les médecins et pharmaciens, dont l'autorité morale et la compétence technique sont reconnues, auront un rôle essentiel de conseiller en santé publique à remplir auprès de la population.

Le corps médical perçoit à la fois le rôle que la population est susceptible de lui demander de remplir dans ces circonstances et son manque souvent très important d'informations adaptées. Interrogés sur les effets médicaux des rayonnements ionisants, les professionnels de santé déclarent être non ou mal informés à près de 90 %. Le même pourcentage souhaite être informé sur la conduite à tenir face à un irradié ou un contaminé et sur les mesures sanitaires de protection. Cette information doit transiter par l'intermédiaire de documents écrits à 58 %, de séances d'Enseignement Post-Universitaire (EPU) à 55 %, ou de colloques à 46 %. Concernant la formation initiale des médecins, il faut rappeler les résultats de l'enquête réalisée par questionnaire en 1992, auprès de 508 facultés de médecine de 24 états membres de l'OCDE. Plus de la moitié de ces facultés n'offrent pas un enseignement organisé de radiopathologie et de radioécologie durant le cursus des études médicales. Sur les établissements qui déclarent enseigner ces matières, seuls 18 % exigent un contrôle de connaissances.

Face à cette attente de formation et d'information des professionnels de santé, EDF a décidé de réaliser une information objective, validée scientifiquement, crédible et adaptée aux lecteurs. C'est dans ce but que des spécialistes hospitalo-universitaires, des représentants des Ordres et des organisations syndicales respectives, des responsables de la formation continue et des grandes administrations impliquées, sont associés à la réalisation des documents publiés au plan national et aux séances d'enseignement post-universitaire organisées au niveau

Information for health professionals on the medical and health effects of ionising radiation and the industrial use of nuclear energy is all too often inadequate and restricted to specialists in this field. Yet the quality of such information depends to a great extent on its acceptance by the population living close to a potential risk and its reaction in the event of an incident or accident. Indeed, from various studies, it would appear that, under normal conditions, health professional represent a reference and credible source of information for the public. However, in crisis situations, the range of rational information will be distorted by rumours and misinformation, the consequences of which, in terms of public health are reflected by the emergence of unsuitable individual and collective behaviour. Under such circumstances, doctors and pharmacists, whose moral authority and technical competence are recognised, will play an essential role in giving public health advice to the population concerned.

The medical profession sees the role that the public is likely to require it to adopt under such circumstances in the light of its often significant lack of suitable information. When questioned on the medical effects of ionising radiation, almost 90% of health professionals admit to being badly informed or uninformed. The same percentage expressed a wish to be informed as to the action to be taken when confronted with a victim of irradiation or contamination and the corresponding health protection measures. Such information should come from written documents (58%), Post Graduate training sessions (55%) or conferences (46%). In so far as the initial training of doctors is concerned, we should bear in mind the results of a study based on a questionnaire sent to 508 universities of medicine of the 24 members states of the OECD in 1992. The medical curriculum of more than half of these faculties did not provide any formal training in the fields of radio-pathology and radio-ecology. Of the institutions which did provide training in these fields, only 18% of them tested the knowledge of their students.

Given the expectations of health professionals for training and information, EDF has decided to publish objective, scientifically validated and credible information suited to its target audience. It is with this aim in mind that university hospital specialists, representatives of the respective trades union organisations, in-service training managers and the public bodies involved have come together to produce documents to be published nationally and to provide post graduate training at local/regional level.

At national level, these activities mainly concern:

loco-régional.

Les principales actions nationales réalisées concernent :

1. Médecins

La brochure *"Médecins et Risque Nucléaire"* constitue un des éléments de l'opération **Isère Département Pilote** dans la prévention des risques majeurs initiée en 1987 par le Ministre de l'Environnement. Des informations sur la conduite à tenir face aux diverses modalités d'accidents radiologiques, les mesures sanitaires et la place du médecin de famille dans l'organisation générale des secours sont proposées. En 1992, la 4ème édition actualisée et restructurée a été publiée et sa diffusion effectuée auprès de l'ensemble des médecins généralistes et des médecins spécialistes concernés. Au total, plus de 100 000 exemplaires ont été diffusés.

La revue *"Médecins et rayonnements ionisants"* publie des articles concernant la radiopathologie, la radioécologie et l'utilisation médicale des rayonnements sous l'égide d'un comité scientifique national présidé par le Professeur M. Tubiana. Sa diffusion est de 45 000 exemplaires. Sa périodicité est de trois numéros par an. Une évaluation de la perception de cette revue par le corps médical a été réalisée début 1993. Sur 500 réponses reçues : 96 % souhaitent continuer à la recevoir et 74 % la conservent. Neuf numéros ont été publiés depuis septembre 1991.

Le Service de Radioprotection d'EDF publie de très nombreux articles dans les revues destinées aux médecins généralistes ainsi que dans les revues spécialisées sous l'égide des Docteurs M. Bertin et J. Lallemand. Ce service organise par ailleurs tous les ans des réunions d'information sur un thème en rapport avec l'actualité scientifique et publie des brochures largement diffusées auprès du corps médical (en 1995 : Radiocancérogénèse : les acquis récents, 64 p., 3 000 exemplaires).

2. Pharmaciens

Deux brochures d'information ont été éditées en février 1995 :

- *Pharmaciens et Nucléaire - Informations Pratiques* (20 pages - 70 000 exemplaires) est diffusée sous l'égide du Conseil National de l'Ordre à tous les pharmaciens exerçant en France. Elle répond aux principales questions posées par les pharmaciens et précise leur rôle en situation de crise.

- *"Pharmaciens et Nucléaire - Conduite Pratique en Cas d'Accident"* (90 pages - 10 000 exemplaires) est diffusée aux pharmaciens directement concernés. C'est un document de référence dont l'objectif est de préciser les éléments à connaître en cas d'incident ou d'accident. Il concerne le pharmacien, en particulier, dans le cadre de son rôle de conseiller

1. Medical doctors

The leaflet *"Médecins et Risque Nucléaire"* (Doctors and Nuclear Risks) is one of the components of the operation **Isère Département Pilote** in the field of the prevention of major risks, which was initiated in 1987 by the Minister for the Environment. Information on the action to be taken in the case of various forms of radiological accident, the corresponding health measures and the role of the family doctor in the general emergency organisation were promulgated. In 1992, the 4th, revised and restructured edition was published and circulated to all general practitioners and specialists concerned. In total, more than 100,000 copies were issued.

The journal *"Médecins et rayonnements ionisants"* (Doctors and ionising radiation) publishes articles on radio-pathology, radio-ecology and the medical use of radiation, under the aegis of a national, scientific committee, chaired by Professor M. Tubiana. Some 45,000 copies of the journal are printed and it is published three times a year. At the beginning of 1993, poll was taken within the medical profession as to its perception of this journal. Of the 500 replies received, 96% expressed the wish to continue to receive the journal and 74% kept past issues. Nine issues have been published since September 1991.

The department of Radiological Protection of EDF publishes a large number of articles in general medical and specialist journals, under the aegis of Doctors M. Bertin and J. Lallemand. This department also organises information sessions on subjects related to recent scientific events and publishes leaflets with a wide circulation within the medical profession (in 1995: Radiocancérogenèse: les acquis récents, 64 p, 3,000 copies).

2. Pharmacists

Two information leaflets were issued in February 1995:

* *"Pharmaciens et Nucléaire - Practical Information"* (20 pages - 70,000 copies) is published under the aegis of the "Conseil National de l'Ordre" (national organising body) and circulated to all pharmacists practising in France. It provides answers to the questions most often put by pharmacists and sets out their role under crisis situations.

* *"Pharmaciens et Nucléaire - Practical action in the event of accident"* (90 pages - 10,000 copies) is circulated directly to the pharmacists concerned. This is a reference document, the aim of which is set out the various elements to be known in the event of incidents or accidents. It concerns pharmacists particularly in so far as their role as health advisers to the public is concerned.

en santé publique.

Leur réalisation a associé des pharmaciens d'officine, des enseignants des facultés de Pharmacie et de Médecine de Grenoble et de Lyon, l'Ordre National des Pharmaciens, l'O.P.R.I., des spécialistes d'EDF et du C.E.A..

These leaflets are produced in collaboration with dispensing chemists, teachers at the universities of Pharmacy and Medicine of Grenoble and Lyon, the "Ordre National des Pharmaciens", the OPRI and experts from EDF and CEA.

3. Vétérinaires

Une série d'articles a été publiée sous forme de suppléments à la *"Dépêche Vétérinaire"* sous la responsabilité de M. Michon, membre de l'Académie Vétérinaire : *Effets des irradiations aiguës sur les animaux domestiques* (09/02/91) ; *Les pollutions radioactives : conséquences pour le cheptel* (26/10/91) ; *Pollutions radioactives de denrées alimentaires d'origine animale : prévisions quantitatives* (25/04/92) ; *Les normes de protection radiologique - l'interprétation sanitaire de la pollution radioactive des denrées alimentaires* (19/06/93) ; *Organisation de la sûreté nucléaire - maîtrise des conséquences d'un accident nucléaire* (19/03/94).

3. Veterinary Surgeons

A series of articles, published in the form of supplements to the *"Dépêche Véterinaire"*, under the responsibility of M. Michon, member of the Veterinary Academy: *"Effects of severe irradiation on domestic animals"* (09/02/91); *Radioactive pollution: consequences for livestock: qualitative forecasts* (25/04/93); *Radiological protection standards - a health related interpretation of radioactive pollution of foodstuffs* (19/06/93); *Organisation of nuclear safety - controlling the consequences of a nuclear accident* (19/03/94).

4. Infirmières

En partenariat avec la Société Française de Biophysique et Médecine Nucléaire, EDF a mis en place un stand d'information lors des *Salons Infirmiers Européens* de 1993, 1994 et 1995 qui se sont déroulés à Paris. Une très importante documentation a été distribuée et des contacts multiples se sont établis avec la profession. Le logiciel *"Radiance"* a été présenté. Une brochure spécifique d'information est prévue en 1996.

4. Nurses

EDF, in association with the "Société Française de Biophysique et Médecine Nucléaire", has set up information stands at the European Nursing Exhibitions, held in Paris in 1993, 1994 and 1995. Many documents have been circulated to and contacts established with the profession. The *"Radiance"* software was presented and a specific information leaflet is planned for 1996.

Malgré cet effort en faveur de l'information des personnels de santé, il reste encore beaucoup de tabous à dissiper. L'actualité nous confirme que des informations plus ou moins alarmistes diffusées par les médias méritent d'être reprises sous la plume de personnalités compétentes et d'être replacées dans leur juste contexte.

Cet effort d'information doit être poursuivi activement pour que le corps médical puisse remplir avec compétence et indépendance son rôle de conseiller en santé publique.

In spite of all these efforts to spread out information to health professionals, many taboos and misconception remain to be dispelled. Current experience shows us that information, which is to a greater or lesser extent of an alarmist nature, published by the media, should be handled by competent personalities and placed in its proper perspective.

These efforts should be actively pursued so that the medical profession is able to fulfil its role as public health adviser,in both a competent and independant way.

1- Conseiller scientifique - Service Général de Médecine du Travail EDF-GDF - Paris ;
2- Service de Radioprotection EDF - Paris;
3- Chef du Service de Cancérologie Radiothérapie - CHU Grenoble ;
4- Praticien Hospitalier - Service de Cancérologie Radiothérapie - CHU Grenoble ;
5- Médecin Chef - Service Général de Médecine du Travail EDF-GDF, Paris ;
6- Production Transport EDF - Paris.

1. Scientific adviser to the General Health Medical Department EDF-GDF - Paris.
2. Radiological protection department EDF - Paris.
3. Head of the Cancerology and Radiotherapy department - CHU (University Hospital) Grenoble.
4. Hospital Doctor - Cancerology and Radiotherapy department - CHU (University Hospital) Grenoble.
5. Doctor in Chief of the General Health Medical Department EDF-GDF - Paris.
6. Production, Transport EDF - Paris.

A BRIEF HISTORY OF FEDERAL SUPPORT FOR HEALTH PHYSICS EDUCATION AND TRAINING IN THE UNITED STATES

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INTRODUCTION

Since the formation of health physics as a profession following the end of World War II, the federal government of the United States has played an active role in the support of education and training of health physicists. The purpose of this paper is to review the types of federal support that have been available from the federal government in the past and to examine the current status of support. Individuals trained in health physics through the nuclear navy programs have not been included in this discussion.

THE FIRST 25 YEARS (1948-1973)

In January 1947, following passage of the Atomic Energy Act of 1946 by Congress, The Atomic Energy Commission (AEC) assumed the responsibility for the nuclear energy program of the United States. The AEC recognized immediately that it could only meet its objectives if a cadre of qualified individuals were available with the needed levels of education and training to deal with the new science and technology. Consequently, the AEC began developing a number of programs to meet this anticipated need. In 1948, the National Research Council selected 20 students as the first group to receive Atomic Energy Commission fellowships in radiological physics (1). Ten were sent to the Oak Ridge National Laboratory, and ten went to the University of Rochester.

In October 1946, the organization now known as Oak Ridge Associated Universities (ORAU) was incorporated under the name Oak Ridge Institute of Nuclear Studies (ORINS). ORINS became a prime contractor to the AEC and began to develop a variety of activities involving faculty and graduate students. One of the AEC's most significant educational activities, carried out through ORINS, was the graduate fellowship program in health physics. In 1949, the AEC requested ORINS to organize and administer a program of special graduate fellowships that would provide one year of graduate study in health physics at Vanderbilt University or at the University of Rochester, followed by a three-month field experience at either the Oak Ridge or Brookhaven National Laboratory. The first group of fellows in the program were appointed for the 1950-51 academic year. The program was designated as the *AEC Radiological Physics Fellowships* for the period 1950 to 1959. Additional universities where added to the program, eventually reaching 18, and other AEC laboratories were selected for the summer rotations with 2 labs used each year (beyond Oak Ridge and Brookhaven) from a list of 7 participating labs.

The 18 universities that participated in the AEC fellowship program between 1950 and 1973 [and the numbers of fellows trained] were as follows (2): Georgia Institute of Technology [9]; Harvard Univ. [16]; New York Univ. Medical Center [4]; Purdue Univ. [8]; Rutgers Univ. [5]; Texas A & M Univ. [3]; Univ. of California at Berkeley [56]; Univ. of Illlinois [14]; Univ. of Kansas [66]; Univ. of Kentucky [5]; Univ. of Michigan [40]; Univ. of Minnesota [7]; Univ. of Pittsburgh [4]; Univ. of Puerto Rico [3]; Univ. of Rochester [208]; Univ. of Tennessee [48]; Univ. of Washington [95]; and Vanderbilt Univ. [347]. The total number of fellows participating in this AEC program between 1950 and 1973 was 940, trained at a total program cost of $6,078,000.

In 1959, the program was expanded to permit three years of support leading to the doctoral degree. It was designated as *AEC Special Fellowships in Health Physics*, a title which continued to 1970. In 1970, the name was changed to *AEC Special Fellowships in Radiation Science and Protection.*

In 1960 the AEC established a special fellowship program for individuals already working in the health physics field who had at least two years of experience. The program operated through 1967 and supported some 30 health physicists at a program cost of $597,000.

The Public Health Service (PHS) Radiological Health Training Grant Program was created in 1961 to provide a training mechanism to meet the national need for "radiological health specialists" as projected by the National Advisory Committee on Radiation in a 1959 report to the Surgeon General (3). As a result, the Department of Health, Education, and Welfare through the PHS was assigned the task of training professional and technical workers in the area of radiological health. A 1960 symposium at Princeton University on "University Curricula in Radiological Health" recommended a core curriculum for participating universities. The training was to operate at

two academic levels, Radiological Health Specialists at the graduate level and Radiological Health Technicians at the undergraduate level.

In 1961, the Division of Radiological Health of the PHS was authorized to administer the institutional training grants. By 1964 some 35 universities were participating in the specialist program and 10 in the technician program. Annual appropriations for the program reached $2.5 million per year. A total of $23,584,928 was appropriated to support this program over its 14-year lifetime. The annual reports (4, 5, 6) of the Public Health Service's Division of Radiolgical Health (later called the Bureau of Radiological Health) identify some 39 Universities that participated in the Specialist (graduate) program. The PHS Technician Training program was typically a two-year Associate of Arts or Associate of Science degree program. As many as 10 institutions participated in this training.

Ingraham (7) indicated that the PHS program goal was to train approximately 1200 individuals. It is unclear from the literature how many individuals were trained through the PHS Training Grants. The numbers should be included in the annual report on radiation protection enrollments compiled by the DOE (8), but the PHS component can not be identified. As an approximation, if the cost per trainee is assumed to be $12,000 to 15,000 as estimated by Moeller and Eliassen (9), then as many as 1800 radiological health specialists would have been trained through the PHS programs.

THE PAST TWENTY YEARS (1974 - 1994)

Following the 1975 reorganization that split the AEC into the Nuclear Regulatory Commission (NRC) and the Energy Research and Development Administration (ERDA), the former AEC Health Physics Fellowship program was continued by ERDA and,later, by the Department of Energy (DOE). During the ERDA years, the program was initially designated as "Traineeships for Graduate Students in Energy Engineering, Industrial Safety and Hygiene, and Environmental Fields", and was later identified as "Graduate Traineeships in Energy Related Fields". With the formation of the DOE in January, 1978, the program emerged as the "Nuclear Science and Engineering and Health Physics Fellowship Program". As before, the program was administered by Oak Ridge Associated Universities. There was an interruption in the program shortly after the DOE was formed when in 1978, DOE announced the suspension of graduate traineeship programs for Fiscal Year (FY) 1979 because of budgetary restraints and reassessment of priorities. In FY-1981, planning money was provided to reinitiate the graduate fellowship programs and fellowship awards resumed in FY-982. The number of fellows supported has been modest compared to the earlier AEC fellowship program. The numbers of new fellows in health physics portion of the DOE program were as follows: 8 in 1982, 5 in 1983, 7 in 1984, 4 in 1985, 1 in 1986, and 1 in 1987 . The values for 1988 -1993 remained modest at about 5 per year. In FY-1995 the program was on "maintenance funding" for the 5 fellows already in the program. No new health physics fellowships were awarded for the 1994-95 school year.

In 1990, the DOE's Office of Environment, Safety, and Health launched a new "Operational Health Physics Fellowship Program" which supplemented the existing health physics fellowship program (10). The "operational" program (now called the "Applied Health Physics Fellowship Program") is primarily a master's level program and is intended to prepare participants to move directly into mission oriented health physics work rather than pursuing the doctorate for research oriented careers. The program includes a practicum experience and DOE has designated 20 practicum sites within the DOE laboratory system.

The first fellowship period for this program began in the fall of 1990. Thirteen Universities were initially designated by the DOE as qualified participants. Four were later added and three of the original ones are no longer participants. These 17 institutions are (10, 11): Univ. of Cincinnati, Colorado State Univ., Univ. of Florida, Georgetown Univ., Georgia Institute of Technology, Massachusetts Institute of Technology, Univ. of Massachusetts at Lowell, Univ. of Michigan, Univ. of Missouri-Columbia, Univ. of N. Carolina, Ohio State Univ., Univ. of Pittsburgh, Purdue Univ., Rutgers Univ., San Diego State Univ., Univ. of Tennessee, and Texas A & M Univ.

The program is budgeted to support approximately 20 new graduate students per year. For FY 1993 there were 20 new students and 14 returning students for a total of 34 participants. The FY 1993 budget for the program was $919,000. The previous year's budget (FY 1992) had been $ 750,000.

A related fellowship program that could provide training opportunities for individuals interested in health physics has been the DOE's Environmental Restoration and Waste Management Fellowship Program which began in January, 1990. At present, the fate of this program for future health physics support is uncertain.

For more than a decade the U.S. Nuclear Regulatory Commission has had concerns about the availability of trained personnel to meet their own agency needs. In 1990, the NRC began its "Graduate Fellowship Program" which was administered by ORAU (12). The program is a 24-month M.S. curriculum designed to produce professionals in the areas of nuclear engineering, health physics or specialty engineering. Participants receive a

stipend of $ 20,400 per year, and participating universities receive an institutional allowance of $ 5000/yr. Unlike the DOE fellowships, the NRC fellowships entail a service obligation. This includes a 9-month pre-fellowship orientation during which the individual is employed at the GS-7 government level, and a post-fellowship obligation of 2 years of service for each year of graduate support as an employee at the GS-9 level. In 1991, there were 4 NRC fellowships awarded, two of which were in health physics (13) In 1992 there were 2 new awards, one of which was in health physics. In 1993, three awards were made. Clearly this program is supplying a very modest number of health physicists.

PROGRAM SUPPORT AND THE ACADEMIC INFRASTRUCTURE

The annual costs for training health physicists through the academic programs can be estimated from past budgets and numbers of students trained. Moeller and Eliassen (9) had made an estimate for the PHS trainees that were in programs during the 1960's. Training costs for the AEC/ERDA/DOE programs have likewise been estimated. Based on information received from Williamson (13, 14) the annual cost for trainees was about $5000 in 1957 and had risen to about $30,000 by 1995.

Historically, the AEC fellowship programs provided "institutional allowances" to assist the universities with costs associated with the programs. In the case of the PHS training grants, each institution was provided not only the funds for the student stipends, but also funds for faculty salaries, equipment and supplies. Over the past 20 years, with the termination of the PHS grants and the reliance on rather meager institutional allowances from the ERDA/DOE programs to support the infrastructure, the number and strength of many programs has diminished. An important step in reversing this situation was taken by the DOE in 1992 with the initiation of the "Health Physics Faculty Research Award Program". The program had a total budget in FY 1992 of $336,000 and provided awards of about $50,000 to each of 4 faculty members. In FY 1993, the program budget was $576,000 and a total of 8 awards (4 new and 4 renewal) were made. These awards include funds for student support, faculty salaries, and research supplies. The FY-1994 budget was similar.

The development of trained health physicists in the United States over the past 45 years has been closely linked to and dependent upon federal support, particularly at the graduate level. In recent years, such support is decreasing. The support for education and training that occurred in the early decades of the profession has resulted in today's strong cadre highly qualified health physicists. As federal support for advanced training erodes, we may expect some adverse impacts to appear in the future. Although private support and increases in short-course offerings may assist in meeting future needs, there will be an ongoing need for federal support of health physics training if we expect to maintain the overall quality of of our professional work force into the 21st century.

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THE REGULATORY EVALUATION OF RADIATION PROTECTION TRAINING PROGRAMMES AT CANADIAN NUCLEAR POWER PLANTS

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ABSTRACT

The responsibility for providing the necessary assurance that the use of nuclear energy in Canada does not pose undue risk to health, safety, security and the environment is vested with the Atomic Energy Control Board (AECB). This responsibility has led the Operator Certification Division of the AECB to develop methods to obtain assurance that nuclear power plant operations personnel are well trained and adequately competent to perform their duties. The features of the AECB approach to evaluation of training programmes based on a Systematic Approach to Training is described. An overview of the Canadian nuclear power plants' radiation protection qualification levels is given. The developing evaluation process is contributing to the improvement of licensee radiation protection training programmes. This is making possible the transfer of part of the responsibility for licensed personnel radiation protection qualification assessment to the licensees, thus enabling a reduction in the Operator Certification Division formal qualification activities.

INTRODUCTION

The federal agency responsible for the regulation of the nuclear industry in Canada is the Atomic Energy Control Board (AECB). Its mission is to provide assurance that the use of nuclear energy in Canada will not cause undue risk to health, safety, security or the environment. The health and safety of the more than 10,000 workers in seven Canadian nuclear power plants (NPPs), and the general public who may be affected by the operation of these reactors, is a primary concern of the AECB. Assurance of adequate radiation protection of the general public is acquired by approval of the design, and surveillance of the operation and maintenance of the reactors. The safety of the plant workers, responsible for good operation and maintenance, is achieved by adherence to good radiation protection practices.

Regulatory evaluation of NPP staff competence is vested with the Operator Certification Division (OCD). The mandate of OCD is to obtain assurance that all NPP operations personnel are well trained and competent to perform all their duties, including adhering to the radiation protection requirements.

NEW APPROACH

The quality of the training programmes in place has been verified to some extent by the AECB since 1962, but only since 1991, have radiation protection training protection programmes received specific attention through systematic regulatory evaluations against pre-determined standards. For many years the principal focus of the regulators was the training given to the two licensed positions in the NPPs: Control Room Operator (CRO) and Shift Supervisor (SS).

Indeed a pre-requisite to being granted approval to fill a position the passing of an AECB written examination in radiation protection. For these two positions, the emphasis is being shifted from the direct AECB examination of candidates at the end of their training, to the regulatory evaluation of the radiation protection training programmes together with licensee administered testing. However, certain areas of radiation protection knowledge for these two positions, for example, emergency response and effluent discharges, which could more directly affect the public, will continue to be examined by the AECB.

The basic radiation protection principle adopted at Canadian NPPs is that all operations personnel shall be able to provide effectively for their own radiation protection. This requires that personnel be fully aware of their roles and responsibilities with respect to access rights at their NPP, and that they be able to plan and perform work in a radioactive environment, using the appropriate protection as necessary. Consequently, all operations personnel must be suitably trained to protect themselves against potential radiological hazards. The new approach undertaken by the AECB also includes the evaluation of radiation protection training programmes for all operations personnel.

RADIATION PROTECTION QUALIFICATION LEVELS

Colour is used by the utilities to indicate the qualification level of personnel and is prominently displayed on the thermoluminescent dosimeter badge. The levels are "Red", for a person with no training in radiation protection at the site, followed by "Orange", "Yellow", and "Green", the highest level of qualification. The Orange level provides sufficient training to allow a plant worker or contractor unescorted access to areas of the station where levels of radioactivity are expected to be very low. A worker with this qualification can only do radioactive work under the supervision of a fully qualified person (Green). The Yellow badge is the intermediate qualification and provides the knowledge and skills necessary for a person to work in an environment where radiation might be present (levels of 10 μSv or more). The associated training includes work planning, dosimetry, radiation detection instrumentation and use of protective clothing. The Green badge is held by experienced individuals who must ensure the radiation protection of other lesser qualified individuals performing radiation work. The Green badge qualification is the entry level requirement to the training of licensed control room personnel.

THE EVALUATION PROCESS

The AECB expects that all NPP radiation protection initial and continuing training programmes conform to a Systematic Approach to Training (SAT). This widely accepted approach is described in detail in IAEA-TECDOC-525[1]. It is an effective and efficient process to provide job competency since it requires that training be designed to meet job performance requirements. The AECB uses a three phase model to describe SAT: 1) Preparation phase, 2) Implementation phase, 3) Evaluation phase. The first phase includes an analysis of the job and tasks requiring training (assessed through the frequency, importance to safety and difficulty of the task) and the design and development of the training material. The second phase, involves the selection of instructors, delivery of training and the testing of trainees. Finally, the third phase covers the use of feedback to maintain and improve the training programme.

In order to use a consistent approach when evaluating training programmes, the AECB has ecently developed the following three-step method: 1- Identify and check the adequacy of ıtility Requirements/Guidance for training programmes, 2-Review the supporting training documentation, 3- Determine if training is implemented in accordance with the documented process. The first two steps are completed at the AECB office. The last one is performed at site and involves observation of training in the classroom (radiation protection principles and practices), laboratory and on-the-job (instrumentation, protective clothing, access control), and interviews with staff and management. The tools used to assess the quality of training are derived from the AECB Objectives and Criteria for Regulatory Evaluations of NPP Training Programmes[2] which are based on SAT. The classroom and field observations as well as interview questionnaires are derived from this document. An interim report is presented before leaving the site, and a final report is later sent to the utility indicating where actions are required in order for the programme to be more effective. To gain assurance of the adequacy of the training, the three step method is used to evaluate all levels of radiation protection qualification at all stations. Attending all training sessions would be an overwhelming task. Therefore, the approach favoured is to select a cross-section of the programme. As an example, for a typical course, a few tasks would be chosen for which the learning objectives, training material, lesson plans, course notes, facilities, delivery and practice sessions, as well as the associated areas of the written and field tests for this task would be reviewed using the guide mentioned above. To better assess the preparation phase, an evaluation team member periodically attends the meetings of a licensee's Radiation Protection Training Group whose current primary role is revising the formal job and task analysis. During interviews, questioning covers all aspects of training, which allows the evaluation team to assess all phases of the training process.

CONCLUSION

The regulatory evaluation process, by expecting that the licensee radiation protection training programmes be in accordance with the SAT process, results in a training programme focussed solely on the job requirements. From the perspective of the licensee, this increases the programme effectiveness and efficiency by eliminating the components of the training that do not contribute to worker competency, thus saving resources. From a regulatory perspective, these evaluation activities, using a three-step process and criteria based on SAT, provide further evidence that, as a result of the training programmes, the plant workers will have the necessary skills and knowledge to perform their job duties effectively and safely. This additional assurance gained by the AECB through training programme evaluation is also making possible the transfer of part of the responsibility for licensed personnel radiation protection qualification assessment to the licensees, thus allowing a reduction in the Operator Certification Division examination activities.

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The CEFRI and Radiological Protection Certification of French Companies and Training Organizations

In the matter of risk prevention, the quality of training is essential. This is why the French Committee for Certifying Companies in Training and Followup of Personnel Working Under Ionizing Radiation (CEFRI) was created in 1990.

Besides the employers of Category A and B workers, represented by the GIIN (nuclear industry trade group), associated within the CEFRI are the operators of French nuclear facilities (EDF, CEA, Cogema, and the military), the French Radiological Protection Society (SFRP), the national health insurance agency, and the Office of Protection Against Ionizing Radiation (OPRI). In addition, the Committee receives support from the Ministries of Labor and Industry.

The fundamental objectives of the CEFRI are as follows:

- To contribute to the reduction and prevention of the risks to personnel working in nuclear facilities; and
- To enable French companies active in this field to objectively demonstrate the preventive actions implemented.

After performing an audit, the CEFRI issues certifications:

- To companies (including companies providing temporary workers) that perform work in nuclear facilities, concerning the quality of personnel training (whether provided by an outside organization that is itself certified or by the company itself) and concerning the organization that the company has set up to ensure the followup of exposure and medical surveillance of its personnel; and
- To specialized training organizations, concerning the contents and quality of the training provided.

The four operators of French nuclear facilities (EDF, Cogema, the CEA, and the military) have already defined a schedule of requirements for this certification, as concerns the training organizations, the companies providing temporary workers, and the companies present in certain specific fields of activity, such as cleanup, industrial logistics, etc.

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A HEALTH AND SAFETY PRIMER FOR THE PRACTICING HEALTH PHYSICIST

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INTRODUCTION

Environmental restoration (ER) is the process of removing a facility from service, the demolition of structures, the identification and disposal of all hazardous and radioactive wastes, the decontamination of equipment and materials, and the restoration of a site for unrestricted use. The number of ER projects encompassing hazardous, industrial, and radiological conditions is expected to increase in response to various program requirements or mission changes. As a result, the practicing health physicist (HP) may have to address unique health and safety (H&S) issues beyond those of performing routine radiological activities. These unique H&S issues could include, but are not limited to the razing of buildings, the removal of radioactive materials and hazardous chemicals, below-grade excavation, confined space entry, storing flammable or combustible liquids, monitoring exposure to hazardous substances, contacting energized systems (e.g., electricity, hydraulics), noise abatement, the nullification of manufacturer warranties, and the operation and movement of heavy equipment. The purpose of this paper is to educate the practicing HP about these issues by reviewing specific regulations governing all H&S activities, and to provide an example of a site-specific H&S primer (e.g., Health and Safety Plan [HASP]). This primer advises the practicing HP about sound H&S principles, furnishes basic strategies for performing a hazard assessment/job safety analysis (HA/JSA) that can be applied to any ER project, and describes various engineering and administrative controls to mitigate hazardous exposures to ER personnel. In addition, 26 inspection checklist topics are available from the primary author to evaluate the adequacy of the engineering and administrative controls, or to necessitate the use of personal protective equipment (PPE) thereby mitigating the corresponding hazard.

REGULATIONS, STATUTES, AND ORDERS

The specific regulations governing all H&S activities are found in *Title 29 of the Code of Federal Regulations, Parts 1910 - General Industry Standard* (1) and *1926 - Construction Standard* (2). Also, applicable state, tribal, and local statutes, including U.S. Department of Energy Orders will have to be considered depending on the geographical location and regulatory jurisdiction of the ER site. Essentially, the regulations, statutes, and Orders are intended to provide uniform guidance that is generic in nature, thus allowing for the evaluation of site-specific H&S conditions at all ER sites. The regulations, statutes, and Orders also address general program guidance issues such as monitoring performance, records and reporting, and requirements for training ER personnel commensurate with the hazards encountered at the ER site.

SITE-SPECIFIC HEALTH AND SAFETY PRIMER

The types of hazards and inherent risks encountered by ER personnel often exceed the scope of performing routine HP activities. Therefore, to educate the practicing HP about these hazards and inherent risks, H&S personnel have developed a site-specific H&S primer. The primary element of this primer that will influence budgets and overall operations significantly, is the HA/JSA of site activities performed by ER personnel. A typical HA/JSA involves the following:

A) The identification of the job, task, or operation.
B) The breakdown of the job, task, or operation into individual steps.
C) The identification of all hazards associated with each step.
D) The application of controls to mitigate exposures corresponding to the identified hazards.

The methods available to mitigate exposures to ER personnel are listed in order of preference:

A) Engineering controls - the most preferred option for worker protection that includes the specification, design, purchasing, and performance of specialized equipment.

B) Administrative or work practice controls - the incorporation of H&S provisions into working-level procedures, and performing in-house inspections.

C) The allocation of PPE - since the complete elimination of the hazard may not be practical, ER personnel may be required to wear applicable PPE (e.g., respirators). However, the use of PPE is always the least desirable option since it may require additional training, medical examinations, as well as the cooperation of ER personnel to wear the PPE properly.

To familiarize the practicing HP about the proper use and application of H&S engineering and administrative controls at ER sites, a list of common elements and their H&S implications are provided for information only, and are not limited to the following:

A) Berms and tarps - designed to contain leakage of chemicals.

B) Below-grade excavation - the application of trenching, shoring, and barriers to prevent the collapse of materials.

C) Confined space entry - the placement of barriers (e.g., fencing) preventing unauthorized entry to confined spaces having limited or restricted means for entry or exit, and are not designed for continuous occupancy. Other requirements may include atmospheric testing, respiratory protection, and capabilities for emergency rescue of ER personnel occupying the confined space work area.

D) Fire hazard analysis - the availability and type of fire equipment shall be posted, and commensurate with the type of combustible materials and ignition sources present at the ER site.

E) Hazard communication program/material safety data sheets (MSDS) - the complete inventory and storage of chemicals in approved cabinets, and proper labeling of containers and tanks.

F) Lockout/Tagout (LOTO) of energized systems - the deactivation of pneumatic, hydraulic, gravitational, and electrical systems prior to initiating maintenance activities.

G) Noise abatement - the selection and placement of noisy equipment in proximity to ER personnel, or the use of sound barriers.

H) Equipment warranties - modifications compromising the intended design and/or operation of equipment shall be removed from service, and may result in nullifying agreements between the manufacturer and the purchaser.

I) Operating heavy equipment - drivers operating trucks, dozers, and scrapers have limited visibility and shall be given right-of-way privileges. The application of brakes and tire chocks shall be applied when equipment is stationary for purposes of inspection, or not in use.

In addition, 26 sample checklist topics are available from the primary author to enhance inspection capabilities by ER personnel. The sample checklists are provided as guides only, and are not limited to the following:

1. Chemicals
2. Compressed Gases
3. Construction Tools
4. Dusts, Gases, and Organic Solvents
5. Dusts, Gases, and Organic Solvents (Medical)
6. Ergonomics
7. Life Safety Code - Exits, Walkways
8. Explosives and Blasting Agents
9. Fire Prevention and Fire Protection
10. Flammable and Combustible Liquids
11. Hand and Portable Tools
12. Heat
13. Machinery and Machine Guards
14. Job Planning and Job Briefing
15. Material Handling and Storage
16. Medical and First Aid
17. Noise
18. Noise Stress (Medical)
19. Personal Protective Equipment
20. Sanitation
21. Temporary Grounding
22. Traffic Barricades
23. Trenching and Shoring
24. Vehicles and Cranes
25. Walking and Working Surfaces
26. Welding, Cutting, and Brazing

Additional components of the site-specific H&S primer should include, but are not limited to the following:

A) A brief summary of the historical information, physical characteristics, and meteorology of the site.

B) A map identifying structures, areas of remediation, and traffic patterns.

C) Access control and training requirements.

D) Medical monitoring and surveillance requirements for ER personnel coming in contact with hazardous substances.
E) First aid and response to medical emergencies.
F) Decontamination of ER personnel and equipment.
G) Transportation safety.
H) A waste minimization/pollution prevention program.

CONCLUSIONS

The types of industrial and hazardous conditions encountered at ER sites will require practicing HPs to address unique H&S issues beyond the scope of performing routine radiological activities. Therefore, the implementation of a site-specific H&S primer developed by certified ER personnel (e.g., Certified Safety Professional [CSP], Certified Industrial Hygienist [CIH]), in conjunction with the application of engineering and administrative controls will offer the practicing HP an opportunity to grow professionally by gaining experience with new disciplines associated with ER projects. In addition, the cooperative effort provided by H&S and HP personnel will provide a strategic approach to affirm balance between disciplines, maintain ease of communication between applicable participants, and identify pertinent issues at multi-hazard (e.g., chemical, industrial, and radiological) ER and similar project sites.

REFERENCES

1. "Occupational Safety and Health Standards for General Industry," *Title 29, Code of Federal Regulations, Part 1910,* Occupational Safety and Health Administration (OSHA), Washington, DC.
2. "Occupational Safety and Health Standards for the Construction Industry," *Title 29, Code of Federal Regulations, Part 1926,* Occupational Safety and Health Administration (OSHA), Washington, DC.

A UNIVERSITY SAFETY PROGRAM FOR USE OF RAMS

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To remain in compliance with 10 CFR Parts 19 and 20, and the conditions of our Byproduct Materials License, the Division of Radiation Safety (DRS) of the Office of Chemical, Biological and Radiation Safety (CBARS) of the University of Missouri-Kansas City (UMKC) has established a three level training program for persons using radioactive materials (RAMs) and those persons frequenting areas in which RAMs are used. Attendance at each level of training is mandatory. Training for new employees before involvement with RAMs and annual refresher training is provided to all personnel. The training is conducted by the Radiation Safety Officer and DRS Health Physicists. Applicable literature is distributed during the training programs.

Faculty and staff who apply for and received approval from the Radiation Safety Committee to use RAMs are called Authorized Users (AUs). Authorized Users, independent of the amount of training and experience they have, must attend the mandatory Level 1 training program. Introduction to the program includes a discussion of the relationship between the radiation safety program, the U.S. Nuclear Regulatory Commission Byproduct Materials License, with conditions, issued to UMKC, and applicable Federal and State regulations pertaining to the use of RAMs. Copies of the UMKC Handbook of Radiological Operations, and NRC Regulatory Guides 8.13, **Instruction Concerning Prenatal Radiation Exposure**, and 8.29, **Instruction Concerning Risks from Occupational Radiation Exposure**, are distributed and discussed. Information contained in 10 CFR Parts 19 and 20 is reviewed with emphasis on those sections that impact directly upon the AU's use of RAMs, e.g., dose limits and posting requirements. License conditions that apply, e.g., experimental animals that have been administered licensed materials shall not be used for human consumption, are described. A looseleaf version of the Handbook is given to each AU and contains the following seven sections: Section 1.0 Management of Radiation Safety, Section 2.0 Radiation Control, Section 3.0 Procedure for Laboratory Inspections, Section 4.0 Radioactive Waste Disposal Program, Section 5.0 Application for Use of Radiation Sources, Section 6.0 Forms and Section 7.0 Definitions and Acronyms. Emphasis is directed toward the sections on Radiation Control, Radioactive Waste Disposal Program and Forms. The section on Radiation Control contains information on personnel and area radiation control, emergency procedures, receipt, storage and use of RAMs and criteria of health physics coverage. The section on Radioactive Waste Disposal Program contains information on waste handling procedures and the responsibilities of the RSO and the AU with respect to the disposal of RAM wastes. The Form section contains copies of the forms used in the conduct of the radiation safety program. A sampling of the forms included in this section are the following: Application for Use of Radiation Sources, Authorization for Possession and Use of Radiation Sources, Laboratory Radiation Safety Orientation Checklist, Radioactive Materials Delivery Form, Radioactive Waste Pickup and Quarterly Inventory.

Radiation safety practices at UMKC including dose limits, ordering and receiving procedures for shipments of RAMs, laboratory survey procedures and emergency procedures to be followed in the event of a minor or major spill are discussed. The dose limits published in 10 CFR Part 20 and the As Low As Reasonably Achievable (ALARA) concept are reviewed. Investigational dose limits are also presented. A generic emergency procedure to be followed by the AUs and their associates in the event of a spill, complete with emergency telephone numbers of DRS personnel and the UMKC Police Department, is distributed. Pager numbers of DRS personnel are also made available.

Two procedures are used to order RAMs. The first follows the Purchasing Department's procedures by requiring the AU to complete a purchase requisition. This purchase requisition must then be approved by DRS to assure that the AU is authorized to use the RAM to be ordered. The processing time for orders placed in this manner may take two or three weeks. Therefore a second ordering procedure allows the AU to order RAMs via the telephone. Authorization must still

be obtained from DRS prior to telephoning the order into the company. Both procedures are discussed in detail. It is emphasized that deviations from these procedures will not be allowed because all RAMs must be accounted for as the authorization limits of the byproduct materials license cannot be exceeded.

A requirement of the radiation safety program is that all shipments of RAMs must be delivered to specific locations. UMKC consists of three separate campuses: Volker, Hospital Hill and Truman. It is explained to the AUs that shipments of RAMs must be received either at Central Receiving for those users on the Volker Campus and at the Medical School for those on the Hospital Hill Campus. RAMs are not currently in use on the Truman Campus. All shipments of RAMs are inspected and delivered to the AU's laboratory by DRS personnel. Another requirement of the radiation safety program is that each laboratory in which RAMs are used must be surveyed after each day's use. The survey must be performed with calibrated equipment and documented. In laboratories where low energy β-emitters are used the survey should consist of wipe sampling of areas in which RAMs are stored, used and the surrounding environs. In laboratories where moderate to high energy β-emitters and γ-emitters are used the survey should include radiation monitoring and wipe sampling of areas in which RAMs are stored, used and the surrounding environs. A further requirement concerns the disposal of RAM wastes. The policy at UMKC is that liquid RAM wastes are not to be disposed of into the sanitary sewerage system other than those incidental to washing laboratory glassware that has been used for low level RAM work and previously rinsed, at least once, into a liquid RAM waste container. Liquid RAM wastes are collected in one-gallon plastic bottles appropriately labeled. Solid RAM wastes are collected in specially marked and plastic-lined 12- and 28-gallon fiberboard drums. Separate waste containers are provided for different radionuclides to facilitate disposal. DRS personnel collect and dispose of all RAM wastes by decay-in-storage, by disposal into the sanitary sewerage system or by shipment to a commercial disposal facility.

The final topic for discussion for Level 1 training is the requirements for recordkeeping. It is stressed that complete records must be maintained for the receipt, use, and disposal of RAMs. Records of laboratory surveys, calibration of survey and laboratory counting equipment and other information pertinent to the description of the experimental program that would be of interest in determining the adequacy of the radiation safety program in each laboratory must be on file. Attendance records attesting to the fulfillment of the training requirements of the radiation safety program must also be maintained by the AU.

The Level 2 training program is directed toward radiation workers, associates of the AUs. Radiation workers conduct experimental protocols under the authorization granted to the AU by the RSC. This level of instruction includes some of the subject matter covered in Level 1 but emphasis is on the hands-on, day-to-day operational aspects of radiation safety in the use of RAMs. A bound pamphlet containing Sections 2 and 4 of the Handbook (important hands-on information and procedures) and NRC Regulatory Guides are distributed and discussed during this level of training. The contents of Section 2 are thoroughly discussed including: smoking, eating or drinking **shall** not be permitted; pipetting **shall** not be done by mouth; RAMs in liquid form **should** be stored and transported in double containers; flammable liquids **shall** be contained in UL approved containers, and work **should** be planned ahead. Conditions of use of protective apparel, e.g., laboratory coats, are addressed. Laboratory coats **should** be buttoned up when worn and **shall** be monitored periodically. The use of the types of personnel monitoring devices, the lapel film badge, the TLD extremity badge and self-reading pocket dosimeter, is explained. The bioassay requirements for users of ^{3}H or $^{125,131}I$ are defined. Section 4 contains information on the collection and disposal of RAM wastes.

Requirements for laboratory survey procedures, contamination levels and decontamination, and recordkeeping are examined closely. The purpose and use of survey meters and laboratory counting equipment applicable to the experimental protocols conducted in the laboratory are discussed. The Laboratory Radiation Safety Orientation Checklist which covers item such as RAMs used in the laboratory, posting, safety precautions and emergency procedures is reviewed. The Radiation Worker Checklist, a compilation of information collected during the training program, which provides DRS personnel with a measure of the individual's ability to use RAM, to understand radiation safety procedures and to use survey and laboratory counting equipment is also covered. Instructions for completing the RAM waste pickup and quarterly inventory forms and completed

examples of each are discussed.

Level 2 training is completed when the radiation worker first uses RAMs. DRS personnel oversee the first use of RAMs to verify 1) that the person can conduct the experimental protocol, 2) demonstrates radiation safe practices, and 3) performs the proper survey after use. If, in the opinion of the DRS person overseeing the first use, the radiation worker has performed satisfactorily, then that individual is free to continue to use RAMs without DRS supervision. However, if the individual has not performed satisfactorily, then subsequent oversee episodes may be required before permitting the person to work unsupervised.

Level 3 training is provided to groups of ancillary personnel frequenting areas in which RAMs may be used. These training sessions are very specialized for each group. The material covered during these training sessions is essentially the **DOs** and **DON'Ts** of frequenting areas in which RAMs are used or stored. There are four groups of ancillary personnel that receive training. They are personnel associated with the following departments: 1) Custodial, 2) Campus Facility Management, 3) Campus Police and 4) Receiving. A listing of all laboratories in which RAMs are used, examples of the warning labels in use on the campus and the Regulatory Guides are distributed to each group. Custodians are instructed not to handled any container with a 'Caution - Radioactive Material' label affixed to it. They are told not to empty any container of waste, solid or liquid, that is so labeled. The Custodians are told not to clean up posted areas in the laboratory, e.g., fume hoods or bench tops, in which RAMs are used and stored. They are not to clean up any spilled liquids unless the liquids have been identified by personnel associated with the experimental work conducted in the laboratory and are not hazardous or radioactive. Custodial personnel are also instructed to report any unusual conditions or findings to the Office of CBARS.

Campus Facility Management, the personnel responsible for maintenance of the buildings and equipment on the campus, are also required to attend a training program. It consists of similar instructions for the Custodians. CFM personnel are instructed not to work on any equipment with a 'Caution - Radioactive Material' label affixed to it until they have contacted DRS to determine if the equipment is free of any residual contamination. Types of equipment could include refrigerators, centrifuges, fume hoods and sinks. In anticipation of potential problems, the fume hood blowers, located on the roof tops of the buildings, have affixed to them 'Caution - Radioactive Material labels containing the name of the AU, the laboratory number and the DRS telephone number. Facilities must be surveyed prior to any remodeling.

The UMKC Police Department is used, by DRS, primarily for emergency response to incidents involving RAMs. The Department is given a callup list of DRS personnel who are available to respond to emergencies during off-hours. Police Department personnel are instructed in procedures to be followed in the event of an emergency involving RAMs, e.g., a spill. Instructions include containing, if possible with minimal risk and exposure, the RAM; if necessary turning off the utilities; holding all persons involved in the spill until DRS personnel can monitored them for contamination and radiation exposure; controlling any crowd that may assemble; and limiting access to the involved area to authorized personnel only. The communications section of the Police Department provides liaison with the Kansas City (KC) Fire and Police Departments if their services are required during an emergency involving RAM. KC Fire and Police Department personnel are familiar with UMKC's facilities through training provided by CBARS personnel. If medical treatment is required, the UMKC Police Department can contact local ambulance services and hospitals.

Personnel who work for Receiving accept delivery of shipments of RAMs. Their training consists of discussions of the following topics: 1) Types of packaging and labeling (NOS, White I, Yellow II and III), 2) Inspection of shipments of RAMs, for physical damage, including wetness, 3) If package is damaged, retain the driver of the delivery service and the package and call DRS, and 4) If package is not damaged, call DRS for to pick up the package, inspect and delivery it to the purchaser's laboratory.

In addition to these training programs, CBARS personnel also conduct training programs in the management of hazardous chemicals and biological agents.

THE CHANGING ROLE OF HEALTH PHYSICISTS AS REFLECTED BY CHANGES IN PROFESSIONAL HEALTH PHYSICS TRAINING COURSES

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Background

Health Physics is a profession with long, honourable traditions; and this paper could be subtitled "Health Physics - The First 100 Years". The discovery of X-rays by Conrad Roentgen in 1895, and of natural radioactivity by Henri Becquerel in 1896, was followed two years later by the isolation of radium by Marie and Pierre Curie; and then, during the last years of the nineteenth century, by explosive world wide growth in the utilisation of both these new discoveries for medical diagnostic and therapeutic purposes. The fact that these new medical tools carried associated risks was very quickly learned. Physicians who most enthusiastically adopted them often experienced severe skin injuries to heavily exposed digits, and there are numerous photographs of the hands of such individuals after experiencing several amputations. Regrettably many ultimately fatal radiation induced cancers also began to appear before the end of the last century, by the first world war there were 200 of these and the death toll already exceeded 50. In the face of this two edged weapon it is not surprising that many of the physicians and medical physicists working in this area turned a great deal of their attention from the exploitation of the new technologies to the protection of their colleagues. These individuals were the pioneer health physicists and, although this name was not used at the time, their background experience in both medicine and physics laid scientific foundations for the new discipline which have remained its keystone ever since.

Initially attempts to develop improved safety standards for radiology concentrated on the development of equipment and procedures which would minimise inadvertent exposure to X-ray beams. A Boston dentist, Dr. William Rollins, was particularly active in this respect, and he has been referred to as "the father of health physics". By the summer of 1915 such developments led the British Roentgen Society to pass a resolution recommending the universal adoption of stringent safety rules designed to ensure the personal safety of operators conducting roentgen-ray examinations. In November of the same year the Society followed up this resolution by publishing what appears to have been the first comprehensive set of such rules. However at this time monitoring for scattered radiation was still carried out either with fluoroscopic screens or by photographic means and any formal scientific basis for health physics awaited the development of appropriate units and instruments for the measurement of radiation exposures. This development took place at the Second International Congress of Radiology in Stockholm in 1928 where a Committee was appointed to study the introduction of an appropriate unit for dose measurements. The Committee was responsible for the introduction of the Roentgen as a unit of measurement. It subsequently developed into what became known as the "International X-Ray and Radium Protection Committee", and in 1934 under this name it recommended the universal adoption of a daily personal dose limit for radiation workers, a requirement that had been first introduced in the USA in 1929. Later it was this committee that developed into the International Commission on Radiological Protection in 1950.

The Evolution of Health Physics as a Profession

It is almost certainly true that during the inter-war years these organizational developments were of less significance in the evolution of health physics as a profession in its own right, than the dedicated work of many extremely competent physicians and medical physicists in hospitals throughout the world. Both fundamental research in radiobiology and sophisticated investigations of dose distributions within the body played a necessary part in the development of improved radiotherapy practices. They also laid a foundation for the safe control of artificial radionuclides when these became available in quantity following the development of the nuclear reactor, and equally important they ensured the availability of the core of expert scientists who were responsible for the safety of workers engaged in the development of atomic weapons during the second world war. These scientists were the first group to be formally referred to as health physicists, partly in order to conceal the exact nature of the work in which they were engaged. At this time the total quantity of radium which had been isolated for use

throughout the world would have comprised less than a one inch cube, yet they all knew that experience in the luminising industry had shown that lethal internal doses of radium could be very quickly absorbed. It has been reported that, when first recruited, they were given a realistic and frightening impression of the scientific problems they faced by a comparison of this quantity with the quantities of artificially produced long life alpha emitters which would be generated during the program to develop atomic weapons! There is no doubt that when the veils of secrecy were finally lifted fifty years ago their dedicated efforts had led to the development of a fully fledged health physics profession. During the next few years general expectations developed that nuclear power would be quickly and almost universally adopted, and consequently a significant number of additional health physicists would need to be trained. The first University M.Sc. courses in health physics were therefore established at this time. Numerous shorter courses were also introduced by a variety of training institutes, these were mostly intended for various groups of workers of very different academic capabilities who used radiation sources in their employment.

Health Physics as an Academic Discipline

Not only the ICRP itself but also national regulatory bodies in many countries have paid a great deal of attention to the need to ensure that all workers exposed to sources of ionizing radiation receive comprehensive training covering both the associated risks and the safety procedures that must be followed. Until recently the development of such training courses has tended to be emphasised at the expense of formal academic training of the senior staff responsible for designing and implementing complete radiation safety programs in large institutions. For example in 1979 the Health Physics Society devoted its Mid-year Topical Symposium to Health Physics Training, and an examination of the proceedings for this meeting shows that it was largely devoted to the training of health physics technicians, and of workers incidentally involved with the uses of ionizing radiations, relatively few papers discussed the academic training of personnel expected to be responsible for conducting health physics programs. This is perhaps not surprising as most such programs were already under the control of very experienced personnel who had learned on the job. Some of the early academic courses even found it difficult to recruit enough students and not all of them have survived. However most of these senior personnel have now retired, or are close to retirement, and the way to senior positions in the profession is no longer readily open to those who have not undergone a formal academic training. A review of the changing syllabus followed by such training courses therefore provides an easy way to assess how the issues of most importance to professional health physicists have changed over the years. The author attended a University M.Sc. course in radiation safety during the 1960's, and has subsequently been involved in both teaching and syllabus preparation for three further academic courses of the same type. It was only an incidental review of the subject matter covered in his first course which led him to full recognition of the substantial changes in the type of work undertaken by a health physicist that have occurred during the past four decades.

Syllabus Evolution in Academic Health Physics Training

The earliest academic courses covering radiation safety were designed for trainee medical physicists. They provided extensive coverage of the fundamental scientific disciplines of anatomy, physiology and biochemistry, together with training in the interaction of gamma radiation with matter, radiation dosimetry, radiobiology, dose distributions in tissue, medical imaging, etc. These courses also usually included a limited amount of information about ICRP recommendations, radiation control regulations and personal monitoring, but they were strongly directed towards students expected to make their career in a hospital. More general radiation safety courses first evolved from such medical physics courses to meet the needs of students planning to work in research centres, industry, or government. In addition to the fundamental scientific disciplines referred to above, these courses included a great deal more emphasis on instrumentation of all types, on the design of radiation shielding, on the in-house calibration of equipment, and on radiation measurements. Generally such topics as the operation of personal monitoring programs, contamination of materials, low level counting, bioassay procedures and whole body counting were covered in some detail. There was also some discussion of the role and recommendations of the ICRP, but regulatory requirements did not loom large at this time. Visits from government inspectors to research centres or industrial operations were relatively infrequent, and at this time the requirements of such inspectors were usually easily complied with.

About ten years after participating in such a course, I became involved in developing the syllabus for another course designed for a similar class of M. Sc. students. The most important changes from the type of syllabus outlined above which were felt to be necessary included increased emphasis on compliance with regulatory requirements, some coverage of the need to maintain adequate documentation of all procedures followed, an introduction to the rapidly increasing role of computers (e.g. for record keeping, calculating shielding requirements, contributing to the design of research projects, etc.), and adequate treatment of the many new developments in radiobiology - particularly at the cellular level. To make time for these topics to be fully addressed it was necessary for some of the material relating to medical imaging to be eliminated, but since many of the students would continue to be involved with work in hospitals, a comprehensive treatment of quality assurance procedures for medical imaging equipment remained.

The third such course with which I was directly involved put much more emphasis on both regulatory requirements and record keeping, since these were matters which were beginning to occupy more and more of the time of a program manager. The increasing power and availability of computers led to more attention being paid to their use in meeting such requirements. Questions of the potential legal liability of the responsible organization, and consideration of the problems encountered in ensuring that all workers complied with required operating procedures, assumed increasing importance. For the first time the public concerns, which were evident following the Three Mile Island accident, and had became much more acute after Chernobyl, led to a considerable amount of attention being devoted to techniques for communicating with the public. To compensate for this, less attention was paid to the work which it was by then assumed would normally be carried out by a health physics technician, and to the development of specialised health physics services such as personal monitoring and whole body counting which had begun to become readily available from commercial suppliers.

Further such changes have gradually continued over the last decade, and have tended to mark the evolution of the role of the radiation safety program manager from that of an operating scientist to one which has become largely program management. In the most recent course with which I have been involved it was found necessary to begin to reduce the attention given to the three fundamental sciences of anatomy, physiology and biochemistry to make room for an extended treatment of risk assessment, risk communication, and risk acceptability. Communication with both the public and employees assumed a very high importance and much more attention needed to be paid to legal issues including probability of causation. The documentation required of a typical organization has mushroomed out of all proportion, a considerable amount of time now needed to be spent both in reviewing these requirements and in considering how best to meet them. Major organizations now have to establish much more complex health physics programs than in the past, and this means that program direction and management has now become an important topic in its own right.

Future Directions in Health Physics and in Health Physics Training

The most critical question which is likely to arise before the end of this century is probably how the somewhat rigid structure that has been developed for radiation safety can best adapt to the implications both of new research and of the changing expectations of society. Examples of such issues are the possible development of techniques for the identification of individuals with abnormally high radiation sensitivity, and the conflicting expectations of some women's groups concerned about risks during pregnancy with those of others more concerned about the impact of pregnancy dose restrictions on their career prospects. Risk communication will lie at the heart of such societal decision making processes. Similarly increasing sensitivity to all forms of pollution in the environment is likely to lead to increasing importance being attached to the effective communication of information about the relative significance of all the associated risks. Answers to such questions will demand familiarity with an extremely broad range of scientific expertise. Society is now being faced with recognition that world energy needs are increasing dramatically, and that the greenhouse effect is a serious threat. This will necessitate a re-examination of the future role of nuclear power. The decisions on all such issues will ultimately have to be made by the public following extensive discussions in the media, and later in legislative assemblies. The professionally trained health physicist will inevitably be at the centre of such discussions. This strongly suggests that the most important professional credential will probably be ability to communicate effectively with the public. The role of the typical health physicist will then have completely reversed from that of a scientist essentially working alone to that of an administrator working largely in the area of public communication.

ASSESSMENT AND EVALUATION OF NURSES TRAINING PROGRAM ON RADIATION

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INTRODUCTION

Many nurses in hospitals and clinics are concerned about the care of patients diagnosed or treated with radiation. Knowledge about radiation effects and radiological protection was in limited supply among nurses. Some nurses are anxious about the effects of occupational radiation, and they have not appropriately coped with their patients' questions about radiation effects of medical exposure.

We investigated the level of knowledge about radiation among nurses and required knowledge for nurses in hospitals and clinics. Based on the results of the investigation, we designed an education and training program of lectures and practice for nurses in hospitals. After the education and training by our program was done, we evaluated the effects of the education and training with an interview and a questionnaire for each nurse.

METHODS

Proper knowledge about radiation and radiation effects and the level of knowledge on those among nurses were investigated with questionnaires. A questionnaire with the following three categories of questions was prepared; a) degree of nurses' anxiety about radiation, b) frequently asked questions from patients about radiation and radiation diagnosis/treatment, and c) information about radiation needed for nurses in the clinical field. The questionnaires were distributed to 319 nurses of 17 departments in two university hospitals. A total number of 303 nurses (95.0%) replied to questionnaires. Based on the result, we prepared a program of the education and training for nurses worked in all hospital departments except the department of radiology . A lecture and practice by our program were performed in two hospitals.

RESULTS

A total number of 214 nurses (70.6%) had experienced taking care of radiological patients. Sixty nurses were registered as radiation workers, and their average annual effective dose in 1994 was only 0.05 (0-0.6) mSv. Two hundred and seventeen nurses (71.6%) had felt anxiety about occupational radiation exposure. Major anxieties about radiation were biological effects such as carcinogenesis and hereditary effects. Two hundred and thirteen (70.3%) nurses had an experience of being questioned by the patients. The frequently asked questions from patients were biological effects such as side effects (60.9%) of the medical exposure.

Based on the results of the investigation with the questionnaire, we planned an education and training program. Matters of interest were listed in Table 1, and these items were arranged to form a lecture and practice as shown in Table 2. The lecture and the practice in our program were applied to 95 nurses and 12 nurses, respectively.

Two weeks after the lecture and the practice, we asked all participants to tell what they had learned from the program. The results of the investigation of the effects of the lectures and

Table 1. Matters of interest

Biological effects	
Relationship between dose and biological effects	(78.8%)
Doses of patients and workers	(46.7%)
Practical protection procedure	
Protection procedure	(40.5%)
Regulation	(40.5%)
Measurement of dose	(23.0%)
Nature of radiation	
Characteristics of radiation	(41.9%)
Physical, chemical and biological action in the body	(25.1%)
Medical use of radiation	
How radiation is used in medicine	(22.7%)
Equipments and apparatuses	(15.5%)

Table 2. Subjects in the lecture and the practice

Lecture (90 min.)
Biological effects of radiation
exposure doses and dose limits
Basic procedure for radiation protection
Patient doses and optimization
Practice (60 min.)
How to use a survey-meter
Measurement of radiation in daily life
Relationship between dose and distance
Shielding effects of some materials (density, thickness)

practice are shown in Figures 1 and 2. More than 95 % of the nurses understood radiation and radiation effects through the lecture. Among the items of the lecture, the basic procedure of radiation protection was easily understood for nurses while the biological effects of radiation were difficult to understand. The useful items of the lecture for nurses were; understanding the level of occupational exposure (31.4%), practically detailed procedure to reduce external doses(18.6%), basic procedure to reduce external doses (17.1%), and biological effects of radiation (14.3%).

Of the 12 nurses who attended the practice, 7 nurses answered that the experimental practice for the shielding effect of some materials was very interesting. Also, 11 nurses answered that they understood radiation in everyday life such as radiation from luminous clocks and natural radiation.

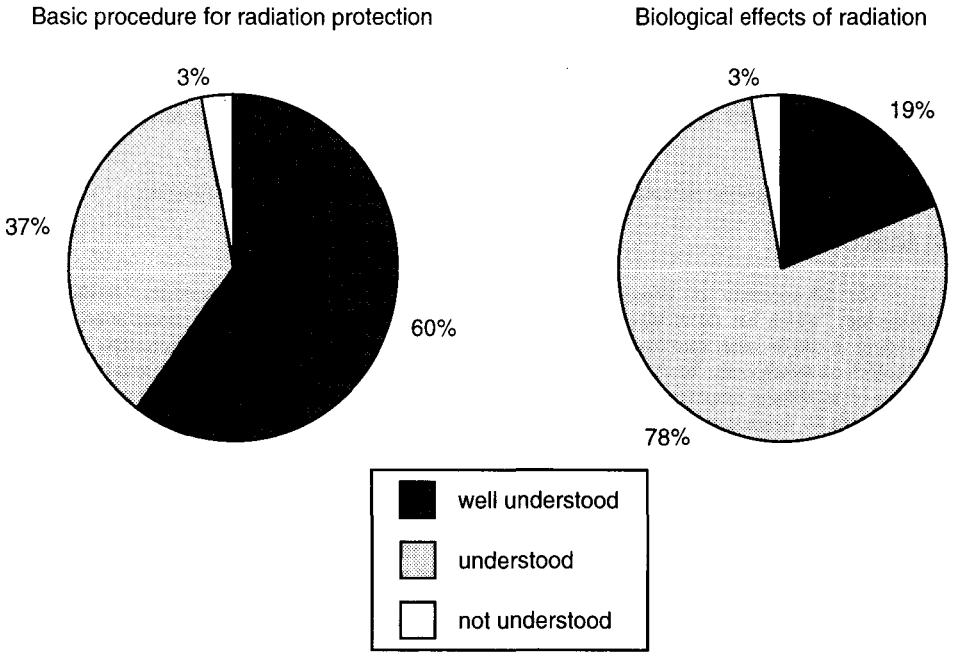

Figure 1. Understanding of the lecture

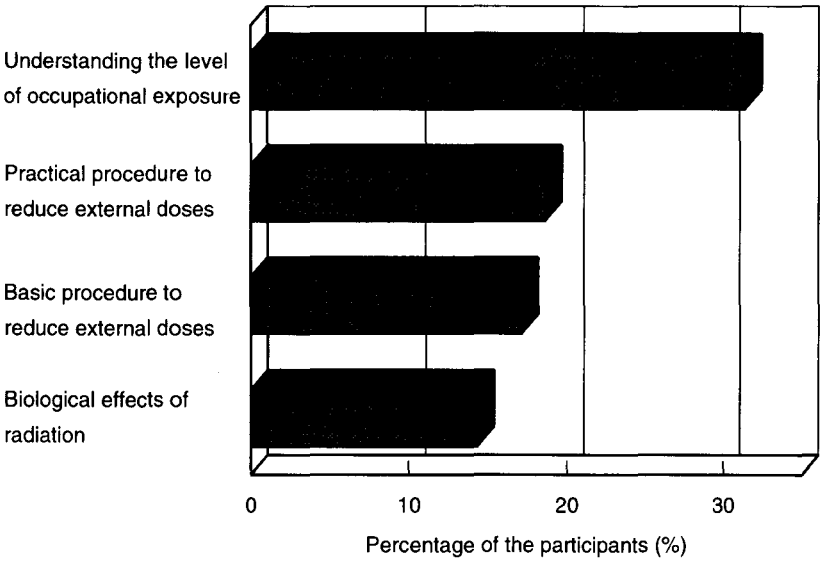

Figure 2. Useful items in the lecture

RADIOISOTOPE METHODOLOGY COURSE. RADIOPROTECTION ASPECTS.

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INTRODUCTION

The advancement of knowledge in molecular and cell biology, biochemistry, medicine and pharmacology, which has taken place during the last 50 years, after World War II finalization, is really outstanding. It can be safely said that this fact is principally due to the application of radioisotope techniques. The research on metabolisms, biodistribution of pharmaceuticals, pharmacodynamics, etc., is mostly carried out by means of techniques employing radioactive materials. Radioisotopes and radiation are frequently used in medicine both as diagnostic and therapeutic tools. The radioimmunoanalysis is today a routine method in endocrinology and in general clinical medicine. The receptor determination and characterisation is a steadily growing methodology used in clinical biochemistry, pharmacology and medicine. The use of radiopharmaceuticals and radiation of different origins, for therapeutic purposes, should not be overlooked.

For these reasons, the importance to teach radioisotope methodology is steadily growing. This is principally the case for specialisation at the post-graduate level but at the pre graduate curriculum it is worthwhile to give some elementary theoretical and practical notions on this subject. These observations are justified by a more than 30 years teaching experience at both levels at the School of Pharmacy and Biochemistry of the University of Buenos Aires, Argentina. In 1960 we began to teach Physics III, an obligatory pre graduate course for biochemistry students, in which some elementary notions of radioactivity and measurement techniques were given. Successive modifications of the biochemistry pre graduate curriculum incorporated Radiochemistry as an elective subject and since 1978, Radioisotope Methodology, as obligatory subject for biochemistry students. This subject is given at the Radioisotope Laboratory during the first semester of each year and its objective is to provide theoretical and practical knowledge to the biochemistry students, even though this knowledge does not imply any possibility to obtain a license to work with radioactive material. Pharmacy students receive since 1975 some very elementary notions of radioactivity in the Physics courses in the same Department and some notions of Radiopharmacy are given in Pharmaceutical Technology.

Since 1962 we are giving every year during a complete semester in the second half of the year, at the same Radioisotope Laboratory of the School of Pharmacy and Biochemistry, the Radioisotope Methodology Course for professionals mostly engaged in the health area. Its objective is to provide a throughout theoretical and practical knowledge for the utilisation of radioactive materials in different disciplines, particularly those related to health. The experience obtained in this course is sufficient to obtain a license from the National Regulatory Authority to manipulate radioactive material "*in vitro*".

COURSE ORGANISATION

The teaching experience acquired during successive courses given since 1962, allowed us to define a well settled complete theoretical and practical program embodying 222 training hours, 122 being theoretical and the remaining 100 are personal and intensive experimental work. It should be added that each graduate has to study for each practical activity and make a report of the obtained results; this activity is not taken into account in the foregoing chronogram. In order to estimate the degree of knowledge, the practical activities are evaluated; we take two partial examinations and a final, written and oral examination. Radioprotection aspects have a relative weighting in the evaluations of approximately 50 per cent.The program includes a first part of basic and general concepts. In the second part all the specific applications of radioisotopes to the biomedical area are included and as new methodologies appear they are incorporated. The titles of each chapter of the program are:

Nuclear stability. Binding Energy. Nuclear models. Radioactive decay mechanisms and kinetic equations. Particles and radiation interaction mechanisms. Instrumentation: ionisation chamber, proportional counters, Geiger-Müller tubes, mono and bidimensional radiochromatogram analysers, solid and liquid scintillation spectrometry, solid state detectors, radioautography (whole sample, histology, cellular), absolute activity determination. Detection efficiency. Statistics in the radioactivity measurements. Production of radioisotopes and radiopharmaceuticals. Purity control and criteria. Activation analysis. Biochemical

applications of radioisotopes: radioimmunoanalysis, receptors, radioautography, precursors and others. Medical applications: image diagnosis and radiopharmaceuticals. Radiotherapy: internal and external. Notions.

Since the objective of this paper is radioprotection, we shall analyse these points of the program with more detail. Radioprotection is one of the most important chapters of the course, since it is our experience that professionals having passed the final examination frequently will be responsible for the radioprotection aspects in his/hers work place. We emphasise from the very beginning that a throughout knowledge of radioprotection is essential for a good professional practice. The theoretical subjects are approximately a 30 per cent of the classes and include:

Definitions, magnitudes and units of: dose, dose equivalent, collective dose equivalent, committed dose equivalent, exposure and their respective rates. Internal and external sources. Shielding. General aspects of radioprotection: justification, optimisation and dose limitation. Biological effects of ionising radiation. Radiobiology. Classification of working areas. Occupational radiological protection. Internal and external contamination. Monitoring. Instruments for radioprotection. Regulation for the transportation of radioactive materials. Regulations for the utilisation of radioactive material in Argentina.

Even though the specific practical activity related to this chapter is only a 10 per cent of the total practical training, in all these activities radioprotection regulations are taken into account and discussed permanently with the graduate students. Specifically in the first practical work the rules to work safely with radioisotopes are given and discussed with the trainees and afterwards problems and shielding calculations are carried out. At the end of the course a substantial part of the final examination is the planning of an experiment with radioisotopes, taking into account radioprotection regulations as well as all the other knowledge throughout the course. The main idea of stressing the importance of radioprotection considerations is to take for granted that, at least in principle, any professional having passed the final examination and consequently licensed by the Regulatory Authority, should be capable to act as security officer.

The objective of the training in radioprotection consists in the acquisition of criteria for the adequate application of the radioprotection *philosophy*, independently on the previous university training of the graduate students participating in the course. The global synthesis include:

Planning of professional practices with an adequate *training* of the personnel involved in order to keep the doses as low as it is reasonably possible and *calculation* of the doses absorbed by the personnel during the practices. At the same time we emphasise those aspects related to the specific future activities of the graduates assisting to the course, principally ^{125}I for biochemists mainly interested in radioimmunoanalysis. In the case of physicians interested in future radioisotope applications to their profession, ^{99m}Tc is the mostly used radionuclide in Argentina. Even as its Γ value is 0.08 Rm^2/hCi, quite similar to that of ^{125}I, in the ^{99m}Tc case activities of 0.5 to 1 Ci are used, which obviously increase the dose due to irradiation.

RESULTS

Since the beginning of the course until 1995, a total of 1106 professionals have passed their examinations. It should be mentioned that the graduates are chosen not only taking into account their academic profiles but also considering the necessity of their respective working places to have well trained personnel in radioisotope techniques. We receive mainly biochemists, physicians, chemists, biologists, engineers, pharmacists and others. In the Figure 1 we show the proportion of the main professions in periods of five years as a function of time. The preponderance of biochemists and physicians is evident. Another interesting point to be noted is the increasing tendency of the number of professionals as a function of time. In the last years the number of physicians is growing steadily. This is due to the increasing expansion of medical diagnosis with radioisotopes, principally in cardiology.

The participating professionals live principally (88 per cent) in Buenos Aires and its surroundings. In the last few years the proportion of graduates from the provinces is increasing. This fact should be due to a growing interest in smaller cities to have, at least, certain minimal radioisotope using health care practices, both in medicine and biochemistry. During these years 10 professionals from different Latin-American countries (Peru, Venezuela, Colombia, Brazil and Uruguay) have passed their examinations.

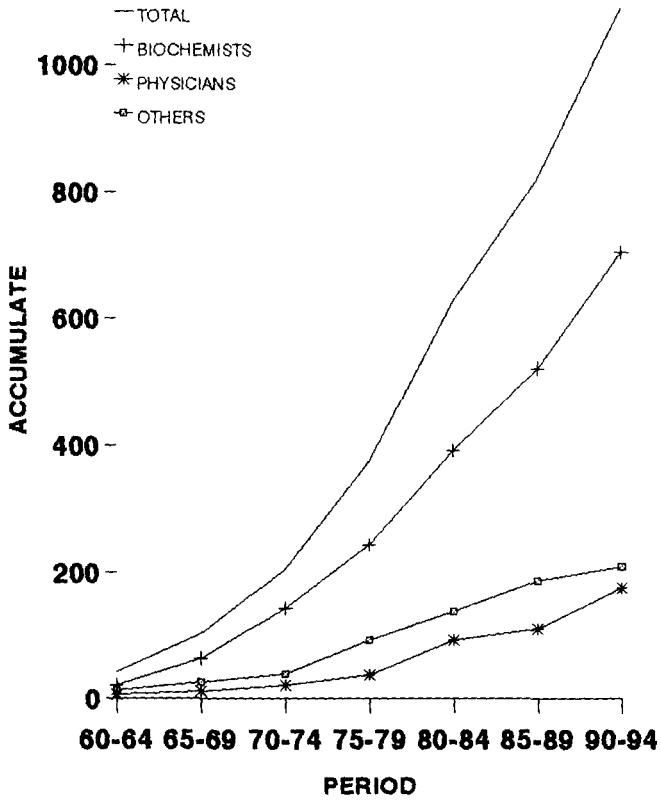

Fig. 1.- Number of professionals trained from 1960 to 1994.

CONCLUSIONS.

The application of radioisotopes and radiation to different professional activities is already a common practice in modern science and technology. However, these activities are only possible in a framework of radiological safety, if adequately trained personnel is employed. The course we describe in the present paper is organised and conceived with this purpose in mind, with a background of more than 30 years of permanent and actualised teaching activity.

RCA - A REGIONAL APPROACH TO RADIATION PROTECTION

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INTRODUCTION

The Regional Cooperative Agreement (RCA) for Asia and Oceania is the oldest of four International Atomic Energy Agency Member State regional programs. Organized in 1972, 17 countries are now members of RCA - Australia, Bangladesh, Peoples Republic of China, India, Indonesia, Japan, Republic of Korea, Malaysia, Mongolia, Myanmar, New Zealand, Pakistan, Philippines, Singapore, Sri Lanka, Thailand, and Viet Nam. A number of projects related to the application of a wide range of nuclear technologies are conducted through RCA. The program is established by national coordinators for each project area, in consultation with IAEA technical officers. Most of the funding comes directly from RCA regional donor countries, with about one third supplied through the IAEA Technical Cooperation program.

In 1986, following the Chernobyl accident, national coordinators and the IAEA staff recognized the value of establishing an RCA project aimed at strengthening regional radiation protection programs. The potential importance of RCA involvement in radiation protection is underscored by the fact that its member states comprise more than half of the world's population.

The regional approach to addressing radiation protection issues allows member states to take advantage of regional resources to solve common regional problems. RCA provides the opportunity for specialists who may have few professional colleagues in their country to develop valuable contacts with regional radiation protection experts. In a very real way, specialists can network with their neighbors, often establishing bilateral programs outside of the RCA auspices.

The current five year RCA ***Project to Strengthen Radiation Protection Infrastructures***, with the IAEA designation - RAS/9/006, will be completed at the end of 1997. The project was developed to address five major areas of activity: Off-site emergency response; Individual monitoring, internal and external; Characterization of the physical, anatomical, physiological and metabolic characteristics of the Asian populations; Regulations, with emphasis on implementation of the new International Basic Safety Standards; and Training and Education.

OFF-SITE EMERGENCY RESPONSE

The objective of this project area is to support establishment of national capabilities in RCA Member States for off-site emergency response to accidents involving radiation sources, nuclear power stations and research reactors. This need was highlighted following the Chernobyl accident. Although only four RCA Member States (China, India, Japan and Korea) and Taiwan (not an IAEA Member) have currently have operating nuclear power plants, an accident within the region is likely to have an impact on regional neighbors. Moreover, other countries within the region are currently developing plans for national nuclear power programs.

The need for an emergency response capability is not limited to nuclear power issues. World wide experience over the last few years has shown that improper use or loss of control of large sources can have fatal consequences, sometimes involving the public. Therefore, RCA work on emergency response is intended to address the potential for large source accidents as well.

The emergency response project has been established by the RCA national coordinators to address three major topics areas or outputs: Recommendations on the regulatory control, safe handling, storage and disposal of sealed sources based on IAEA Safety Series guidance (covering medical uses, nuclear gauges, irradiation facilities, and decommissioning, dismantling, storage and disposal); Development of national capabilities for handling off-site emergency plans and countermeasures; and Establishing a regional network to assist in emergency situations.

The mechanisms identified to achieve the emergency response objectives include:

- conduct of regional consultants meetings on formulation of legal requirements and development of national services to produce procedures and guidance on assistance in the event of an accident.
- workshops for national specialists on recommendations on the regulatory control, safe handling, storage and disposal of sealed sources; preparation of off-site emergency plans and counter measures assessment of dose arising from a radiological accident.

- training courses on licensing and control of sealed sources and recent developments in basic radiation protection
- intercomparison program for measurement of environmental radioactivity

On an individual basis, support for the upgrade of emergency response capabilities is provided through recruitment of experts, either from the region or outside the region, to advise on specific areas of need, and fellowships which allow people having national responsibilities to receive on-the-job experience in neighboring countries. It is often through individual contacts such as these that on-going, bilateral relationships are established with the RCA region.

INDIVIDUAL MONITORING

A major objective of the RCA project to strengthen radiation protection infrastructures is to promote the harmonization and upgrading of individual monitoring programs for occupational protection. This is seen as an area where regional cooperation can be particularly effective. The tasks under the current project address both external and internal dose assessment. For external monitoring, the program is particularly timely in that it provides an opportunity to stimulate introduction and use of the operational quantities defined by the International Commission on Radiation Protection and Units (ICRU) and endorsed in the ***International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*** (BSS) for demonstration of compliance with the prescribed dose limits.

RAS/9/006 primarily addresses dosimetry needs through regional intercomparisons and workshops. Intercomparisons offer the participants access to a number of resources for calibration and assessment of their measurement systems that would not otherwise be available to them. This, for example, may include the opportunity for calibration using x-ray fields that can provide information on the energy response characteristics of their dosimetry systems. Participants compare the performance of their systems with their regional colleagues, which can help them identify areas for improvement.

The first RCA regional personal dosimetry intercomparison for photons was conducted in three phases over three years during the period 1990 to 1992. Seventeen dosimetry service organizations from 14 member states participated in the programs. Irradiations and data compilation were conducted by Japan Atomic Energy Research Institute (JAERI). Progress over the three phases demonstrated clear improvement in the participants' understanding of the concepts and their ability to carry out dose assessment. The second intercomparison is currently underway with 37 participating services from 15 Member States. The Australian Radiation Laboratory, JAERI, Japanese Power Reactor and Nuclear Fuel Development Corporation (PNC), and the New Zealand National Radiation Laboratory are providing irradiations. Emphasis is being given to calibration in terms of the ICRU operational quantities, including use of the 30cm x 30cm x 15cm water filled backscatter phantom now being recommended in draft guidance of the International Standards Organization (ISO).

Regional workshops bring together experts and those responsible for establishing and managing various elements of occupational monitoring programs. In 1994, RAS/9/006 sponsored two highly effective workshops on external dose assessment and calibration techniques for external photon irradiations. Again, a major thrust of these workshops was calibration and individual monitoring in terms of the ICRU operational quantities. Expert presentations, laboratory exercises and open discussions with participants helped clarify the principles involved in use of the "new" quantities and allay fears about their implementation. In particular, the workshops were valuable precursors to the second RCA regional intercomparison for individual monitoring. In 1996, another workshop is planned on contamination monitoring.

Additional RCA regional activities include tasks intended to address internal dosimetry and biological dosimetry. One task is aimed at the need to establish regionally harmonized measurement techniques and dosimetry for radon and thoron. A workshop on use of physical phantoms and mathematical models for occupational radiation protection is also being planned for 1997.

REFERENCE ASIAN MAN

Remembering that the RCA Asian Member States represent more than half of the world's population, the national coordinators identified the need to compile a database on the physical, anatomical, physiological and metabolic characteristics of the Asian populations for radiation protection purposes. The compilation process was initiated as an IAEA Coordinated Research Program (CRP) in 1988 under funding by the Japanese government. Eleven RCA Member States participated in the CRP, collecting national data in four major areas:

- Anthropomorphic measurements
- Organ mass measurements
- Nutritional and dietary intake
- Pulmonary function and water balance

Data was compiled on over 1,000,000 people. Results show a range in anthropomorphic and organ mass characteristics within the Asian region that is nearly as great as the difference between the Asian and western populations. Moreover, in some countries such as India there is a wide national diversity due to contributions of ethnic sub-populations, while others such as Japan are quite homogeneous. It is clear that each country in the region will need to analyze its own situation and use the available data as appropriate to national needs.

The extent and quality of data on nutritional and dietary intake, and pulmonary function and water balance were not as extensive and complete as for the anatomical parameters. This has led to initiation of a second CRP which will focus on analysis of trace elements important to radiation protection in national dietary samples and selected tissues. A more detailed report on the CRP appears elsewhere in this proceedings.

REGULATIONS

Publication of the ***International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources*** has brought a focus on the requirement to harmonize implementation of the new recommendations. The RCA offers an excellent mechanism to address this need in the Asian region through regional workshops, seminars and training programs. A specific task has been established to facilitate BSS implementation with the objective of assisting Member States in understanding and implementing the recommendations of BSS in the region. Activities under this task include: Regional training on implementation of the Basic Safety Standards and National seminars on the consequences of the introduction of the BSS and their implementation on a national and regional basis.

TRAINING

The RCA program for strengthening radiation protection infrastructures offers the opportunity to develop and upgrade the quality of RCA Member State training and education programs on a regional basis. The major activity in this area is the development of distance learning materials for radiation protection. The distance learning material program, which has been, developed by the Australian Nuclear Science and Technology Organization (ANSTO) and supported by the Australian government, recognizes the difficulty that is often encountered in providing effective national training in countries where resources are separated from those who need the training by great distances. The distance learning materials have been developed in a modular format to enhance flexibility in their use. Topics range from basic radiation interaction and detection principles to transport of radioactive material and waste management. Each module includes question sets to assess the student's comprehension. The materials are designed to be used and administered by a facilitator or mentor.

Recently, a regional seminar to promote education and training in radiation protection and nuclear safety was conducted under joint sponsorship of the RCA and the regular IAEA Technical Cooperation training program. Although not strictly an RCA activity, the regional seminar brought together training specialists from RCA member states to discuss national training needs, approaches and solutions. A clear outcome of the seminar was the potential for using a regional approach to meet training needs.

SUMMARY

The IAEA/RCA ***Project to Strengthen Radiation Protection Infrastructures*** brings together 17 countries in Asia and Oceania to address radiation protection issues of common interest on a regional basis. Activities within the Project combine technical resources from within the region and interregional expertise to conduct training activities, workshops, Co-ordinated Research Programs (CRPs), expert advisory group meetings etc. Emphasis is placed on use of regional radiation protection experts to respond to RCA country needs, and local host institutions for fellowship training. Although RCA communications begin with the network of national coordinators, networks of specific technical specialists have evolved to expedite information exchange in emergency response and dosimetry. The RCA has become an excellent vehicle for developing a regionally harmonized approach to implementation of the International Basic Safety Standards.

MANAGEMENT AND TRAINING ASPECTS OF THE EMERGENCY PLAN

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INTRODUCTION

The main objectives of an emergency management system are to prevent or reduce the likelihood of consequential loss in the event of an emergency occurring. In the event of a nuclear accident the effectiveness of measures for the protection of the public will depend on the advance preparation especially in education and training. This paper reviews two recent initiatives and concludes with comments on the future development of this subject.

There is an increasing requirement in legal and moral terms for industry to inform the population of health hazards to which they are exposed. In a report (1) published by the Nuclear Energy Agency (NEA/OECD) radiation protection was described as a subject which is impenetrable to the layman and as wide as it is complex. For this and other reasons radiation hazards are perceived to exceed all others and the public appear to have a poor image of the radiation protection specialists. Communication with the public and the media is widely recognised (2) as a key part of an emergency plan. This view is supported in the European Union which has sponsored the book on "Radiation and Radiation Protection - a Course for Primary and Secondary Schools" (3) which is described in this paper.

The training of emergency teams includes the use of drills and exercises (4) to maintain skills and can also be used to test the adequacy of plans. Every effort should be made to simulate the pressure on time and resources which would occur in a real event. Radiation emergencies are fortunately rare and so there is little practical experience of these events. The emergency worker must gain some radiation protection skills and must be able to use some technical language when communicating with specialist advisors. For this reason the European Union has sponsored the book "Radiation Protection for Emergency Workers" (5) which is also described in this paper.

A COURSE FOR PRIMARY AND SECONDARY SCHOOLS

In the European Union the Council Directive (6) on public information was adopted in 1989. This emphasised that the public should understand the issues involved and be prepared in advance of the emergency. The Commission organised a Seminar (7) in 1988 and the outcome was a suggestion that teaching material on radiation protection should be developed for use in Schools of Member States. The Directorate General XI (DG XI) (8) developed a first draft of the Teacher's Manual. A "spiral curriculum" was chosen because radioactivity, ionizing radiation and non-ionizing radiation are fairly complex and abstract subjects in particular for younger and less advanced pupils. This means that items recur in a gradually more complex form. The material was set out in five levels covering ages 6 - 16. In the first three levels emphasis is placed on relating the pupils personal and every day experiences and they are made aware of the risks and benefits of ionizing radiation. In the final two levels a more detailed examination is made of the subject from both a technical and social point of view. The original text was tested by the University of Utrecht and edited into a convenient loose leaf format. A test printing of the text was produced in 1993 and presented to a group of European Communities experts together with a formal presentation of the survey results. Reports were given by

teachers involved in the testing and the two original authors responded to the comments presented their views. The course was modified in response to recommendations made at this meeting. New Chernobyl information was added and potentially confusing material on the "green house effect" was deleted. The first edition in loose leaf format was published in 1994. This will not be the end the story; success will depend crucially on the sharing of the experience gained by teachers who choose to use this book.

RADIATION PROTECTION FOR EMERGENCY WORKERS

This CEC book (4) is intended to support emergency team training and introduces the principles of radiation protection and the properties of radioactive materials. Radiation doses, their consequences and principles of radiation detection are explained. Guidelines are given for protection against radiation from external sources and from internal sources. The annex to this paper gives an example of the guidelines used in the book. A number of radiation emergencies are described to provide a basis for the lessons learned. The booklet concludes with appendices on the atom, on nuclear fission and the regulations for transport of radioactive materials. A list of radionuclides and a list of SI prefixes precedes a glossary which covers the terms used in the book and related topics.

FUTURE ACTIONS

Education. Experience of exercises and incidents occurring in the nuclear energy programme reveals that there is sometimes a gap (9) between the desired response from the engineering team and the resources available. This is where professional skills and engineering judgement are demanded, and where precisely defined training schemes and rule driven response is likely to fail. This gap probably occurs in the reponse to many other emergency situations and may be a neglected area of the engineer's education. A new syllabus has been published for an undergraduate awareness course emphasising the engineer's responsibility for health and safety of the public (10). This duty of care to the towards the public should also apply to the engineer's rôle in the response to emergencies and so an extension to this syllabus is proposed.

Performance under stress. Experience in the conduct of exercises and in real emergencies suggests that emrgency response teams do improve their performance with practice and experience. Performance under stress demands a cool heads which probably can be acquired in training. Correlation between training performance standards and the real thing is difficult, more research is needed especially in the study and alleviation of stress.

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ANNEX - GUIDELINES ON PERSONAL DOSE REDUCTION *.

Emergency personnel must take precautions to protect themselves against all hazards at the site of an emergency. These guidelines are provided as an aid to the application of the ALARA principle in emergency response. Seven steps for your protection against radiation are summarised below:-

1. See the Source - make every effort to identify the type of radiation hazard. Search for the trefoil symbol, recognise the shape of standard sources and their containers, get expert advice, notify proper authorities.
2. Seek containment - do not move a damaged source, avoid breaking protective enclosures, close doors and windows. Cover all exposed skin. Do not eat, drink, smoke, rub eyes or apply skin preparations such as sun tan lotion and cosmetics.
3. Stop Spread - monitor and cover or decontaminate surfaces, do not spread activity by your own movements and set up a barrier or control line at a safe distance out of range of the source.
4. Shorten Exposure - keep exposure time as short as possible, reduce the time of working close to the source, move away into a clean or low dose rate area to think, to discuss. Rehearse all actions.
5. Step away - keep as far away as possible, never touch the source, use appliances such as long handled tongs, view the scene through a telescope, use a mirror to see round a corner. Position vehicles and command post up-wind, out of smoke and away from liquids.
6. Shadow Shield -install a radiation shield which is as thick and as heavy as possible, place it as close to the source as possible, work behind a shield, make use any available structure as a shield.
7. Self check - check your dosimeter frequently, check your own clothing for radioactive contamination at the barrier, look out for non-radiation hazards. Communicate with the controller.

* extract from reference 5

Early radiation protection - the Austrian contribution

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Abstract

Immediately after the discovery of x-rays and radioactivity. an avalanche of applications and development developed also in Austria. Also immediately, deleterious effects of radiation were observed and hence the precursor of the profession of "radiation protection" was founded.
This paper presents some documents and information on early activities in radiation protection, as done in the countries belonging to the previous Austrian empire. The examples include information on some scientific background on radiation effects at this time, the worldwide first licensing procedure in 1899 and some early environmental radioactivity measurements. A pre-1895 epidemiological study is also shown as well as very early papers associated with the field of RADIOECOLOGY.

Radioactivity

Pre-1896, the only possible sources of radioactivity leading to an exposure with radiological consequences were daughter products of radium. As we know today, substantial concentration of radon may take place in mines, and a strange disease only among miners was described even some centuries ago. Famous names as Agricola (1500, cited by /1/) and Paracelsus (1567, cited by /2/) are reported. One area with a high concentration of valuable material and hence mining was the region "Erzgebirge" at the border between Saxony and Bohemia, where Bohemia was at this time part of the Austrian empire. Therefore, both the "Schneeberger (Saxony)" and the "Joachimstaler (Bohemia)" "Bergkrankheit, mountainous disease" were found among miners with the roughly the same incidence /3/. The disease remained unidentified until /4/ proved an enhanced incidence of lung cancer is associated with mining. Later, the relation between Radon concentration and enhanced incidence of lung tumours was proved, and numerous papers were considering this question after the discovery of radioactivity

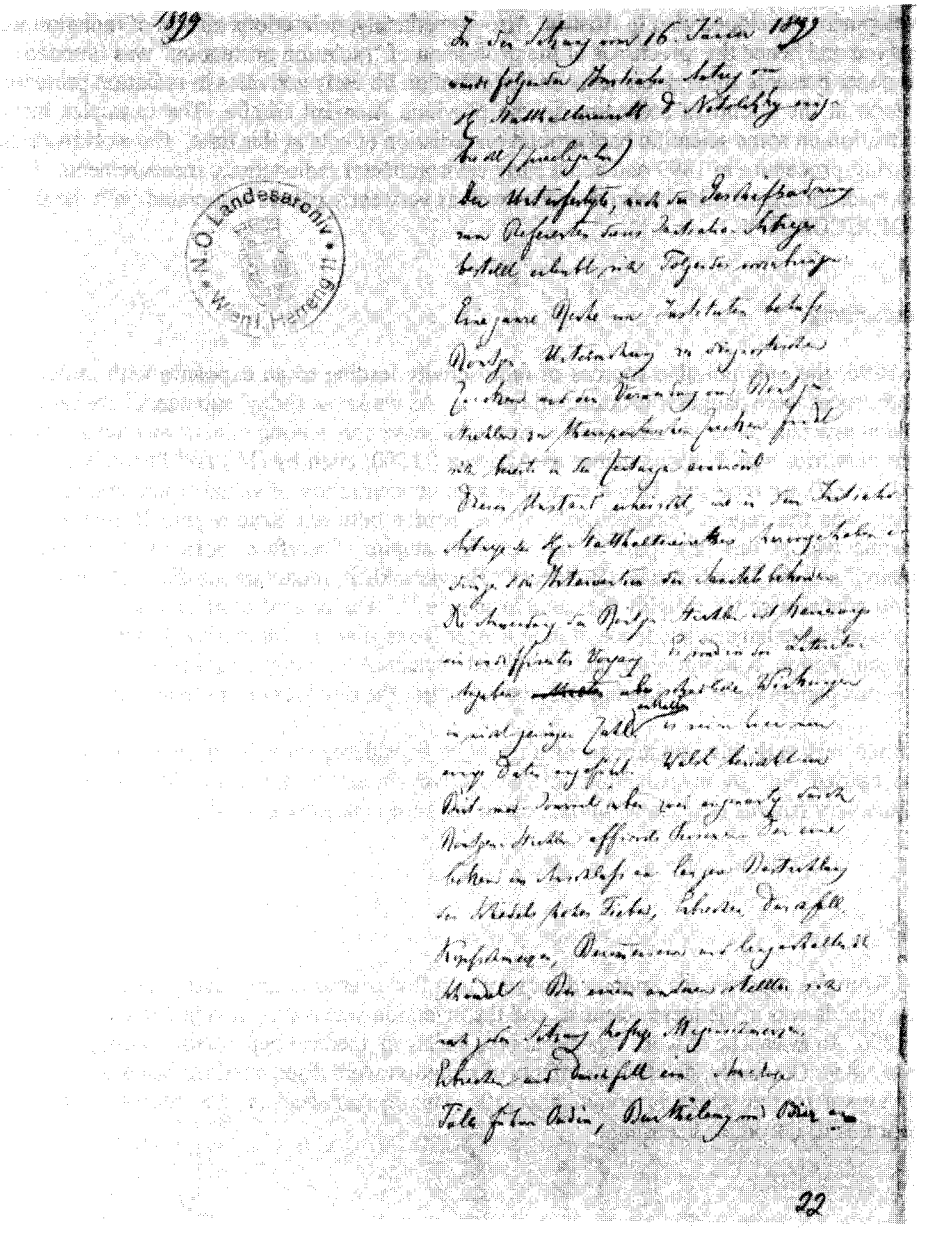
Another very early investigations which might be considered today as roots of "radioecology" were carried out by investigations of radioactivity in air /5/ and water/6/ by scientist who become very famous later. In addition, cosmic rays were also discovered /7/

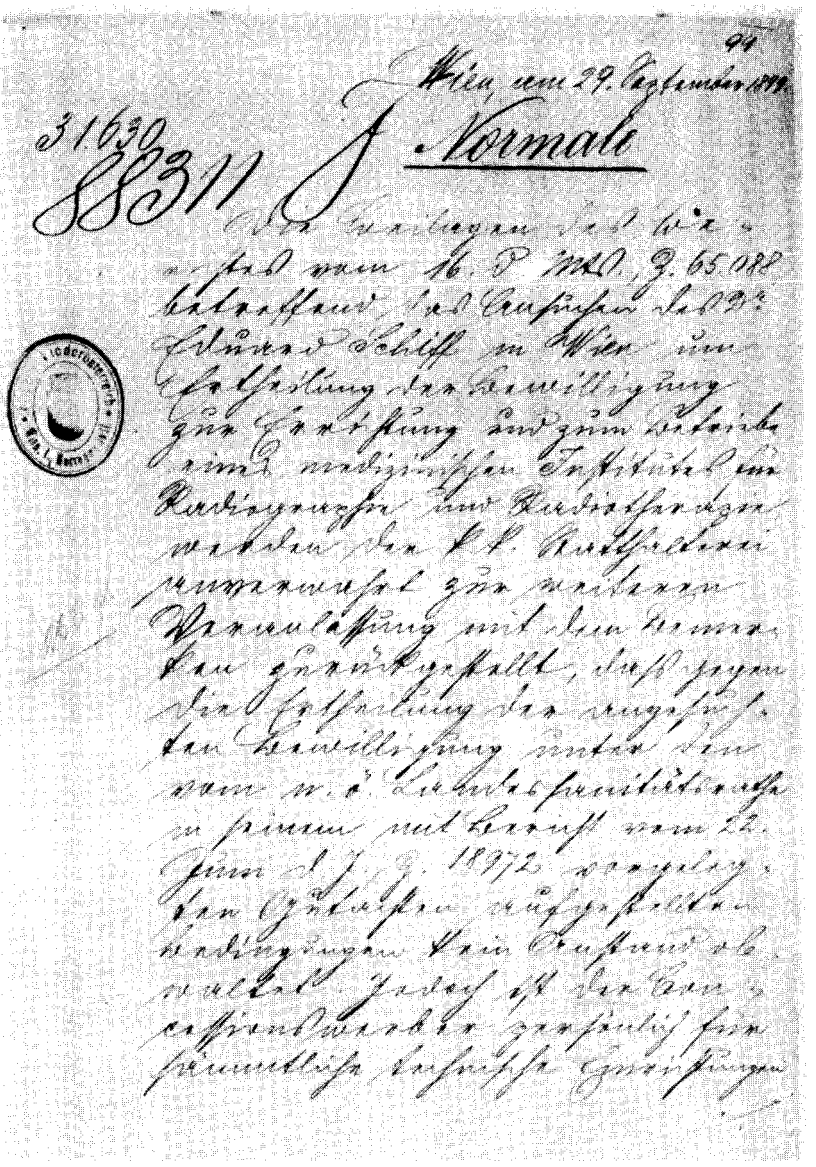
X-rays

W.C.Röntgen reported his famous discovery to a few friends before final publication. Among these friends was a Viennese scientist, and the invention was soon published in a newspaper at 5.1.1896. An avalanche of work was developed, mainly in medical application as diagnostics and therapy /8,9/. Obviously, deleterious effects of radiation were discovered immediately and were well aware to the scientific community /10/. This in turn lead to the interest of the local authorities to introduce licensing.

The worldwide first licensing procedure was started in 1898/10/. This was a formal inquiry of the administration of Lower Austria (*k.k.NÖ Statthalterey*) to the muncipiality of Vienna (*Wiener Magistrat*) to check what is going on in Drs Schröder und Ehrenfeld`s *Institute für Röntgenuntersuchungen*. This is because x-rays are still called Röntgenstrahlung in German speaking countries. This action lead after a meeting of the medical council and a statement of a medical expert, who was in a position to give a well proved statement of the deleterious effects of radiation very soon to a licence, The licence for the institute was issued in October 1899 including some *cautelen* (protective measures as not to irradiate the same part of the body frequently).

Some examples of the early documents are presented below





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/3/ H.Sikl: Zeitschrift für Krebsforschung **32**(1930)610
/4/F.H.Härtling, W.Hesse: Vierteljahresschrift für gerichtliche Medizin, Neue Folge Vol. XXX (1879) 296- 309 and Vol. XXXI(1879)102-132 und 312-337
/5/ E.Schrödinger: *Sitzungsberichte der Österr. Akademie der Wissenschaften* CXXII Dec.1913
/6/M. Bamberger: *Sitzungsberichte der Österr. Akademie der Wissenschaften* CXVI Dec.1907
/7/ V.F.Hess, R.Steinmaurer: *Helv.Acta* III (1929)
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/10/M.Tschurlovits, P.Karacson: *Appl.Radiat.Isotopes* **37**(1986)373

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